OKLO

Part II: Final Safety Analysis Report

THE SAFETY CASE

Safety by design

Oklo designed the Aurora according to the high-level Aurora safety principles, which identify high-level safety goals and operational goals (Chapter 5), as well as principal design criteria (Chapter 4). These principles and criteria guided the design from the beginning of the design process.

Safety and defense-in-depth are fundamentally accomplished in the Aurora design by its inherent characteristics, including:

- Small size, low power output, low power density, and low decay heat output
- Low fuel burnup, small inventory of fuel, and limited available source term
- Low decay heat term, removed by inherent and passive means
- High thermal conductivity materials reduce temperature hot spots, and large thermal mass provides capacity for heat dissipation
- Inherent reactivity feedbacks ensure reactor power is controlled during overpower or overtemperature events
- Multiple barriers to fission product release
- High thermal conductivity materials reduce temperature hot spots, and large thermal mass provides capacity for heat dissipation
- Ambient pressure system removes sources of pressure and limits driving forces for release
- No use of water cooling, uses dry heat rejection instead

The Aurora produces 4 MWth, which is far smaller than any commercial reactor in the U.S., and smaller even than some research reactors. The lower power of the Aurora also leads to low decay heat production. For reference, 30 seconds after shutdown, the reactor is generating 153 kW of decay heat, 30 minutes after shutdown, the reactor is producing 65 kW of decay heat, one day after shutdown the reactor is producing 21 kW of decay heat, and 1 month after shutdown, the reactor is producing 7 kW of decay heat. This small amount of heat is generated and dissipated in a module that contains tens of tons of metal, and tens more tons of shielding and other structural material.

For comparison, one fuel assembly at Diablo Canyon produces approximately 4 times as much power as the entire Aurora core, and contains approximately 0.5 tons of fuel, cladding, and structural material (or about 0.6 tons when the assembly is immersed in water and the water mass is included).

The safety case methodology

The safety of the Aurora is communicated in this Final Safety Analysis Report (FSAR) as a "safety case." The safety case for the Aurora powerhouse presented in the FSAR utilizing the historical standard of a systematic search for a maximum credible accident (MCA), analyzing and utilizing precedent for historical plant methodology as well as in-depth internal and external event analyses, to identify the worst credible accident based on the single worst credible failure or single worst credible common cause of failures. Background on the MCA is given in Section 5.1.1.

The internal event types analyzed, with relevant discussion incorporated into Chapter 5, include:

- Generic events to all nuclear reactors
- Metal-fueled fast reactor events and operating experience
- Compact reactor operating experience and analytical methods
- Review expert opinion on similar conceptual designs
- Light water reactor events and methodology, including:
    - Increase in heat removal by the secondary system
    - Decrease in heat removal by the secondary system
    - Decrease in reactor coolant system (RCS) flow rate
    - Reactivity and power distribution anomalies
    - Increase in reactor coolant inventory
    - Decrease in reactor coolant inventory
    - Radioactive release from a subsystem or component

Common cause failures due to 36 external hazards were analyzed, with relevant discussion included in Chapter 1. Some external hazards have chapters dedicated to their discussion. Seismic analysis is described in Chapter 7 and Fire Analysis is described in Chapter 6.

The maximum credible accident is identified based on iteration between these analyses and is then analyzed with added defense-in-depth from risk analysis. The analysis of the MCA with the addition of defense-in-depth is presented to show that the Aurora is safe, even in events beyond those previously licensed against. There is no credible accident within the site envelope which leads to a release.

Then, inherent safety features or inherent design parameters that are assumptions in these safety analyses are codified into design bases, assured by design commitments, and tied to programmatic controls as described, by system, in Chapter 2. Programmatic controls include quality assurance, license conditions (Part VI of the application), pre-operational testing and startup testing (Chapter 14 of this Part), ITAAC (Part VI of the application), and technical specifications (Part IV of the application).

The result is a robust and integrated system for describing and assuring safety from design, to the as-built and operated plant. The intent of this FSAR is to describe the safety case through this holistic methodology. The inherent safety characteristics of the Aurora – such as its very small size and inventory, very low power density, low burnup, robust fuel, and not requiring water for cooling – not only affect the safety analysis but also other portions of the application such as the environmental analysis and security analysis.

Introduction to the Aurora design

Oklo designed the Aurora with inherent and passive safety features. The description of all structures, systems and components is given in Chapter 2. Oklo used engineering practices and insights from deterministic analyses, defense-in-depth practices, and risk analyses to design the Aurora. The Aurora has large operating and design margins, and relies on passive and inherent characteristics to ensure materials are contained and heat is removed.

The Aurora operates with a low power density and is thermally connected via conduction to large thermal masses provided by structures and shielding. Heat is normally transported from the fuel to the power conversion system via heat pipes which carry heat from the fuel to the power conversion heat exchanger. During normal shutdown operations, residual heat is removed via the power conversion heat exchangers. However, the low power density and large thermal mass allow heat to be removed from the fuel by conduction throughout the system, and to the boundary of the shell where it is removed by convection, radiation, and conduction to the environment without the use of the power conversion system heat exchangers. This means fuel temperatures can remain below operating limits relying purely on conduction, convection, and radiation.

The Aurora has negative reactivity feedbacks due to thermal expansion of the fuel and structural materials, as well as doppler broadening. These feedbacks ensure reactor stability during operations and can help shut the reactor down should the reactor rise in temperature. Furthermore, the Aurora uses multiple, independent and redundant reactivity control systems. These include three rotating control drums, and three shutdown rods. Only one rod is needed to shutdown the reactor.

The Aurora is very small reactor with low fuel burnup, which results in a small inventory of radionuclides. In addition to a small inventory, the Aurora has multiple layers and barriers to prevent the release of radionuclides. The fuel matrix is the first barrier. The low burnup of the Aurora design, and the characteristics of metal fuel mean that most radionuclides remain in the fuel matrix over the course of the fuel lifetime. Next, the fuel is surrounded by a steel envelope, called the cell can. The cans are then placed in a capsule, which is a steel vessel that houses the cells. The capsule is sealed and placed within the shell, another steel vessel which houses shielding and structures, as well as the capsule. The shell is also sealed. The shell is emplaced in the reactor emplacement in the basement of the Aurora powerhouse. The Aurora does not operate at elevated pressures, and there is not a source of driving pressure in the core. Altogether, these barriers provide defense-in-depth to the release of radionuclides to the environment.

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1 SITE ENVELOPE AND BOUNDARY

1.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(1) requires the following be submitted:

(i) The boundaries of the site;

(ii) The proposed general location of each facility on the site;

(iii) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site with appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated;

(iv) The location and description of any nearby industrial, military, or transportation facilities and routes;

(v) The existing and projected future population profile of the area surrounding the site;

(vi) A description and safety assessment of the site on which the facility is to be located. The assessment must contain an analysis and evaluation of the major structures, systems, and components of the facility that bear significantly on the acceptability of the site under the radiological consequence evaluation factors identified in paragraphs (a)(1)(vi)(A) and (a)(1)(vi)(B) of this section. 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. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that:

(A) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).

(B) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE ;

Table 1-1 shows where each 10 CFR 52.79(a)(1) requirement is addressed in this chapter.

Table 1-1. Organization of Chapter 1

Section	Section title	Requirement
1.1	Proposed site overview	10 CFR 52.79(a)(1)(i)
		10 CFR 52.79(a)(1)(ii)
		10 CFR 100.21(a)
		10 CFR 100.21(b)
		10 CFR 100.21(h)
1.2	Evaluation of the proposed site	10 CFR 52.79(a)(1)(iii)
		10 CFR 52.79(a)(1)(iv)
		10 CFR 52.79(a)(1)(v)
		10 CFR 52.79(a)(1)(vi)
		10 CFR 100.21(c)
		10 CFR 100.21(d)
		10 CFR 100.21(e)
		10 CFR 100.21(f)
		10 CFR 100.21(g)
		10 CFR 100.23(c)
		10 CFR 100.23(d)
1.3	Safety assessment of the proposed site	10 CFR 52.79(a)(1)(vi)
Appendix A	External hazards evaluation	10 CFR 52.79(a)(1)(iii)
		10 CFR Part 100, Subpart B
Appendix B	Generic site envelope	None

The purpose of this chapter is to provide a description of the proposed site and to evaluate any safety impacts of the proposed site on the Aurora.

1.0.1 Definitions

The following terms are specific to the Aurora and are used throughout this chapter.

Aurora INL site: The area that is used for the siting of the Aurora, usually referred to as the "site boundary." The only significant building onsite is the Aurora powerhouse. Other small structures include microgrid interconnections and the backup thermal storage. This site is leased from Idaho National Laboratory (INL) as per the Site Use Permit. It is possible for the public to walk on the site and up to the Aurora powerhouse. The area within the site boundary may be frequented by the public and encompasses the owner-controlled area and parking lot. This term refers to the specific site that will be chosen from the candidate sites.

candidate sites: The sites, determined as per the Site Use Permit, that could be locations for the Aurora.

Idaho National Laboratory (INL): The part of the U.S. Department of Energy's complex of national laboratories, which is headquartered in Idaho Falls, ID. The mission of INL is to discover, demonstrate and secure innovative nuclear energy solutions, other clean energy options and critical infrastructure.

INL Site: The INL Site is an administratively controlled area with access to facilities granted for official business. The INL Site contains nuclear facilities is located in the desert, about 45 miles away from Idaho Falls, ID. Public access is allowed on the highways, the Big Lost River rest area, and at the Experimental Breeder Reactor No. 1 (EBR-I) visitor center. It is the area controlled by the Department of Energy and comprises Idaho National Laboratory.

Aurora powerhouse: The only building onsite. The Aurora powerhouse houses the reactor module and secondary system as well as other supporting equipment. The Aurora powerhouse makes up the exclusion area, low population zone, and emergency planning zone.

1.1 Proposed site overview

The purpose of this section is to satisfy the requirements of 10 CFR 52.79(a)(1)(i)-(ii), 10 CFR 100.21(a)-(b), and 10 CFR 100.21(h). This section describes the proposed site and the boundaries of the proposed site.

1.1.1 Description of the proposed site

An Aurora will be sited at Idaho National Laboratory (INL) Site in southeast Idaho, which is referred to as the "Aurora INL site." The Oklo Inc. site use permit request was evaluated by the Department of Energy Office of Nuclear Energy (DOE-NE), a field office of the DOE, through the site use permit process and received a permit on September 26, 2019. This Site Use Permit grants Oklo personnel access to the land on the INL Site leased to Oklo Inc. Oklo Inc. subsidiary Oklo Power LLC (Oklo Power) will own and operate the Aurora at the INL Site.

One of the missions of DOE is to advance nuclear power as a resource capable of meeting the Nation's energy, environmental, and national security needs by resolving technical, cost, safety, proliferation resistance, and security barriers through research, development, and demonstration as appropriate. Idaho National Laboratory, previously known as the National Reactor Testing Station (1949), the Idaho National Engineering Laboratory (1977), and the Idaho National Engineering and Environmental Laboratory (1997), has been the home to 52 reactors, with 3 currently operating. Since the main objective for a first-of-a-kind Aurora is to optimize the design, INL Site is an optimal location for a first-of-a-kind reactor. The lessons learned from the Aurora INL site will allow for the proliferation of advanced reactors in the U.S. in places needing reliable, affordable, and carbon-free electricity, while promoting new U.S.-based reactor designs worldwide.

Idaho National Laboratory is approximately 890 square miles and is located in 5 separate counties in eastern Idaho including Butte County, Bingham County, Bonneville County, Clark County, and Jefferson County. The Aurora INL site is in Bingham County, approximately 35 mi northwest of Idaho Falls as can be seen in Figure 1-1.

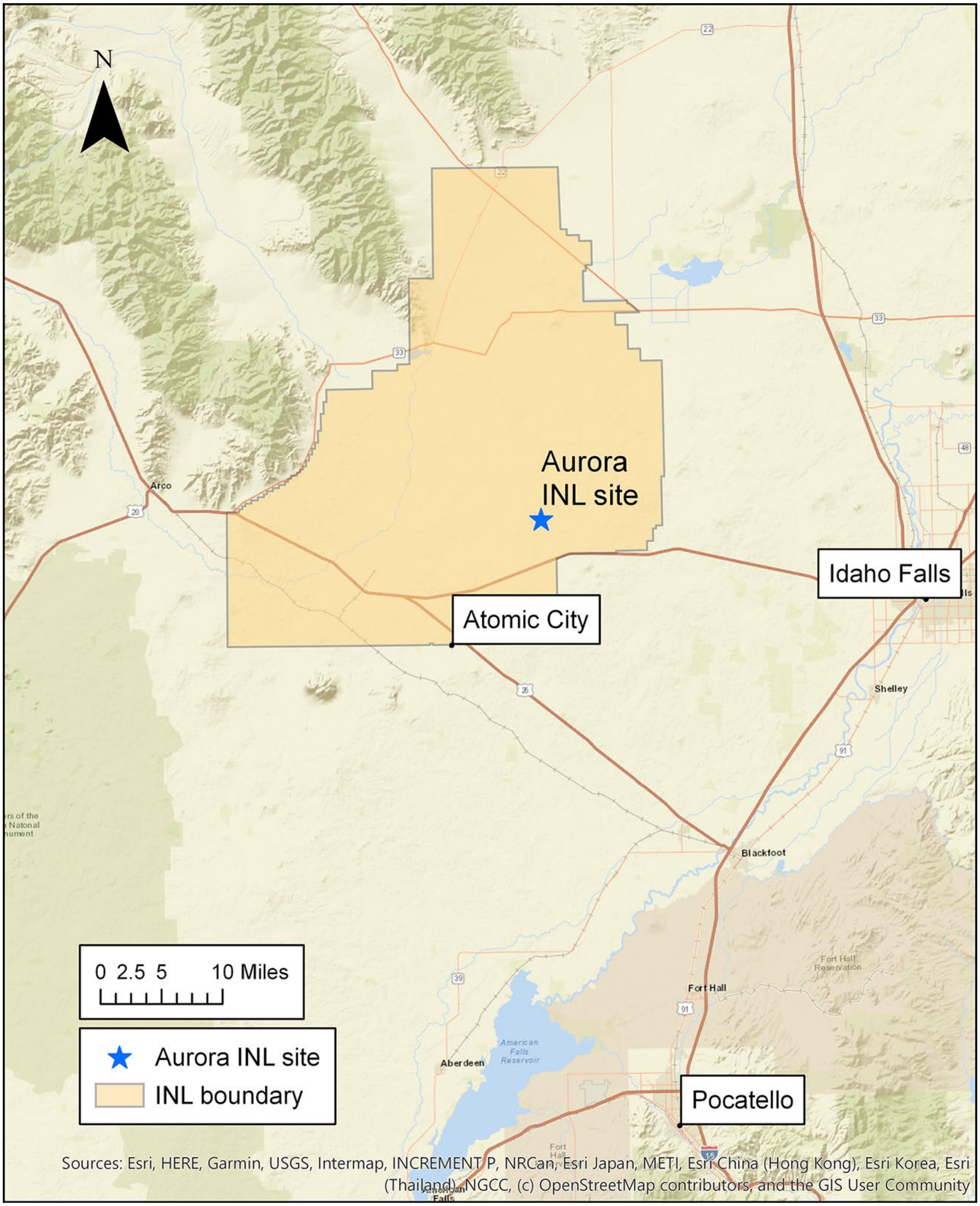

Figure 1-1: Aurora INL site and surrounding area

More specifically, the Aurora INL site will be by the Materials and Fuels Complex (MFC) at INL, which is located close to the southeast corner of the INL Site, as shown in Figure 1-2. There are six major facilities located at the MFC: (1) the Hot Fuel Examination Facility,

(2) the Experimental Fuels Facility, (3) the Irradiated Materials Characterization Laboratory, (4) the Analytical Laboratory, (5) the Fuel Manufacturing Facility, and (6) the Transient Reactor Test Facility. Most famously, EBR-II and the supporting fuel cycle facilities operated at MFC from 1961 to 1994. A layout of the facilities in the MFC can be seen in Figure 1-3.

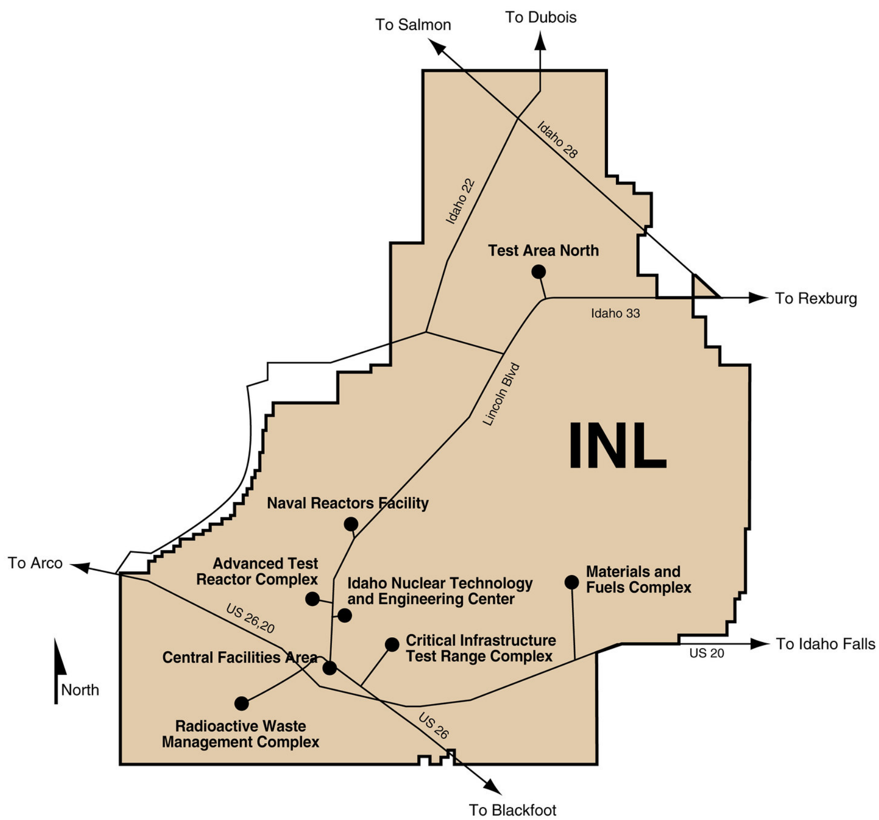

Figure 1-2: Key facilities on the INL Site [1]

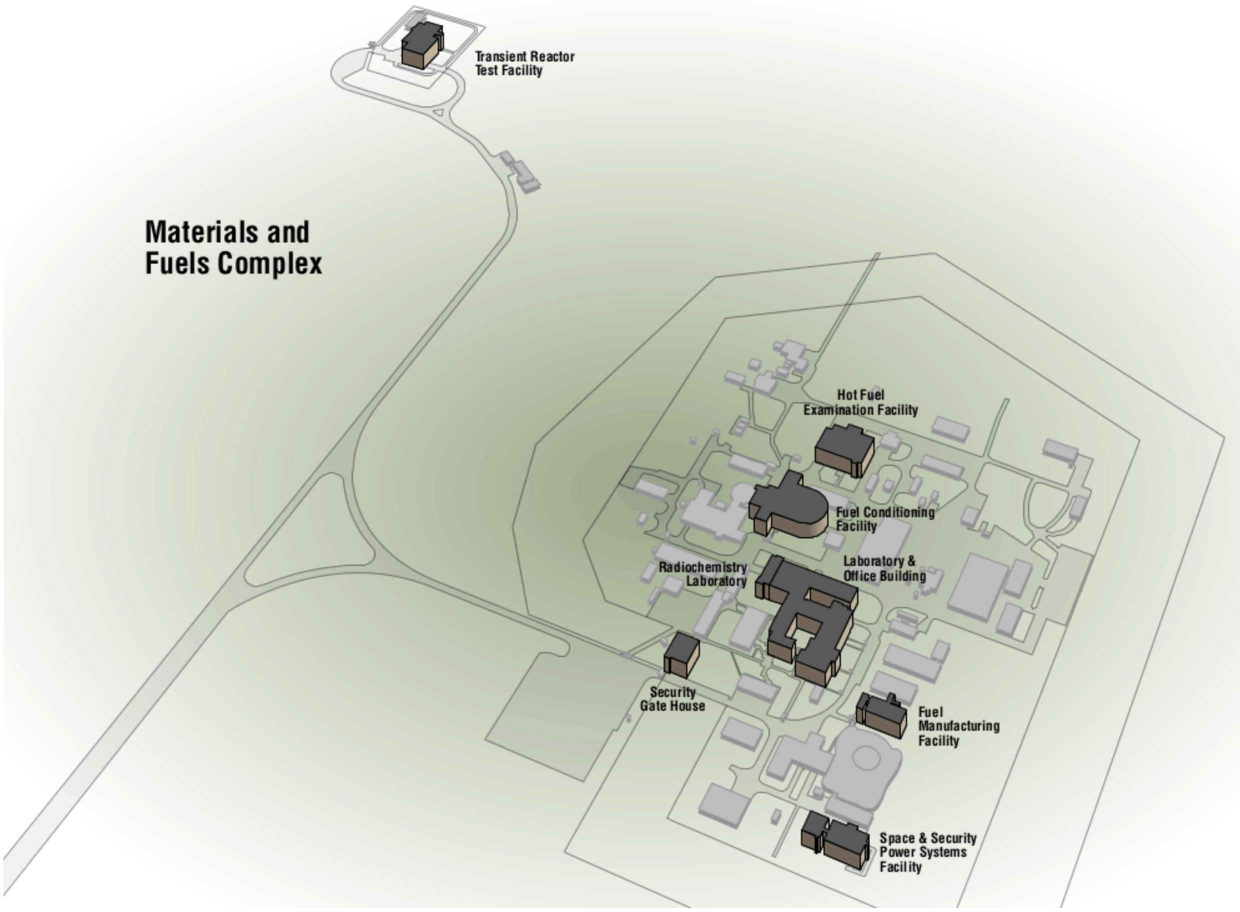

Figure 1-3: MFC layout [2]

The Aurora INL site will be chosen from five candidate sites, each of which is located on a dry flat desert on a bed of basalt within roughly one mile of the fence surrounding the MFC as shown in Figure 1-4. The MFC is located at latitude 43º 35’ 40” N and longitude 112 º 39’ 17” W.

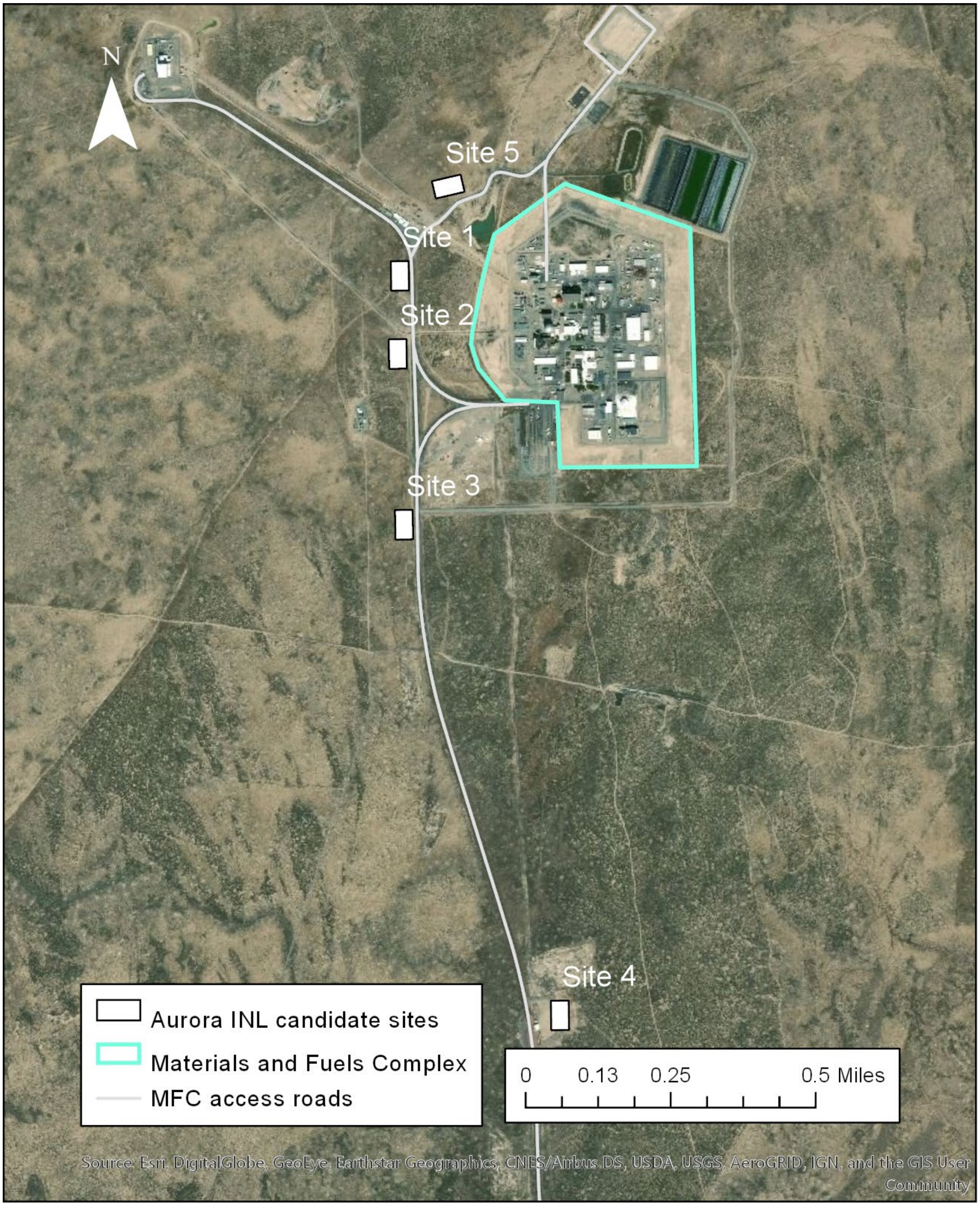

Figure 1-4: Location of the candidate sites

The exact location of the Aurora INL site will be determined by Oklo Inc. and DOE-NE, through the Site Use Permit process. Figure 1-4 and Figure 1-5 show the candidate sites, which are labeled Site 1-Site 5. These five sites are the five potential locations for siting the Aurora.

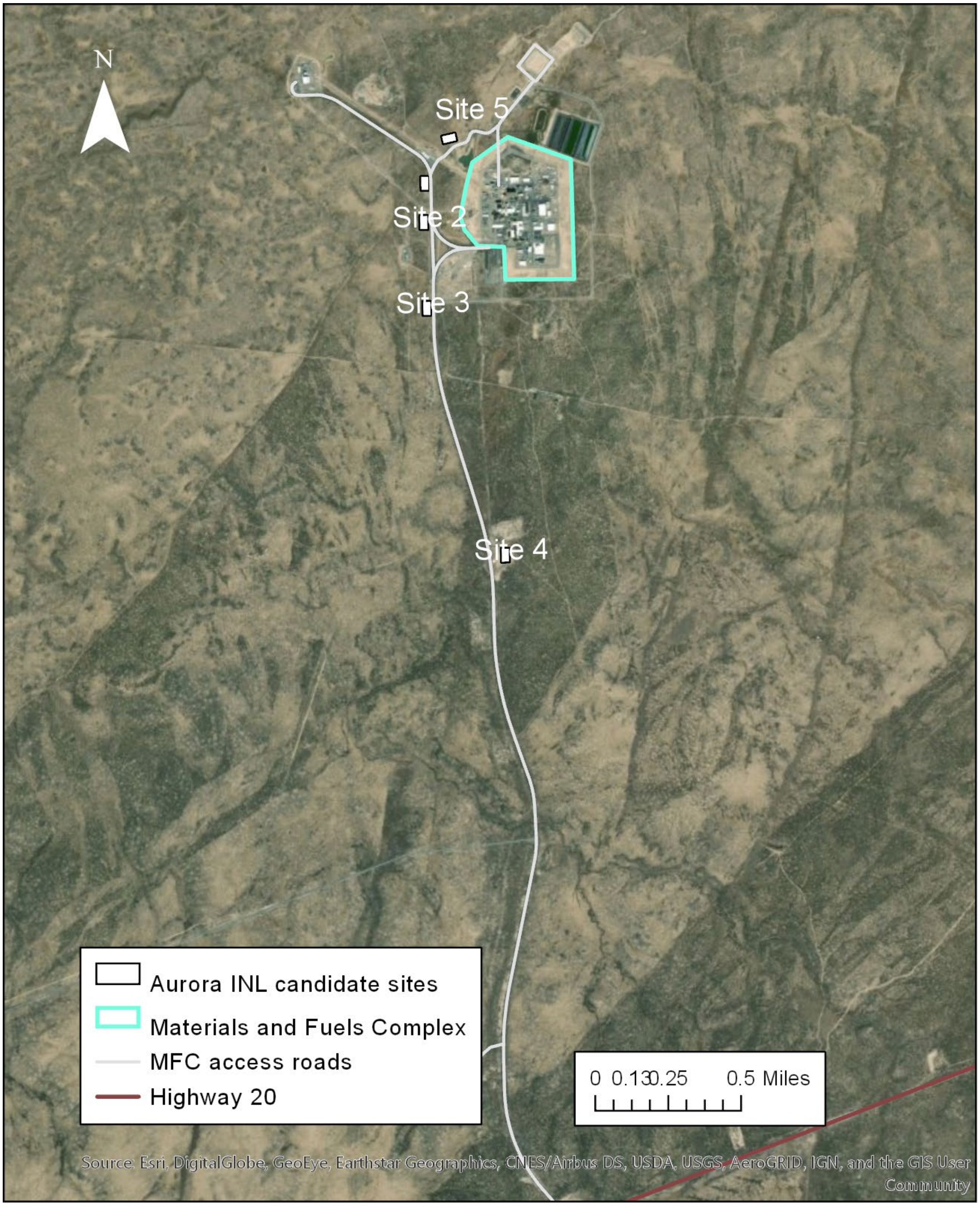

Figure 1-5: Location of the candidate sites by MFC

1.1.2 Boundaries of the proposed site

1.1.2.1 Determination of exclusion area and low population zone

Section 100.3, “Definitions,” of 10 CFR, defines an exclusion area and a low population zone.

The definition for an exclusion area is as follows:

> …that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result.

The definition for a low population zone is as follows:

> …the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific area, or instructed to take shelter, on a timely basis will depend on many factors such as location, number and size of highways, scope and extent of advance planning, and actual distribution of residents within the area.

The exclusion area and the low population zone for the Aurora are set at the Aurora powerhouse boundary. There are no residents in the low population zone. The Aurora powerhouse is within the Aurora INL site as can be seen in Figure 1-6.

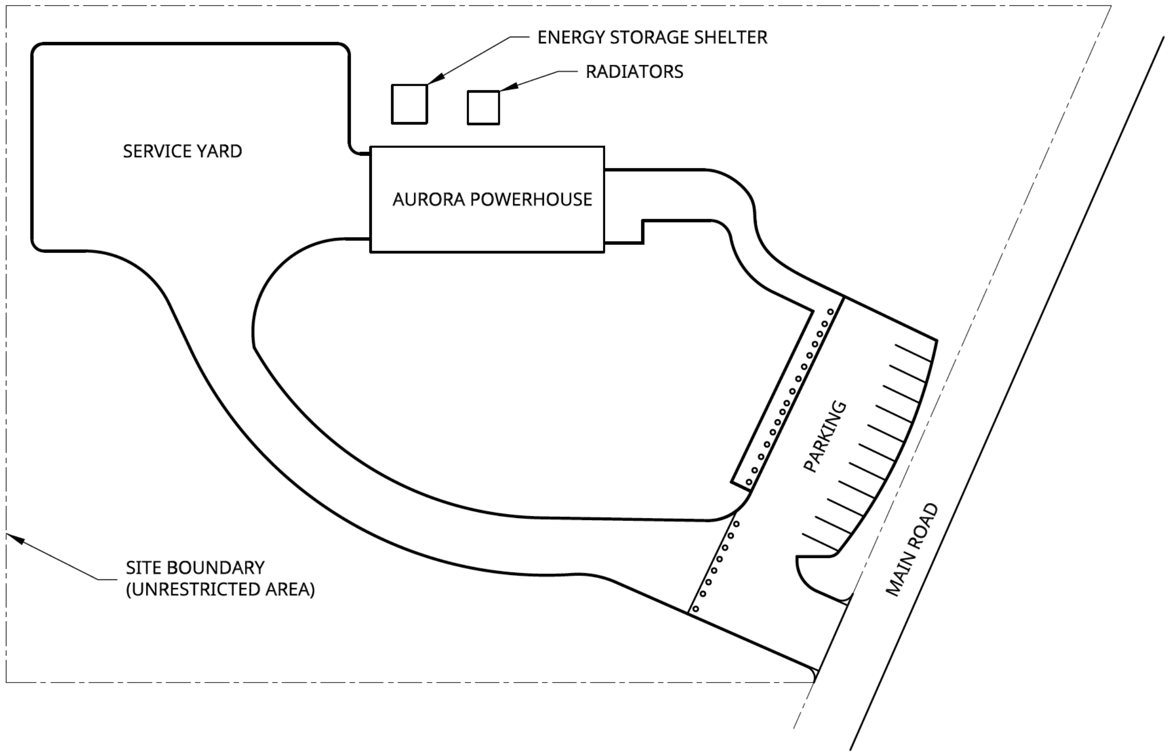

Figure 1-6: Schematic of an Aurora site

1.1.2.2 Determination of population center

Section 100.3, "Definitions," of 10 CFR, defines a population center distance as "the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents." Additionally, 10 CFR 100.21(b) requires the following:

> The population center distance, as defined in § 100.3, must be at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, the boundary of the population center shall be determined upon consideration of population distribution. Political boundaries are not controlling in the application of this guide;

Since the distance from the reactor to the Aurora powerhouse wall varies between 8.5 and 47 feet, the longer of the two values is assumed. This results in a population center distance of 63 feet from the powerhouse. This distance is significantly smaller than the distance to the closest densely populated center, which is Idaho Falls, which is 25 miles away.

1.2 Evaluation of the proposed site

The purpose of this section is to evaluate the proposed site against the generic site envelope and to provide information related to site characterization. The purpose of providing an evaluation of the proposed site against the generic site envelope is to satisfy 10 CFR 52.79(a)(1)(iii)-(v). The purpose of providing information related to site characterization is to satisfy 10 CFR 52.79(a)(1)(vi) which requires, in part, “Site characteristics must comply with part 100 of this chapter,” which further requires 10 CFR Part 100, Subpart B, “Evaluation Factors for Stationary Power Reactor Site Applications on or After January 10, 1997.” Therefore, the purpose of this section is to meet the requirements of 10 CFR 52.79(a)(1)(iii)-(vi), 10 CFR 100.21(c)-(g), and 10 CFR 100.23(c)-(d)[1] by providing an evaluation of the proposed site, which includes a description against the Oklo Aurora generic site envelope, described further in this section, as well as information on seismic, meteorological, hydrologic, geologic, man-made hazards, and population demographic characteristics of the proposed site.

1.2.1 Generic site envelope evaluation

The generic site envelope is the set of bases to ensure safe operation of the Aurora and is outlined in detail in Appendix B, "Generic site envelope,” of this chapter. The generic site envelope defines five site bases: (1) coastal site, (2) external fire, (3) geologic, (4) man-made hazards, and (5) seismic. Each basis has specific site commitments, which are derived from the external hazards evaluation, described in Appendix A, "External hazards evaluation,” of this section. A site commitment is made by Oklo Power to perform a specific action when undergoing site selection for the Aurora. This section discusses each of the generic site envelope site bases.

1.2.1.1 Coastal site evaluation

The generic site envelope requires further investigations for coastal sites. Since the proposed site is not located on a coastline, the proposed site meets the coastal site basis of the generic site envelope by default.

1.2.1.2 External fire evaluation

The generic site envelope requires that 30 feet within the Aurora foundation be cleared from vegetation and any vegetation slash, as per National Fire Protection Association (NFPA) 1144, “Standard for reducing structure ignition hazards from wildland fire.” This requirement is not dependent on the specific location of the site and is therefore applicable to the Aurora INL site. Oklo Power makes the appropriate site commitment, summarized below and verified by an Inspection, Tests, Analyses, and Acceptance Criteria (ITAAC), to ensure the Aurora INL site meets the requirements of the generic site envelope to mitigate the external fire hazard.

[1] The remainder of the 10 CFR Part 100, Subpart B, requirements that apply to the Aurora are 10 CFR 100.21(a)-(b) and 10 CFR 100.21(h). These requirements are discussed in other sections of this chapter.

Site basis:

SB.01 The proposed site will not damage the Aurora facility due to an external fire.

Site evaluation summary:

The INL Aurora site is described in Section 1.1. It is evaluated against the generic site envelope in Appendix B of this chapter. As described in the generic site envelope, the external hazard presented by an external fire can be mitigated by clearing the area surrounding the reactor. A site commitment is taken to clearing the area, and the appropriate programmatic controls are in place to verify it.

Site commitments and programmatic controls:

SC.01.A The area directly surrounding the Aurora powerhouse will be cleared during site preparation in accordance with NFPA 1144.

ITAAC.SS.01.A (see Part VI)

1.2.1.3 Geologic evaluation

The generic site envelope requires further geologic investigations to satisfy the geologic site basis. These investigations are discussed in the following sections.

1.2.1.3.1 Avalanches

The generic site envelope requires information be provided on if the proposed site is in an avalanche-prone environment. The proposed site is considered to be in an avalanche-prone environment if the proposed site has both of the following characteristics:

- Is within 1 mi of a slope greater than 25 degrees, judged by 100 ft contour lines
- Has data indicating avalanches have occurred in the region or geomorphologic indicators of avalanches

The area directly surrounding the proposed site is flat, as can be seen in Figure 1-7. There are no slopes near any of the sites with a greater gradation than 25 degrees as judged by 100 foot contour lines. Additionally, historical data shows no evidence of avalanches in the area and there are no geomorphologic indicators of avalanches.

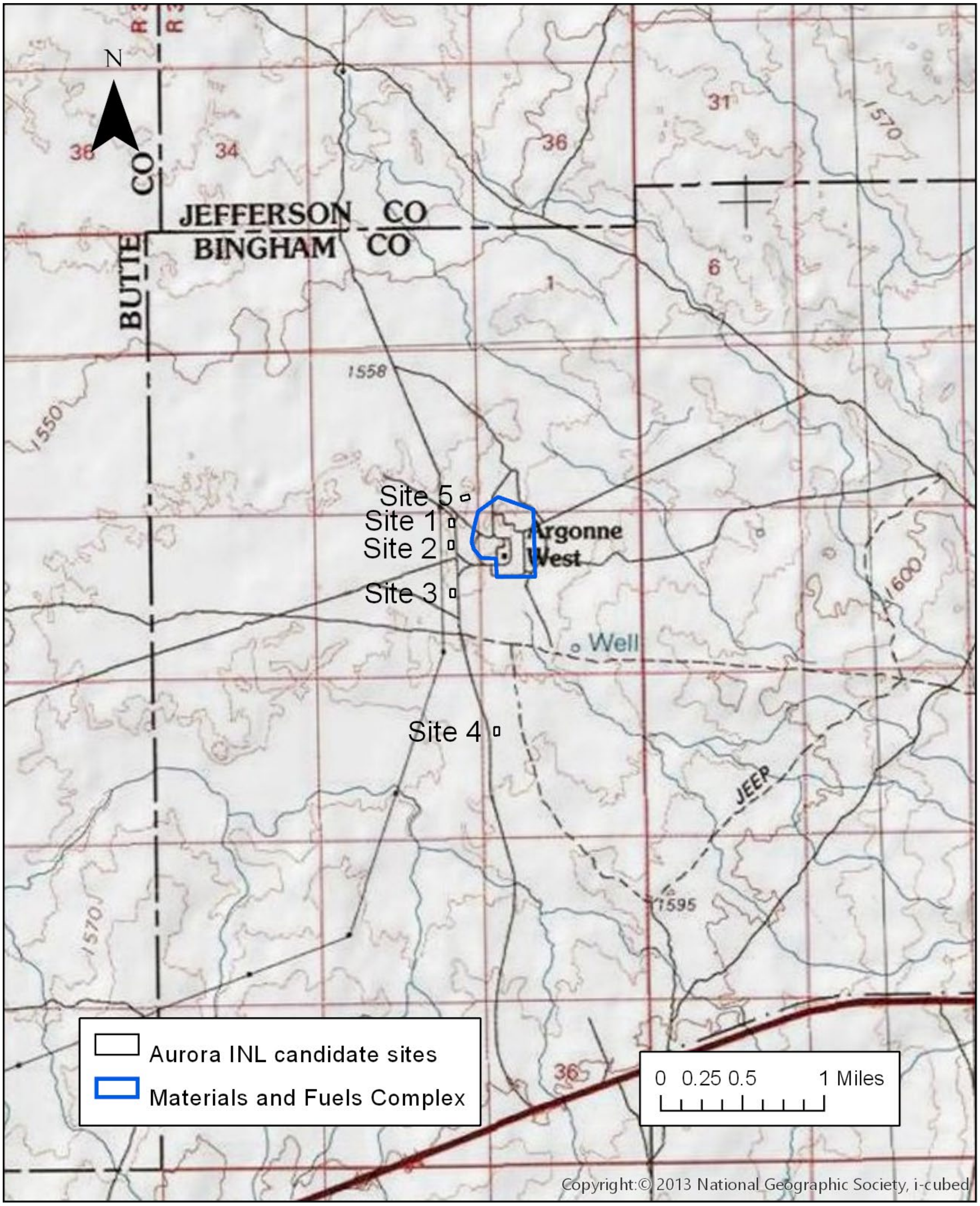

Figure 1-7: Topographic map of the area surrounding the proposed site

Since the proposed site is not located in an avalanche-prone environment, the proposed site meets the avalanche site commitment of the geologic basis by default.

1.2.1.3.2 Landslides

The generic site envelope requires further investigations for landslide-prone environments. If the proposed site has either of the following characteristics:

- Is within 2 mi of a slope greater than 15 degrees, judged by 100 ft contour lines

- Has data indicating landslide have occurred in the region

Then the proposed site is considered to be in landslide-prone environment and requires further landslide investigations.

As can be seen in Figure 1-7, the area directly surrounding the proposed site is flat. There are no slopes near any of the sites with a greater gradation than 15 degrees as judged by 100 foot contour lines. Additionally, historical data does not indicate landslides in the region [3].

Since the proposed site is not located in a landslide-prone environment, the proposed site meets the landslide site commitment of the geologic basis by default.

1.2.1.3.3 Sinkholes

The generic site envelope requires further investigations to determine if the proposed site is in a karst terrain, i.e., terrain where water can drain below the ground and dissolve water-soluble evaporate rock such as salt, gypsum, or carbonate rocks. As can be seen in Figure 1-8, the area directly surrounding the site does not have karst terrain. Therefore, the proposed site meets the sinkhole site commitment of the geologic basis.

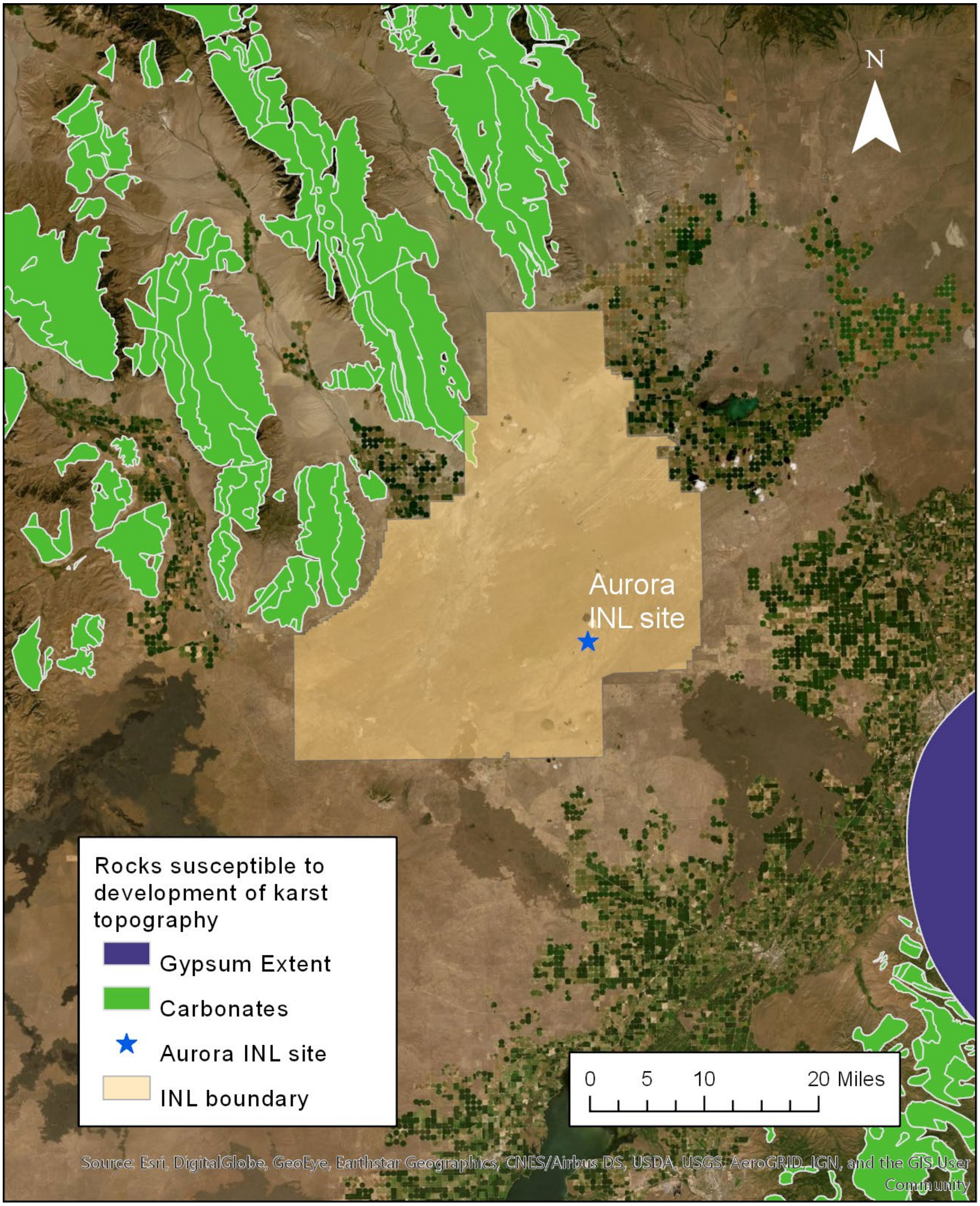

Figure 1-8: Karst terrains surrounding INL

1.2.1.4 Man-made hazards evaluation

The generic site envelope requires further investigation for man-made hazards. Specifically, it requires that an evaluation be conducted surrounding the proposed site to determine if there are any explosive hazards. This requirement is not dependent on the specific location of the site and is, therefore, applicable to the Aurora INL site. Oklo Power makes the appropriate site commitment, summarized below and verified by an ITAAC, to ensure the Aurora INL site meets the requirements of the generic site envelope to mitigate man-made hazards.

Site basis:

SB.02 The proposed site will not damage the Aurora reactor by an explosion.

Site evaluation summary:

The INL Aurora site is described in Section 1.1. It is evaluated against the generic site envelope in Appendix B of this chapter. As described in the generic site envelope, the external hazard presented by man-made hazards, namely explosive hazards, must be evaluated for every site. If blast hazards are identified, further evaluation is required. A site commitment is taken to conducting the required evaluation(s), and the appropriate programmatic controls are in place to verify it.

Site commitments and programmatic controls:

SC.02.A The area surrounding the proposed site will be evaluated for explosive hazards. Blast hazards that are identified will be evaluated to determine if their resulting pressure exceeds the blast capacity.

ITAAC.SS.02.A (see Part VI)

1.2.1.5 Seismic evaluation

The generic site envelope requires that further evaluations be performed if the largest recorded PGA, as per ASCE 7, is greater than 0.5 g. The ASCE 7 PGA value for the proposed site is determined by looking up the coordinates of the proposed site, with the following settings:

- ASCE/SEI 7-16
- Risk category IV
- Site soil class “A” – hard rock

The ASCE 7 PGA for the proposed site is found to be 0.106 g.[2] Since this value is smaller than 0.50 g, no further evaluations are required, and the seismic basis of the generic site envelope is met by default.

[2] This value is current as of February 29, 2020 via the ASCE 7 online tool, available at https://asce7hazardtool.online/.

1.2.1.6 Generic site envelope results

The generic site envelope is established in Appendix B of this chapter. The generic site envelope defines four site bases: coastal site, external fire, geologic, and man-made hazards. The coastal site basis is inherently met by the proposed site since the Aurora INL site is farther than one-half of a mile from a coastline. The external fire basis is met through an appropriate Oklo Power commitment and verified by ITAAC.SS.01.A. The geologic basis is met since there are no slopes greater than 15 degrees within the proposed site, historical data does not indicate avalanches or geomorphic indicators of avalanches, historical data does not indicate landslides, and the proposed site is not within a quarter mile of karst terrain. The man-made hazards basis is met through an appropriate Oklo Power commitment and verified by ITAAC.SS.02A. The evaluation of the proposed site against the generic site envelope is shown in Table 1-2.

Table 1-2: Generic site envelope evaluation results

Basis	Commitment	Parameter	Value
Coastal			
	Coastal	Distance to a coast (mi)	> 0.5
External fire			
	Fire	Brush clearing (ft)	30
Geologic			
	Avalanche	Distance to slope greater than 25 degrees (mi)	> 1
		Historic avalanche data or geomorphologic indicators of avalanches (Y/N)	N
	Landslide	Distance to slope greater than 15 degrees (mi)	> 0.25
		Historic landslide data (Y/N)	N
	Sinkhole	Distance to karst terrain (mi)	> 0.25
Man-made hazards			
	Man-made hazards	Blast hazards investigation (mi)	0

1.2.2 Proposed site characterization information

Site characterization information is required by 10 CFR 52.79(a)(1)(iii)-(v) and 10 CFR Part 100, Subpart B, and is provided for purposes of meeting the regulatory requirements, but is not needed in order to evaluate the safe operation of the Aurora at the proposed site. Therefore, this section provides only high-level information. Table 1-3 provides information on the organization of this section, as well as which regulatory requirements that are being addressed.

Table 1-3: Organization of proposed site characterization information

Section	Section title	Requirement
1.2.1	Seismic evaluation	10 CFR 52.79(a)(1)(iii)
		10 CFR 100.21(d)
		10 CFR 100.23(c)

		10 CFR 100.23(d)
1.2.2	Meteorological evaluation	10 CFR 52.79(a)(1)(iii)
		10 CFR 100.21(d)
1.2.3	Hydrologic evaluation	10 CFR 52.79(a)(1)(iii)
		10 CFR 100.21(d)
1.2.4	Geologic evaluation	10 CFR 52.79(a)(1)(iii)
		10 CFR 100.21(d)
		10 CFR 100.23(c)
		10 CFR 100.23(d)
1.2.5	Man-made hazards evaluation	10 CFR 52.79(a)(1)(iv)
		10 CFR 100.21(e)
1.2.6	Population demographics evaluation	10 CFR 52.79(a)(1)(v)
		10 CFR 100.21(h)
1.2.7	Atmospheric dispersion characteristic evaluation	10 CFR 100.21(c)
1.2.8	Security plan impact evaluation	10 CFR 100.21(f)
1.2.9	Emergency plan impact evaluation	10 CFR 100.21(g)

1.2.2.1 Seismic evaluation

Site-specific seismic information is typically provided for large reactors for many reasons, including hazards to the powerhouse, failure of safety systems, and other reactor safety disturbances. However, the Aurora is evaluated and found robust against an extreme seismic event, as part of the external hazards evaluation in Appendix A, "External hazards evaluation," of this chapter. This major seismic event assumes a complete collapse of the powerhouse and applies an extreme ground acceleration to the reactor module, without challenging the safety of the reactor. Since it is impractical to bound all large ground accelerations, an extreme ground acceleration was assumed as part of the deterministic analysis for the seismic event family. Therefore, the generic site envelope provides for a site commitment related to the assumed ground acceleration, located in Appendix B, "Generic site envelope."

1.2.2.1.1 Generic site envelope consideration

There is one seismic commitment included in the generic site envelope that relates to the safe operation of the Aurora. This commitment relates to evaluating the proposed site to ensure that the recorded PGA is bounded by the external hazards seismic analysis. This commitment is discussed in the generic site envelope evaluation in Section 1.2.1.5.

1.2.2.1.2 Proposed site considerations and evaluations

The information provided in this section is strictly to meet the requirements of 10 CFR 52.79(a)(iii), 10 CFR 100.21(d), and 10 CFR 100.23(c)-(d), but does not contribute to the safety of the plant.

Sections 52.79(a)(1)(iii) and 100.21(d) of 10 CFR require general seismic information regarding the site in order to evaluate the safety impact of the proposed site to the facility. This information is addressed through the external hazards evaluation and the generic site envelope, since they relate to the safety of the Aurora, and are located in Appendix A, "External hazards evaluation," and Appendix B, "Generic site envelope," of this chapter of the final safety analysis report. Section 100.23(c)-(d) of 10 CFR contains specific requirements for seismic information to be submitted, which are included in this section, and are as follows:

- Data on vibratory ground motion
- Tectonic and nontectonic surface deformation
- Earthquake recurrence rates
- Fault geometry and slip rates
- Seismically-induced flooding
- Site foundation material
- Safe Shutdown Earthquake Ground Motion
- Liquefaction potential

Other requirements from 10 CFR 100.23(d), such as, soil and rock stability, natural and artificial slope stability, cooling water supply, and remote safety-related structure siting, are discussed in Section 1.2.2.4.

1.2.2.1.2.1 Vibratory ground motion

The intensity of an earthquake, experienced at a given distance from the epicenter, depends on the underlying bedrock and soil characteristics as well as the magnitude of the earthquake. The rock underlying the western U.S. does not transfer vibratory ground motion as far as the east coast. This is due, in part, to the existence of more faults in the western U.S., which do not allow a transfer of energy as easily as land with fewer faults (i.e., the eastern U.S.) [4]. After a thorough review of the region around the proposed site, a 175 mi radius around the Aurora INL site was found to be sufficiently large in order to include earthquakes that could likely be felt at the Aurora INL site. Figure 1-9 shows earthquakes within 175 mi of the site.

Table 1-4 provides recorded 6.0 Richter-scale earthquakes experienced in the 175 mi radius around the Aurora INL site to provide an illustration of the overall historic vibratory ground motion of the region. Figure 1-10 provides an illustration to convey the relative size and frequency of earthquakes in the region.

Table 1-4: Distance from earthquake epicenters to the Aurora INL site [5]

Rank	Magnitude (Richter)	Date	Location	Distance from site (mi)
1	7.3	1959	W. Montana	106
2	6.9	1983	S. Idaho	68
3	6.6	1934	N. Utah	131
4	6.5	1959	Yellowstone	124
5	6.4	1897	W. Montana	118
6	6.1	1947	W. Montana	97
7	6.1	1944	S. Idaho	106
8	6.1	1975	S. Idaho	132
9	6.0	1945	S. Idaho	140

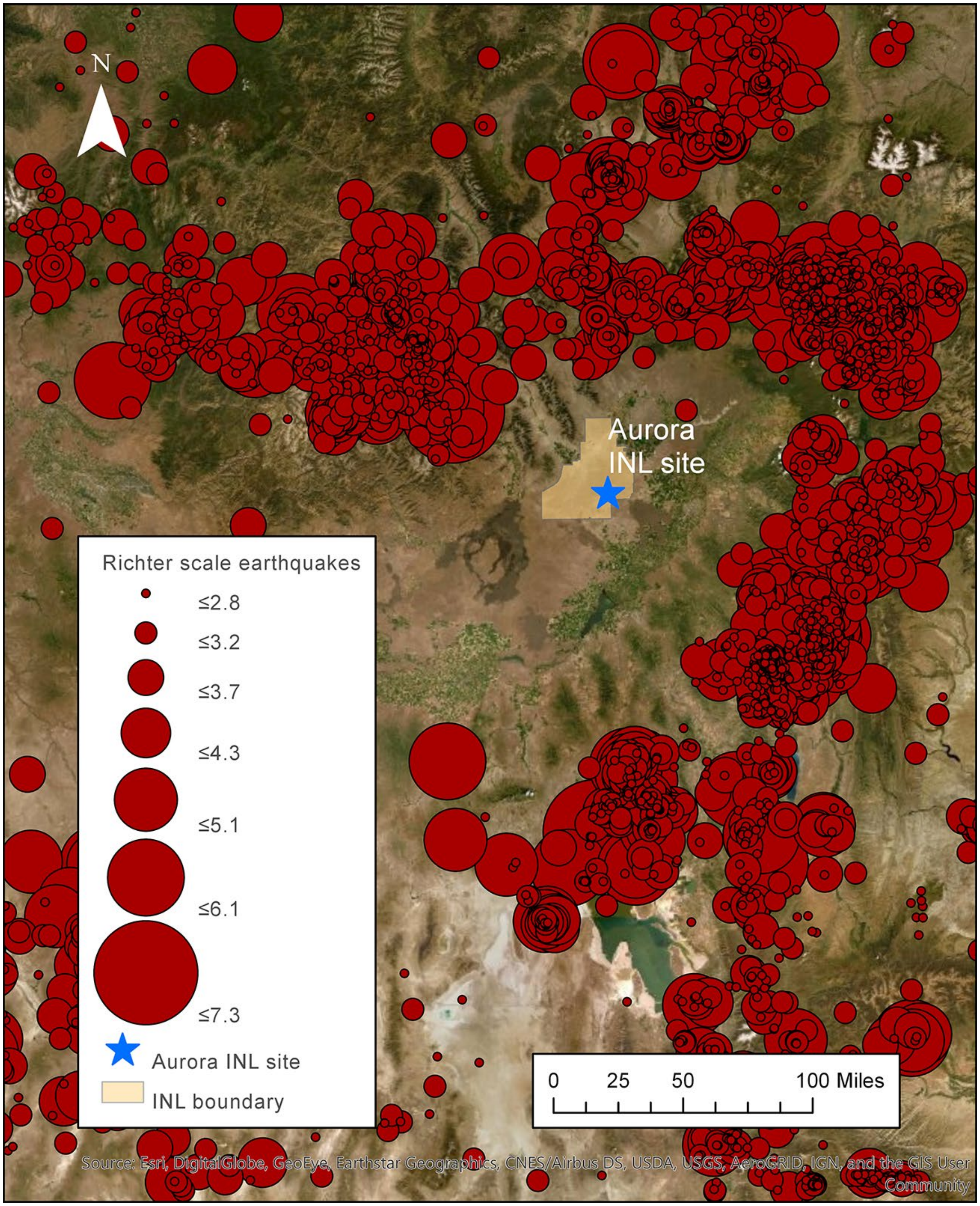

Figure 1-9: Earthquakes since 1890 within 175 mi of the Aurora INL site [5]

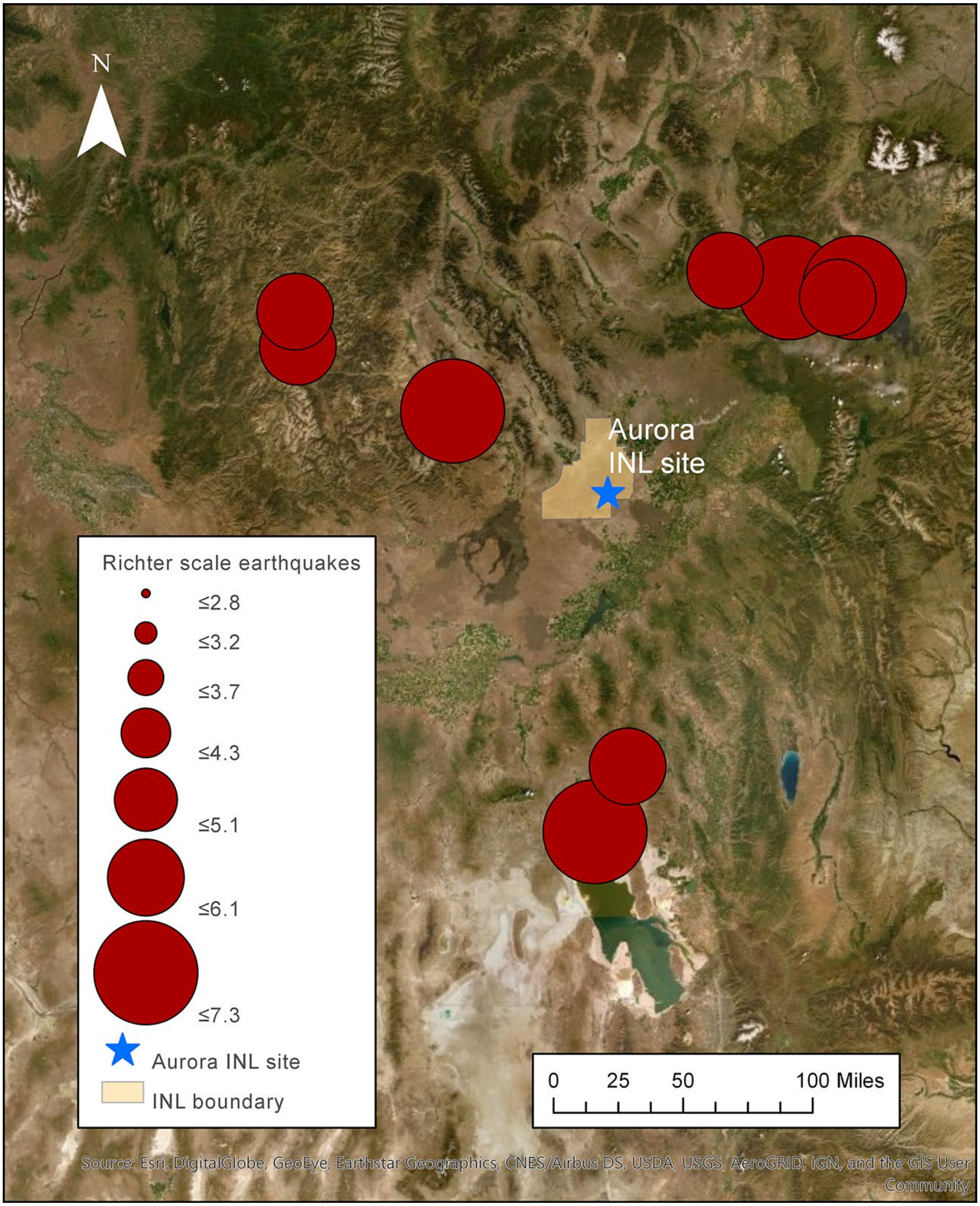

Figure 1-10: Earthquakes since 1890 with a Richter scale magnitude of 6.0 or greater [5]

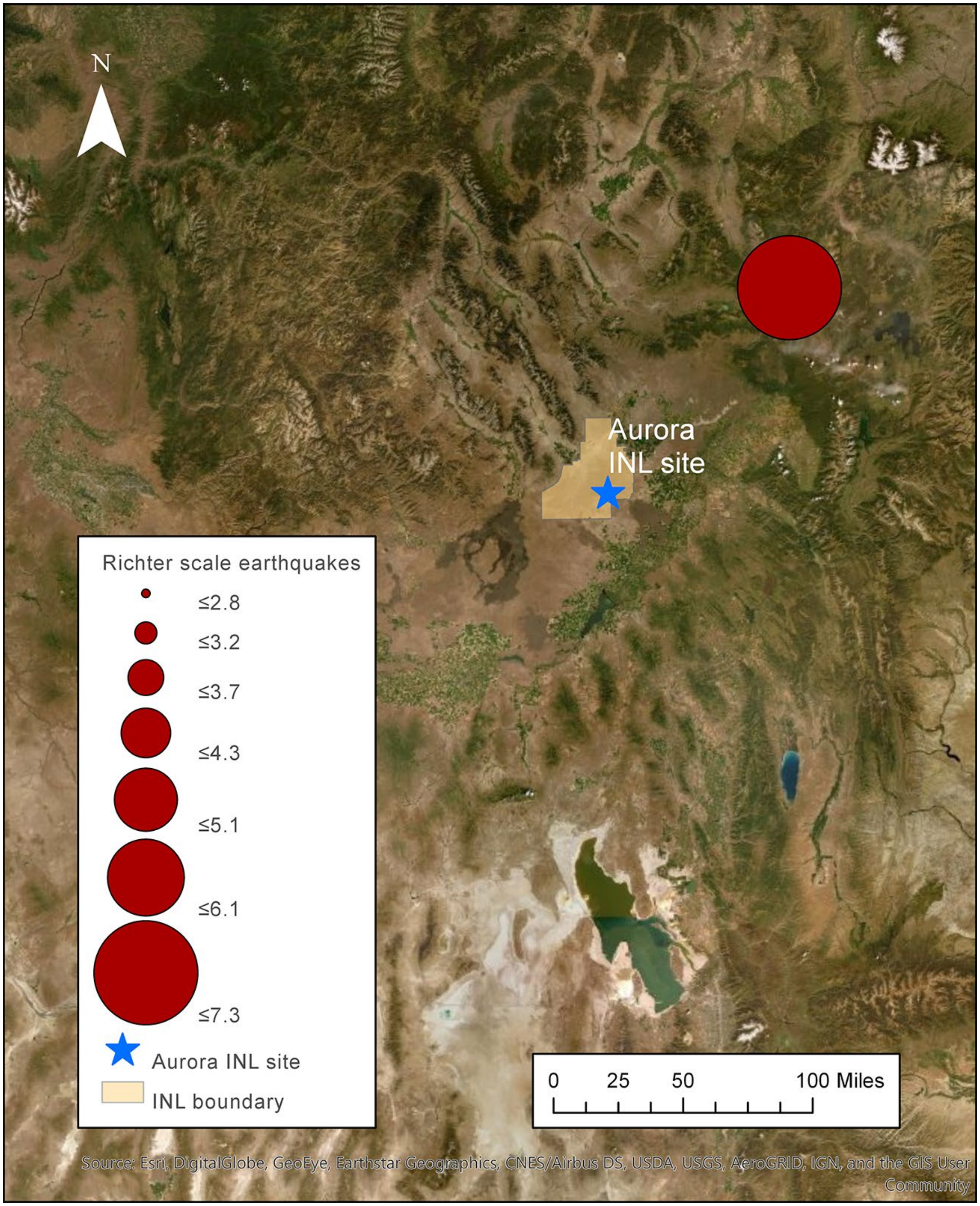

Figure 1-11: Earthquakes since 1890 with a Richter scale magnitude of 7.0 or greater [5]

1.2.2.1.2.2 Tectonic and nontectonic surface deformation

Surface deformation[3] is a typical concern for large light water reactors, which have large sites and rely on water for cooling. These large facilities have site boundaries that are on the order of square miles and reactors that rely on water that must be obtained from a local ultimate heat sink (e.g., lake, river, ocean). The proposed site for the Aurora, which is less than an acre, is extremely small in comparison, so there are no concerns associated with moving fluids long distances.

The primary concern for tectonic and nontectonic surface deformation, aside from seismic accelerations, is differential displacement of the site. For the Aurora, there is only one building, the powerhouse, which houses the reactor and the secondary system, and the ultimate heat sink does not require piping outside of the Aurora powerhouse. Additionally, the Aurora does not use water for cooling, nor any other fluid that must be imported from offsite. As a result, there is no concern surrounding differential displacement across a fault. Therefore, surface deformation is not a concern and no further information is provided.

Figure 1-12 shows the active faults found within 300 mi of the Aurora INL site. Descriptions of the classes of active faults are given in Table 1-5. The class B active faults could be surface faults. There are several class B active faults near the site, but none directly under the site. The closest class B active fault is approximately 7.8 mi away.

[3] Surface deformation, as defined in 10 CFR 100.3 is "a distortion of geologic strata at or near the ground surface by the processes of folding or faulting as a result of various earth forces."

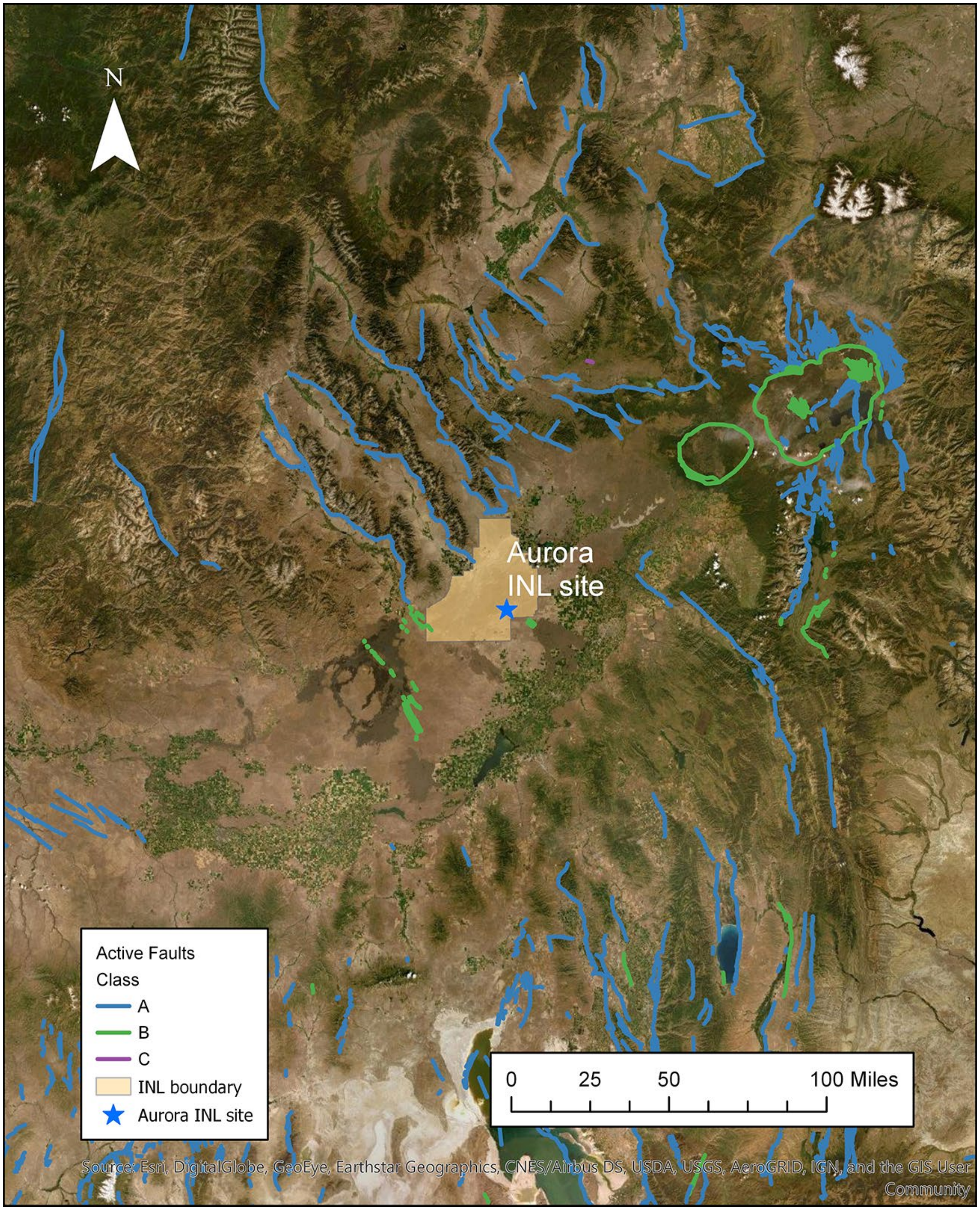

Figure 1-12: Active faults and their associated class [6]

Table 1-5: Classes used for active faults, liquefaction features, and deformation [7]

Class category	Definition

Class A	Geologic evidence demonstrates the existence of a Quaternary fault of tectonic origin, whether the fault is exposed for mapping or inferred from liquefaction or other deformational features.
Class B	Geologic evidence demonstrates the existence of a fault or suggests Quaternary deformation, but either (1) the fault might not extend deeply enough to be a potential source of significant earthquakes, or (2) the currently available geologic evidence is too strong to confidently assign the feature to Class C but not strong enough to assign it to Class A.
Class C	Geologic evidence is insufficient to demonstrate (1) the existence of tectonic fault, or (2) Quaternary slip or deformation associated with the feature.
Class D	Geologic evidence demonstrates that the feature is not a tectonic fault or feature; this category includes features such as demonstrated joints or joint zones, landslides, erosional or fluvial scarps, or landforms resembling fault scarps, but of demonstrable non-tectonic origin.

1.2.2.1.2.3 Earthquake recurrence rates

Figure 1-13 includes the approximate recurrence rates of different magnitude earthquakes impacting INL. However, earthquake recurrence rates do not have a significant impact on the Aurora since the Aurora is extremely seismically robust against the earthquakes possible in the region. Additionally, the operating life of the Aurora is shorter than large light water reactors.

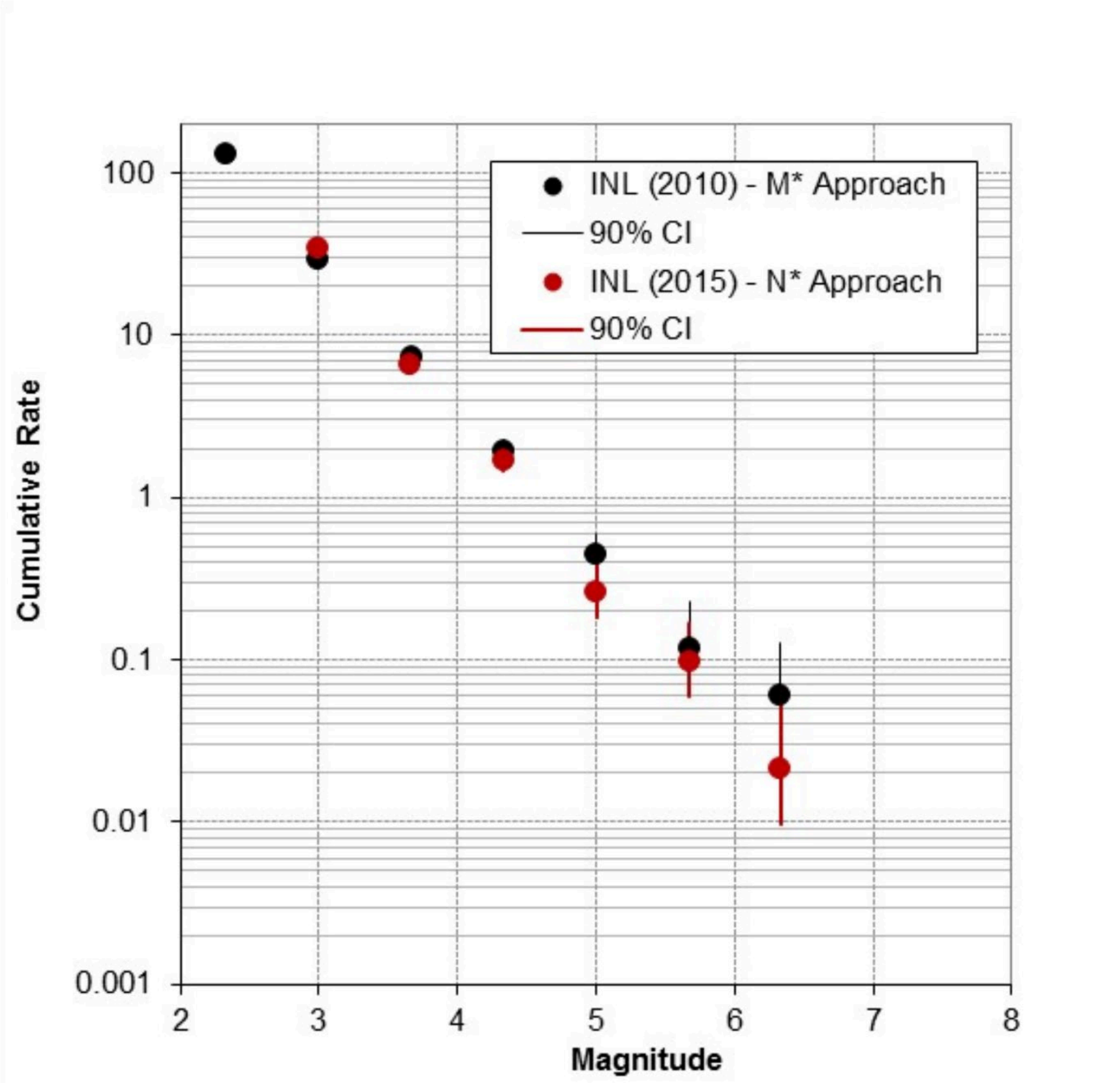

Figure 1-13: Earthquake recurrence rate from the INL SSHAC-S analysis [8]

1.2.2.1.2.4 Fault geometry and slip rates

The active faults within the Aurora INL site are presented in Figure 1-14. The class B active faults nearest to the site have unknown slip rates. As the faults are class B, they are unlikely to result in a significant seismic event. The majority of active faults in the region have a slip rate of less than 0.2 mm/yr.

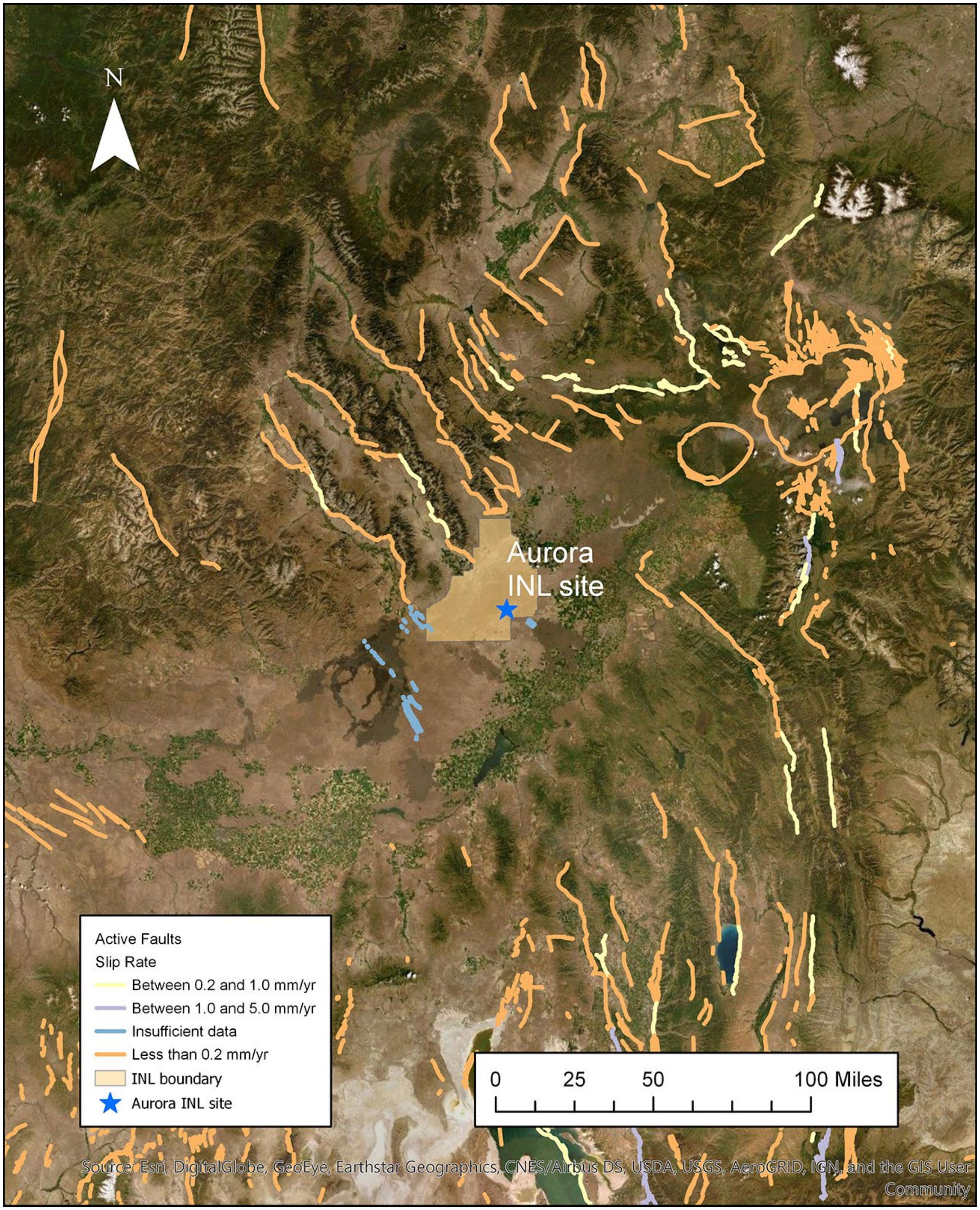

Figure 1-14: Slip rates associated with regional active faults [6]

1.2.2.1.2.5 Seismically-induced flooding

The Aurora INL site is not located along a coastline. There are no concerns or provided information related to seismically-induced flooding.

1.2.2.1.2.6 Site foundation material

The site foundation material is primarily concrete, and there are no safety concerns associated with the site settling. As the Aurora facility is contained in one building, there are no concerns with differential displacement of the site across a fault.

1.2.2.1.2.7 Safe Shutdown Earthquake ground motion

The safe shutdown earthquake (SSE) is site agnostic for the Aurora, rather than determined on a site-by-site basis. This is possible due to the robustness of the Aurora against a large seismic event, as is shown by the external hazards evaluation. The Aurora is analyzed to ground accelerations that are greater than any forecasted ground acceleration in most of the U.S. The PGA of the proposed site is 0.106 g and the Aurora is analyzed to 0.50 g. The seismic site commitment that relates to this analysis is in Section 1.2.1.5. Therefore, the proposed site does not challenge the design and no additional design features are needed to alleviate the seismicity of the region.

1.2.2.1.2.8 Liquefaction potential

Liquefaction can occur during a seismic event due to a reduction in the volume of soil as void spaces are filled due to earthquake shock. The soils that are most vulnerable to liquefaction are poorly drained granular soils. Liquefaction is most common in areas with shallow groundwater, which generally means that the water table is less than 50 feet from the ground surface [7][8]. As can be seen in Figure 1-15, the groundwater at the MFC is significantly lower than 50 ft, so there are no liquefaction concerns. The values given at each well site are the most recently measured value as of January 2020 and are the values for depth to water level, feet below land surface.

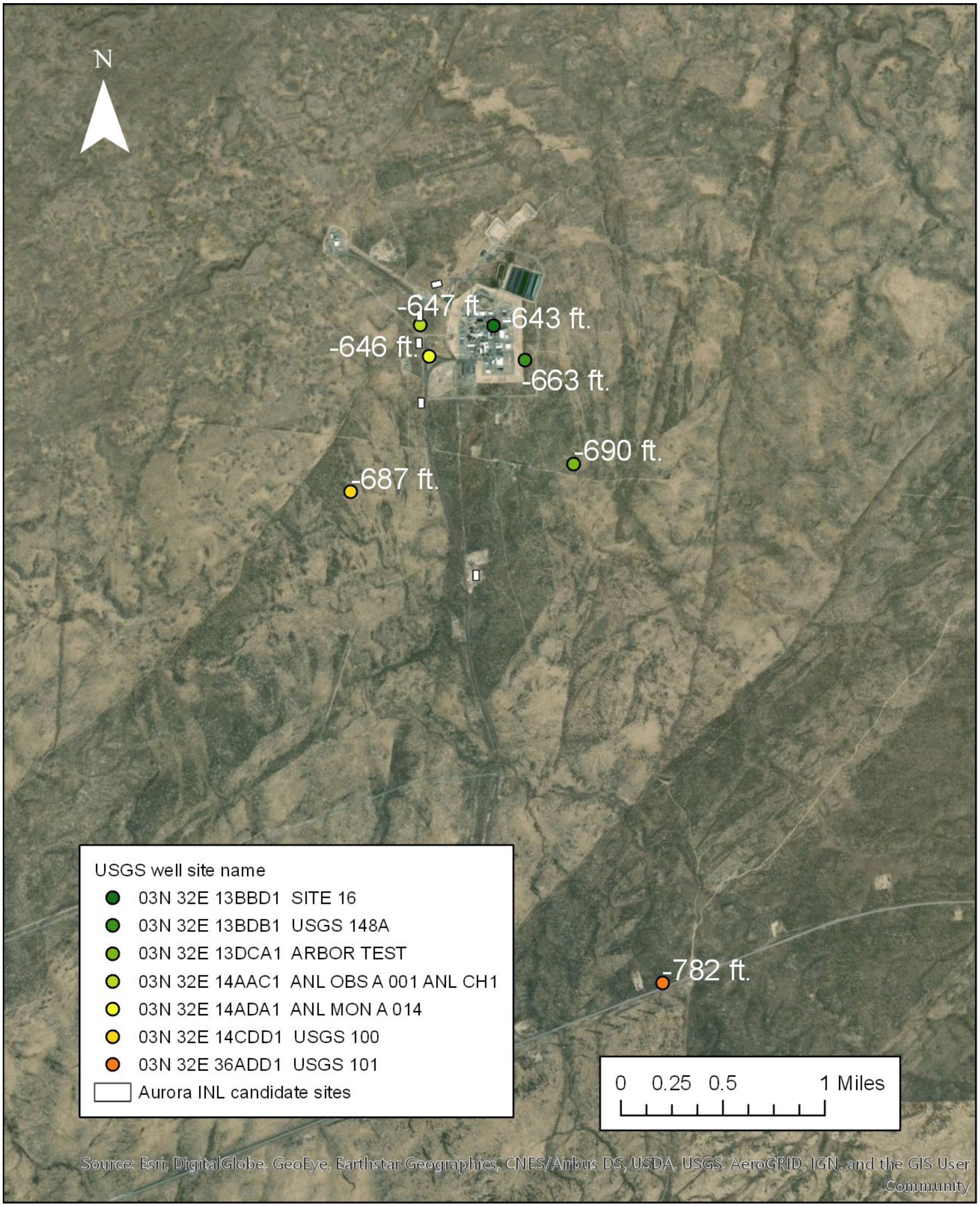

Figure 1-15: Groundwater levels measured near the MFC [11]

1.2.2.2 Meteorological evaluation

Site-specific meteorological information is typically provided for many reasons, including hazards to the Aurora powerhouse, potential flooding, and meteorological parameters that become part of the atmospheric dispersion calculations in the dose analysis. However, the external hazards evaluation in Appendix A, "External hazards evaluation," of this chapter of the final safety analysis report found the Aurora robust against extreme meteorological hazards, which assume a complete collapse of the powerhouse and water inside the facility. Lastly, site-specific atmospheric dispersion parameters are not relevant because no fission product release is postulated, as addressed in Section 1.3. Additionally, dose calculations done in the design phase of the Aurora used wind speeds that would maximize dose, which is another conservative mechanism to analyze atmospheric dispersion effects on dose consequences without requiring meteorological data. Therefore, the information provided in this section is strictly to meet the regulations and is not needed for determining the safety of the plant at the proposed site.

1.2.2.2.1 Generic site envelope consideration

The generic site envelope does not contain any meteorological evaluations that need to be performed for the proposed site to ensure the safety of the Aurora.

1.2.2.2.2 Proposed site considerations and evaluations

The information provided in this section is strictly to meet the requirements of 10 CFR 52.79(a)(iii) and 10 CFR 100.21(d). Since 10 CFR 52.79(a)(1)(iii) and 10 CFR 100.21(d) require general meteorological information regarding the site in order to evaluate the safety impact of the proposed site to the facility, those requirements are encapsulated in the generic site envelope. Since there is no additional meteorological information that is required, nothing further is provided.

1.2.2.3 Hydrologic evaluation

Hydrologic information, as it relates to a proposed site, is typically provided to assess whether a site is prone to flooding and how that flooding might affect the safety of the facility. For large light water reactors, there are diverse hydrologic considerations typically due to reliance on large bodies of water of certain characteristics for use in cooling. Aurora does not require water for cooling and flooding to the Aurora facility is generally not challenging. The introduction of water in the facility is covered by the external hazards evaluation in Appendix A, "External hazards evaluation," of this chapter of the final safety analysis report. This analysis assumed water inside the facility and finds that the safety of the Aurora is not challenged by flooding, so no site-specific evaluation is needed for safety purposes. Therefore, the information provided in this section is strictly to meet the regulations and is not needed for determining the safety of the plant at the proposed site.

1.2.2.3.1 Generic site envelope consideration

The generic site envelope does not contain any hydrologic evaluations that need to be performed for the proposed site to ensure the safety of the Aurora.

1.2.2.3.2 Proposed site considerations and evaluations

The information provided in this section is strictly to meet the requirements of 10 CFR 52.79(a)(iii) and 10 CFR 100.21(d). Since 10 CFR 52.79(a)(1)(iii) and 10 CFR 100.21(d) require general hydrologic information regarding the site in order to evaluate the safety impact of the proposed site to the facility, those requirements are encapsulated in the generic site envelope. Since there is no additional meteorological information that is required, nothing further is provided.

1.2.2.4 Geologic evaluation

Geologic site information is typically provided to ensure that soil and topography does not negatively affect the safety of the plant. As can be seen in the external hazards evaluation included in Appendix A, "External hazards evaluation," of this chapter, the Aurora is generally not susceptible to geologic hazards due to the robustness of the reactor module. However, several specific external hazards are not evaluated as part of the external hazards evaluation and have related site commitments, provided in Appendix B, "Generic site envelope" of this chapter. Therefore, the information provided in this section is strictly to meet the regulations and is not needed for determining the safety of the plant at the proposed site.

1.2.2.4.1 Generic site envelope consideration

There are several geologic site commitments included in the generic site envelope that relate to the safe operation of the Aurora. These commitments are for avalanches, landslides, and sinkholes and are discussed in the generic site envelope evaluation in Section 1.2.1.3.

1.2.2.4.2 Proposed site considerations and evaluations

The information provided in this section is strictly to meet the requirements of 10 CFR 52.79(a)(iii), 10 CFR 100.21(d), and 10 CFR 100.23(c)-(d), but does not contribute to the safety of the plant.

Sections 52.79(a)(1)(iii) and 100.21(d), of 10 CFR require general geologic information regarding the site in order to evaluate the safety impact of the proposed site to the facility. This information is addressed through the external hazards evaluation and the generic site envelope since they relate to the safety of the Aurora. The external hazards evaluation and the generic site envelope are located in Appendix A, "External hazards evaluation" and Appendix B, "Generic site envelope" of this chapter. Specific requirements for seismic information to be submitted, which partially include geologic information, are discussed in Section 1.2.2.1. Other geologic requirements from 10 CFR 100.23(c)-(d) are discussed in this section and include the following:

- Soil and rock stability
- Liquefaction potential
- Natural and artificial slope stability
- Cooling water supply
- Remote safety-related structure siting

1.2.2.4.2.1 Soil and rock stability

As can be seen in Figure 1-16, the candidate sites are located on sands over basalt or loess. Table 1806.2 of the 2018 International Building Code states the vertical foundation pressure applied to sand, silty sand, clayey sand, silty gravel and clayey gravel should be no greater than 2000 pounds per square foot (psf). The maximum vertical load given for clay, sandy clay, silty clay, clayey silt, silt and sandy silt is 1500 psf. The Aurora powerhouse and reactor module both have vertical loads less than 1500 psf, thus there are no concerns surrounding the vertical load bearing capacity of the soil at the proposed site.

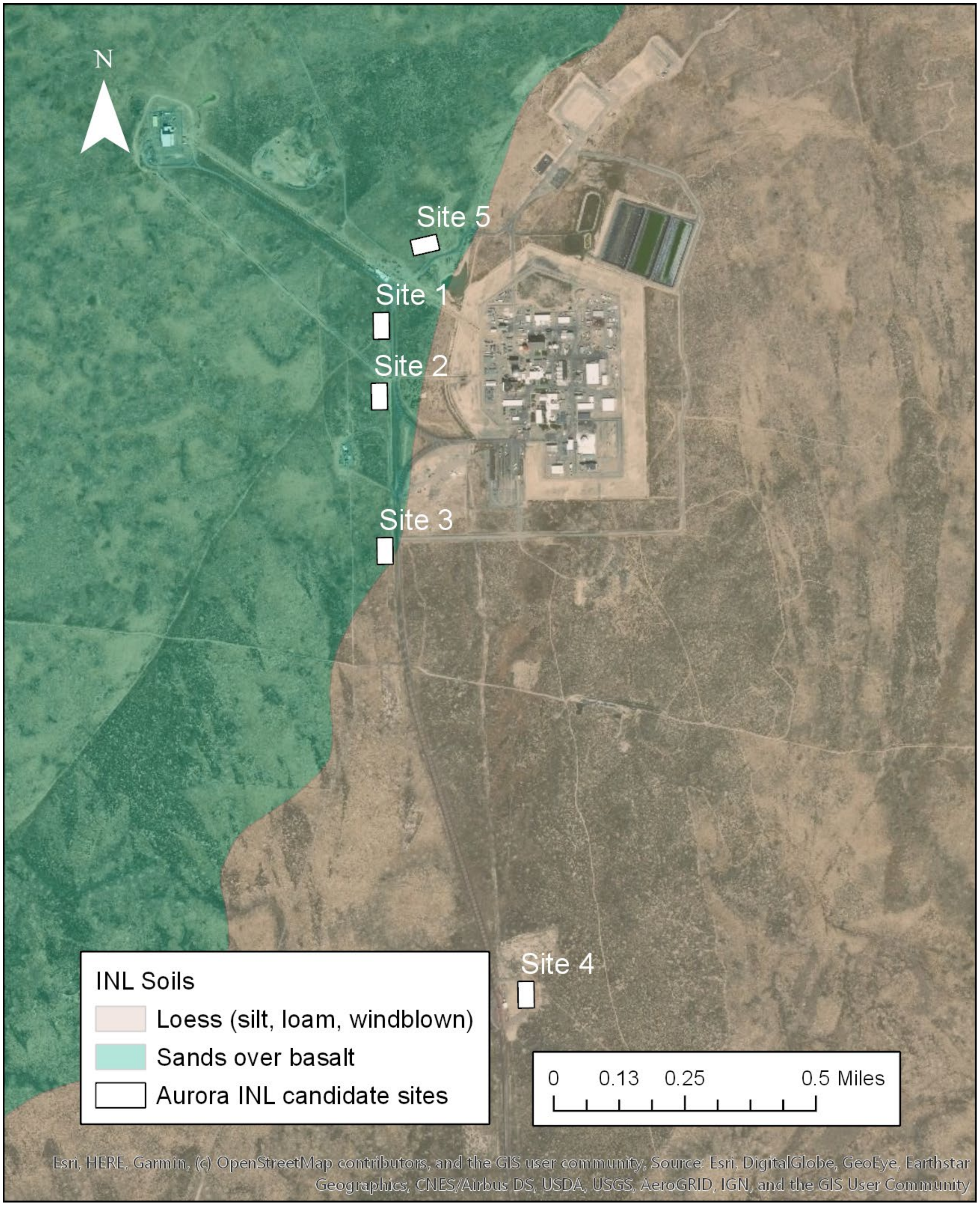

Figure 1-16: Soil underlying the are nearby the MFC [12]

Additionally, groundwater can impact the stability of the soil. Groundwater at the proposed site does not run within the underlying reactor module or building, thus groundwater is not further considered as impacting the soil and rock stability.

1.2.2.4.2.2 Natural and artificial slope stability

Information relating to natural slope is provided in Section 1.2.1.3. The only slight artificial slope is the small berm around the Aurora powerhouse, which is several feet tall. The artificial slope created by the berm around the powerhouse will be designed to allow appropriate drainage and with soils compacted in such a way as to mitigate the likelihood of failure. If a berm does fail, that does not challenge the safety of the Aurora.

1.2.2.4.2.3 Cooling water supply

The Aurora does not use water for cooling, nor any other fluid that must be imported from offsite, therefore no information on cooling water supply is provided.

1.2.2.4.2.4 Remote safety-related structure siting

The Aurora does not have associated remote safety-related structures, as the site is comprised mainly of the powerhouse. Therefore, no information on remote safety-related structure siting is provided.

1.2.2.5 Man-made hazards evaluation

For large reactors, man-made hazards can be seen as a potential threat to the operation of a reactor, in particular if nearby sites are capable of producing a large explosion. As can be seen in the external hazards evaluation included in Appendix A, "External hazards evaluation" of this chapter of the final safety analysis report, the Aurora is not generally susceptible to large explosive hazards due to the robustness of the reactor module. However, it is not possible to bound all blast scenarios and therefore, the generic site envelope provides for a site commitment, located in Appendix B, "Generic site envelope" of this chapter of the final safety analysis report. Therefore, the information provided in this section is strictly to meet the regulations and is not needed for determining the safety of the plant at the proposed site.

The purpose of this section is to satisfy the requirements of 10 CFR 52.79(a)(1)(iv) as well as the requirements of 10 CFR 100.21(e).

1.2.2.5.1 Generic site envelope consideration

There is one man-made hazard commitment included in the generic site envelope that relates to the safe operation of the Aurora. This commitment relates to evaluating explosive hazards within a certain distance of the site and the commitment is discussed in the generic site envelope evaluation in Section 1.2.1.4.

1.2.2.5.2 Proposed site considerations and evaluations

Information relating to man-made hazards for the proposed site is provided in Section 1.2.1.4.

1.2.2.6 Population demographics evaluation

Historically, population demographics have been submitted in prior large reactor applications due to the large radionuclides inventories and large site boundaries. Large reactors have radionuclide inventories that are orders of magnitude above that of the Aurora reactor. Further, the Aurora does not have a credible accident that could lead to a fission product release, as discussed in Section 1.3. Therefore, the information provided in this section is strictly to meet the regulations and is not needed for determining the safety of the plant at the proposed site.

1.2.2.6.1 Generic site envelope consideration

The generic site envelope does not contain any evaluations related to population demographics that need to be performed for the proposed site to ensure the safety of the Aurora.

1.2.2.6.2 Proposed site considerations and evaluations

The purpose of this section is to satisfy the requirements of 10 CFR 52.79(a)(1)(v), which requires, "The existing and projected future population profile of the area surrounding the site," and 10 CFR 100.21(h), which requires:

> Reactor sites should be located away from very densely populated centers. Areas of low population density are, generally, preferred. However, in determining the acceptability of a particular site located away from a very densely populated center but not in an area of low density, consideration will be given to safety, environmental, economic, or other factors, which may result in the site being found acceptable.

The Aurora INL site is located in southeastern Idaho and is in an area of low population density. No one resides within the Aurora INL site, nor within the INL Site. INL has 3,900 employees, and the proposed site is approximately 25 miles away from Idaho Falls and approximately 50 miles away from Pocatello.

Idaho Falls and Pocatello are the two largest cities in southeastern Idaho. The U.S. Census Bureau states the population, as of 2018, for Idaho Falls and Pocatello as 61,535 and 55,193 people, respectively, and that Idaho Falls has experienced a growth rate of 7.7% over the last 8 years for an average annual growth rate of slightly less than 1% per year, which is slightly greater than the average U.S. growth rate. The Pocatello population had an approximate 3.7% increase over the last 8 years, which is significantly less than the average U.S. growth rate.

Figure 1-17 displays the INL Site in the irregularly shaped shaded tan-colored area, and the location of the Aurora INL site with a blue star. The size of the blue bubbles represent the size of the population centers in the area. As can be seen in Figure 1-17, there is no one living nearby the site, and very few people living within the general region. Because of the small size of the Aurora INL site and low population density of the region, no further detailed information is included.

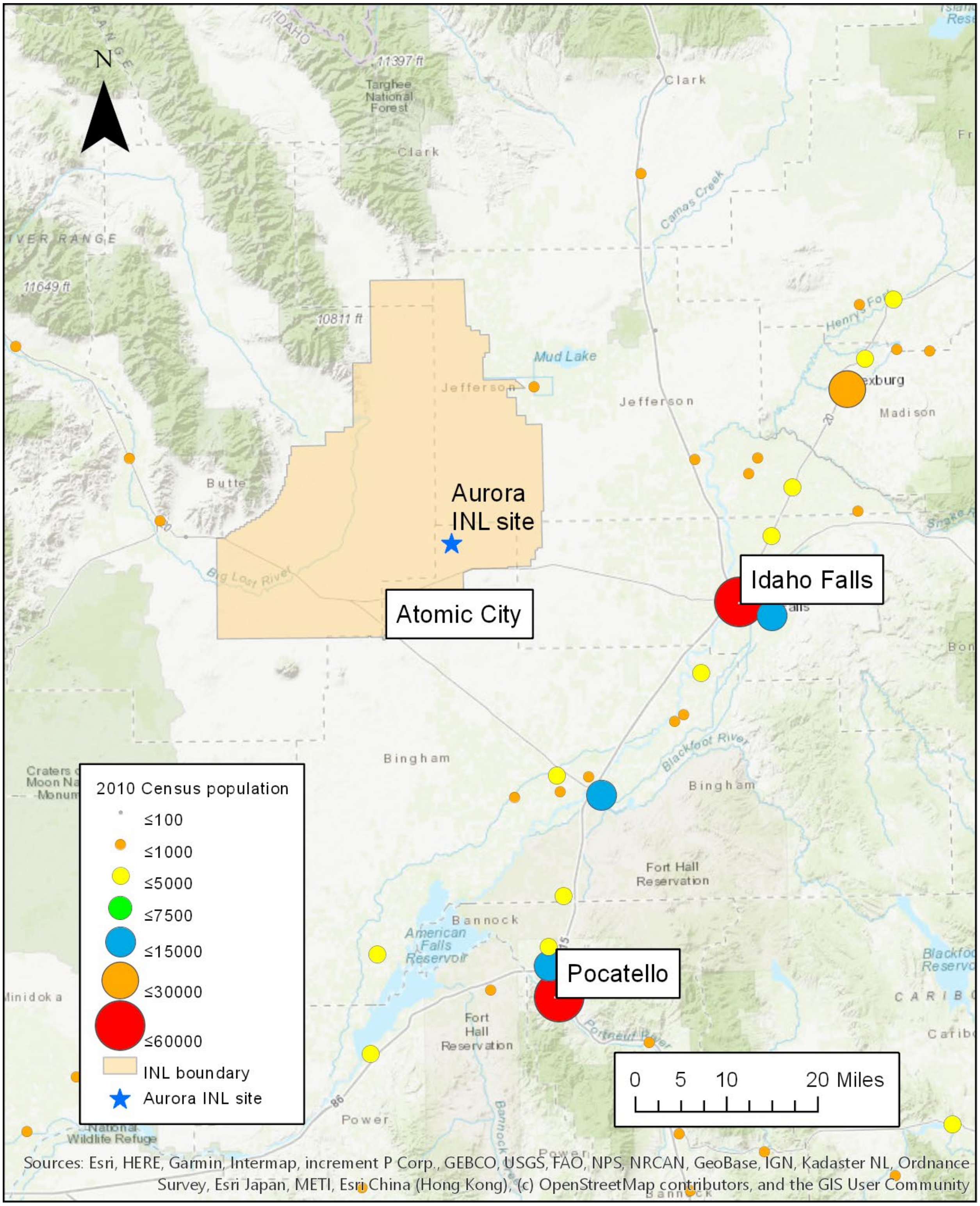

Figure 1-17: Population surrounding INL [13]

1.2.2.7 Atmospheric dispersion characteristics evaluation

Atmospheric dispersion characteristics of a site are typically provided because they are key in determining the effect of a pollutant from the plant to the local environment. However, the Aurora does not have significant effluent inventories nor a credible accident that could lead to a fission product release, as discussed in Section 1.3, so atmospheric dispersion parameters are not needed to be considered for the proposed site. Therefore, the information provided in this section is strictly to meet the regulations and is not needed for determining the safety of the plant at the proposed site.

1.2.2.7.1 Generic site envelope consideration

The generic site envelope does not contain any evaluations related to atmospheric dispersion that need to be performed for the proposed site to ensure the safety of the Aurora.

1.2.2.7.2 Proposed site considerations and evaluations

This purpose of this section is to satisfy the requirements of 10 CFR 100.21(c), which requires site-specific information for atmospheric dispersion characteristics for radiological effluent release limits, associated with normal operations, and for radiological dose consequences of postulated accidents.

1.2.2.7.2.1 Radiological effluent release limits

The Aurora utilizes almost no fluids during normal operation. The two fluids that are used of note are: (1) the backfill noble gas, sealed in the reactor module; and (2) the secondary system coolant, sealed in the secondary system. Because these two fluids do not become significantly activated, as discussed in Chapter 3, “Radioactive materials to be produced in operation,” and are completely contained within the respective systems over plant life, the Aurora does not have anticipated normal effluent releases during normal operations. Therefore, atmospheric dispersion parameters are not used, and no further information is provided.

1.2.2.7.2.2 Radiological dose consequences of postulated accidents

The Aurora does not have a credible accident that could lead to a fission product release, as discussed in Section 1.3, therefore atmospheric dispersion parameters are not used, and no further information is provided.

1.2.2.8 Security plans impact evaluation

Specific site characteristics have historically been taken into account when evaluating potential sites to determine if there are physical characteristics of the site that could impact security plans. However, the security plans used for the Aurora are generally transferable to most sites due to the small size of security footprint. The Physical Security Plan is included under Part VII, “Enclosures.”

1.2.2.8.1 Generic site envelope consideration

The generic site envelope does not contain any security plan impact evaluations that need to be performed for the proposed site, as security plan impact evaluations are outside of the scope of the external hazards evaluation.

1.2.2.8.2 Proposed site considerations and evaluations

This purpose of this section is to satisfy the requirements of 10 CFR 100.21(f), which requires that, “site characteristics must be such that adequate security plans and measures can be developed.”

The Physical Security Plan is simple and are generally transferable between sites. The proposed site does not have physical characteristics, such as challenging topography, that could limit the actions required by the Physical Security Plan. Additionally, security response times are not challenged at the proposed site, as the security force responsible for the Aurora is located adjacent to the Aurora INL site.

1.2.2.9 Emergency Plan impact evaluation

Regional population is not important to the operations nor the safety of the plant since the emergency planning zone (EPZ) boundary is the Aurora powerhouse.

Specific site characteristics have historically been taken into account when evaluating a potential site to determine if there are physical characteristics of the site that could impact the emergency plan of a facility. However, the Aurora Emergency Plan is generally transferable to most sites due to the small EPZ and minimal necessary actions. The Emergency Plan is included under Part VII.

1.2.2.9.1 Generic site envelope consideration

The generic site envelope does not contain any emergency plan impact evaluations that need to be performed for the proposed site, as emergency plan impact evaluations are outside of the scope of the external hazards evaluation

1.2.2.9.2 Proposed site considerations and evaluations

This purpose of this section is to satisfy the requirements of 10 CFR 100.21(g), which requires that, “physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans must be identified.”

The Aurora Emergency Plan is simple and is generally transferable between sites. The proposed site does not have unique physical characteristics, such as challenging topography, that could limit the actions required by the Emergency Plan. Additionally, the Aurora EPZ is small, and the actions required by the Emergency Plan pertain largely to actions within the Aurora INL site. Emergency Plan offsite responder response times are unchallenged by the proposed site because the offsite responders are located adjacent to the Aurora INL site.

1.3 Safety assessment of the proposed site

The purpose of this section is to address the requirements of 10 CFR 52.79(a)(1)(vi). However, since there is no credible accident that could lead to a fission product release, it is not assumed in this assessment; the relevant exemption is located in Part V, "Non-applicabilities and requested exemptions." Because there is no credible accident that could lead to a fission product release, the requirements of 10 CFR 52.79(a)(1)(vi)(A) and 10 CFR 52.79(a)(1)(vi)(B) are met by default.

Appendix A: External hazards evaluation

A.1 Overview of methodology

Since a design objective of the Aurora is the ability to be sited in the majority of the U.S., external hazards must be evaluated for a wide range of possible extreme events. Due to the small size of the Aurora and the simple safety case, it is possible to evaluate the design against most bounding extreme external hazards for the U.S.

Traditional external hazards evaluations utilize a probabilistic risk assessment (PRA) to determine which external hazards are likely to occur for a given site. Alternatively, it is possible to use deterministic analyses as a method of screening external events from a PRA. Deterministic analyses can show the resiliency of the facility against extreme external hazards and obviate the further analysis typically done by a PRA. For most external hazards, deterministic analyses have been performed for the Aurora. Hazards associated with accidents resulting from purposeful human-induced security threats (e.g., sabotage, terrorism) and risks associated with accidental radiological exposures to onsite personnel are explicitly excluded from the external hazards evaluation.

The steps for analyzing external hazards are the following:

1. Identify all potential external hazards that may affect the plant considering all plant operating states.

2. Perform a preliminary screening to group external hazards that have a common challenge or to identify the external hazards as site-dependent.

3. Define event families for the external hazards grouped by a common challenge.

4. Perform bounding deterministic analyses for each event family, evaluated against quantitative screening criteria.

5. Define commitments to address limitations from the bounding deterministic analyses or to resolve site-dependent external hazards.

These steps are also shown in Figure 1-18.

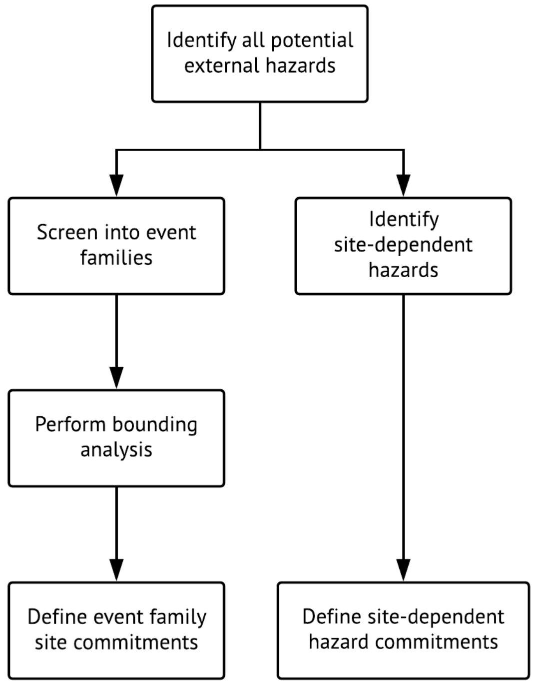

Figure 1-18: External hazards methodology overview

A.2 Definitions

bounding event: A challenging event at the most extreme end of the range of possible events.

challenge, common challenge: The specific phenomenon that could pose damage to the Aurora facility.

deterministic analysis: A type of analysis that does not take probabilities into account, and instead assumes specific causes completely and certainly determine specific outcomes.

event family: A method of grouping common challenges for the purposes of conducting a single, bounding, deterministic analysis.

generic site envelope: The set of parameters the site must meet to ensure the site does not adversely impact the safety of the Aurora.

site bases: Siting principles that assure that the safety of the Aurora is not affected due to the natural features of the proposed site.

site commitment: A commitment to perform a specific action when undergoing site selection for the Aurora. Site commitments are derived from the external hazards evaluation.

site-dependent hazard: An external hazard whose impacts are mitigated through site commitments.

A.3 Identification of all potential external hazards

This section follows step one of the external hazards methodology. To ensure a full range of potential hazards are taken into consideration, a thorough literature review was done, including:

- ASME/ANS RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications"
- ASME/ANS RA-S-1.4-2013, "Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants"
- NUREG/CR-2300, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants"
- NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States "

From the above literature, the potential external hazards identified as the first step of this methodology are listed in Table 1-6.

Table 1-6: All potential external hazards

Avalanche	Hurricane	Seismic event
Biological events	Ice cover (causing blockage of river)	Sinkholes
Coastal erosion	Landslide	Snow
Drought	Lightning	Soil shrink-swell
External flooding	Low lake or river water level	Storm surge
Forest fire	Low winter temperature	Tornadoes (extreme winds)
Frost	Non-safety building fire	Toxic gas
Grass fire	Precipitation, intense	Transportation accident [1] [2]
Hail	Release of chemicals from onsite storage	Tsunami
High summer temperature	River diversion	Turbine-generated missiles
High tide	Sandstorm	Volcanic activity
High winds	Seiche	Waves

[1] - Transportation accidents include the following external hazards: aircraft impact, industrial or military facility accident, pipeline accident, railroad accidents, ship impact, vehicle impact, and vehicle/ship explosion. This binning of transportation accidents is in accordance with accepted methodology in NUREG-5042.

[2] - The fog external hazard is not included as an independent event on this list. Fog is an external hazard that raises the probability of transportation accidents, because it increases the likelihood of man-related error. Since no probabilistic risk assessment is performed on the transportation accident hazard, the fog external hazard is not included as an independent event.

A.4 Screening external hazards for common challenges

All of the potential external hazards, shown in Table 1-6, are evaluated to determine the challenge to the facility. A challenge is defined as the specific phenomenon that could pose damage to the Aurora facility. For hazards that present multiple challenges to the facility, the hazard is screened into all challenges.

A.4.1 External hazards that do not apply to the Aurora

The following external hazards do not apply to the Aurora design:

- Biological events
- Drought
- Lightning
- Low lake or river water level
- Non-safety building fire
- Release of chemicals from on-site storage
- River diversion
- Soil shrink-swell
- Toxic gas
- Turbine-generated missiles

Biological events, drought, low lake or river water level, and river diversion are hazards that are associated with a loss of cooling via a natural water source. Biological events include growths such as detritus and zebra mussels that degrade the ultimate heat sink performance. Since the Aurora does not depend on water for cooling, these hazards do not apply.

Lightning is a hazard that is associated with a loss of power to the facility. Since the Aurora does not depend on electric power for any safety function, this hazard does not apply.

Non-safety building fire is a hazard that is associated with a fire, which originates in a non-safety building and propagates to a safety building. However, the Aurora has only one major building onsite, which is the Aurora powerhouse. Since internal fires in the powerhouse are already analyzed and there are no other major buildings onsite, this hazard does not apply.

Release of chemicals from onsite storage and toxic gas are hazards that are associated with incapacitation of onsite personnel. Since the Aurora does not have any safety-related human actions, this hazard does not apply. Human actions that are discussed in this license application related to emergency preparedness, security plans, and others, are largely required by operating procedures to ensure the safety of the onsite personnel and are not needed to assure the safety of the Aurora reactor. Further, it is important to note that the Aurora does not contain large amounts of chemicals onsite.

Soil shrink-swell is a hazard that occurs when soils expand when wet and retract when dry. This phenomenon can cause cracks in the foundation, leading to a reduction in structural strength and an opportunity for differential displacement between buildings on a site. Cracking does not immediately result in significant structural damage and can be managed by maintenance. Differential displacement is not a concern for the Aurora since there is only one major building onsite. Therefore, the soil shrink-swell hazard does not apply.

Turbine-generated missiles are hazards that originate from the dislodging of a turbine blade from a large, typically steam-driven, turbine. The turbine utilized by the Aurora is of significantly different design than a typical steam-driven turbine. It is smaller, roughly the size of a desk in length and cannot propel a missile in a similar way to a steam-driven turbine. Even if it could, such a small turbine missile would not pose a challenge to the facility because it would likely not travel past the secondary system casing. Additionally, the secondary system and reactor module are located on different elevations, such that a turbine-generated missile would not be able to reach the reactor module. Therefore, the turbine-generated missiles external hazard does not apply.

These external hazards do not pose a challenge to the safety of the Aurora, because they do not apply to the design of the facility. Therefore, these external hazards are not further analyzed as part of the external hazards evaluation.

A.4.2 External hazards that could challenge the heat sink

External hazards that could challenge performance of the heat sink are as follows:

- Frost
- High summer temperature
- Low winter temperature
- Volcanic activity

Frost, high summer temperature, and low winter temperature are hazards that are associated with unexpected extreme temperature deflections that could challenge the heat sink. Volcanic activity is a hazard that is associated with potential ash in the region and could challenge the heat sink. All of these hazards could potentially cause a challenge to the cooling of the secondary system at the Aurora, which would degrade the capability of the secondary system.

The secondary system is what removes heat from the Aurora reactor. A complete loss of secondary system is called a “loss of heat sink” in the internal events analysis, located in Chapter 5.1 of the final safety analysis report. A loss of heat sink results in a reactor trip for the Aurora. Due to the simple nature of the Aurora, only decay heat needs to be analyzed following a complete loss of heat sink. Because the decay heat of the reactor is so small, no fuel damage is possible in the event of a loss of heat sink, even for a long period of time.

Therefore, since a complete loss of heat sink is already analyzed as an internal event and does not challenge the safe state of the reactor, these external hazards are not further analyzed as part of the external hazards evaluation.

A.4.3 External hazards that could result in an explosion

The external hazard that could challenge the reactor module by a high incident pressure, due to an explosion, is a transportation accident.

A transportation accident could result in an explosive force nearby the Aurora powerhouse from incidents such as a pipeline accident, railroad accident, or an industrial or military facility accident. The specific phenomenon associated with an explosive force is the resulting pressure wave to nearby structures. Specifically, the resulting incident pressure on the reactor module is of interest in this analysis and the reactor module is analyzed further for overpressure.

Therefore, a high incident pressure on the reactor module is further analyzed, and these hazards are grouped under the "explosions event family" for the purposes of the external hazards evaluation.

A.4.4 External hazards that could cause a fire

The external hazard that could challenge the facility by a fire inside the Aurora powerhouse is a transportation accident.

Transportation accidents relate to industrial accidents that could potentially cause a fire inside the powerhouse from incidents such as a vehicle impact or an industrial or military facility accident. These transportation accidents could cause an internal fire from the leak of a flammable fluid inside the powerhouse. Fires within the building are analyzed in the deterministic Fire Hazards Analysis (FHA), which assumes a large fire inside the powerhouse.

Therefore, the external hazards that can cause a fire inside the building are grouped under the "fire event family" and are further discussed in this external hazards evaluation.

A.4.5 External hazards that could cause a flood

External hazards that could challenge the facility by water inside the Aurora powerhouse are as follows:

- External flooding
- High tide
- Hurricane
- Ice cover (causing blockage of river)
- Precipitation, intense
- Seiche
- Snow
- Storm surge
- Tsunami
- Waves

The above external hazards could cause unexpected water levels to approach the facility. Although it is unlikely that water could enter the facility, the phenomenon of interest is an analysis of water in the basement of the powerhouse, where the reactor module is located.

Therefore, these external hazards that could cause standing water in the powerhouse are grouped under the "flood event family" and are further discussed in this external hazards evaluation.

A.4.6 External hazards that could challenge the Aurora powerhouse

External hazards that could challenge the integrity of the powerhouse are as follows:

- Hail
- High winds
- Precipitation, intense
- Sandstorm
- Seismic event
- Snow
- Tornadoes (extreme winds)
- Transportation accident

Hail, high winds, sandstorm, and tornadoes (extreme winds) are hazards associated with extreme winds that could also potentially cause damage to the powerhouse. Precipitation (intense) and snow are hazards associated with extreme roof loadings that could also potentially cause damage to the powerhouse. A transportation accident could potentially cause damage to the powerhouse by a collision from a vehicle impact or an industrial or military facility accident. Seismic events have an associated ground acceleration, which could cause structural damage, and could cause damage to the powerhouse.

Therefore, damage to the powerhouse is further analyzed. These hazards are grouped under the "seismic event family" for purposes of the external hazards evaluation.

A.4.7 External hazards that could cause a ground acceleration

The external hazards that could challenge the integrity of the reactor module by a ground acceleration are as follows:

- Seismic event
- Tsunami

A tsunami or an independent seismic event could have large ground accelerations that would cause structural loading on the reactor module.

Therefore, structural loading of the reactor module is further analyzed, and these hazards are grouped under the "seismic event family" for purposes of the external hazards evaluation.

A.4.8 Summary of external hazards screening for common challenges

The above screening of external hazards resulted in the external hazard being identified in one of the following ways:

- Is not applicable to the Aurora design
- Relates to a heat sink challenge
- Has a common challenge
- Is site-dependent

The external hazards that are not applicable to the Aurora design are not further analyzed. The external hazards that relate to a heat sink challenge are part of the internal safety analysis and are not further discussed in this external hazards evaluation. The external hazards that have a common challenge are grouped together into event families and analyzed in this external hazards evaluation. The external hazards that are site-dependent are handled through site commitments and are described further in this external hazards evaluation. The external hazards screening is summarized in Table 1-7.

Table 1-7: Summary of external hazards screening

Hazard	Not applicable	Heat sink challenge	Common challenge	Site-dependent
Avalanche				x
Biological events	x			
Coastal erosion				x
Drought	x			
External flooding			x	
Forest fire				x
Frost		x		
Grass fire				x
Hail			x	
High summer temperature		x		
High tide			x	
High winds			x	
Hurricane			x	
Ice cover (causing blockage of river)			x	
Landslide				x
Lightning	x			
Low lake or river water level	x			
Low winter temperature		x		
Non-safety building fire	x			
Precipitation, intense			x	
Release of chemicals from onsite storage	x			
River diversion	x			
Sandstorm			x	
Seiche			x	
Seismic event			x	
Sinkholes				x
Snow			x	
Soil shrink-swell	x			
Storm surge			x	
Tornadoes (extreme winds)			x	
Toxic gas	x			
Transportation accident			x	
Tsunami			x	
Turbine-generated missiles	x			
Volcanic activity		x		
Waves			x	

A.5 Event family definitions

The purpose of this section is to define event families, based on a common challenge. The external hazards that have a common challenge fall into one of the following categories:

- Result in an explosion
- Cause a fire
- Cause a flood
- Challenge the Aurora powerhouse
- Cause a ground acceleration

Hazards that could result in an explosion are analyzed by assuming a large explosion nearby the facility and are included in the explosions event family. Hazards that could cause a fire are analyzed by assuming an internal fire and are included in the fire event family. Hazards that could cause a flood are analyzed by assuming an internal flood and are included in the flood event family. Hazards that could challenge the powerhouse and the reactor module are analyzed by a powerhouse collapse and large earthquake, respectively, and are included in the seismic event family. Therefore, the resulting event families are explosion, fire, flood, and seismic. A summary of the event families, their common challenge, the bounding event analyzed, and the bound hazards is shown in Table 1-8.

Table 1-8: External hazard event families

Event family	Challenge	Bounding event	Bound hazards
Explosion	High incident pressure on the reactor module	Large nearby explosion	Transportation accident
Fire	Fire in the powerhouse	Internal fire	Transportation accident
Flood	Water in the powerhouse	Internal flood	External flooding High tide Hurricane Ice cover (causing blockage of river) Precipitation, intense Seiche Snow Storm surge Tsunami Waves
Seismic	Integrity of the powerhouse	Powerhouse collapse	Hail High winds Precipitation, intense Sandstorm Seismic event Tornadoes (extreme winds) Transportation accident Tsunami
	Integrity of the reactor module	Large earthquake	Seismic event

For the event families, bounding events are selected for deterministic analyses that are performed to evaluate the resilience of the Aurora. Each event family deterministic analysis is discussed in this external hazards evaluation, in the following sections:

- Section A.6 Explosion event family analysis
- Section A.7 Fire event family analysis
- Section A.8 Flood event family analysis
- Section A.9 Seismic event family analysis

The quantitative criterion used for the bounding deterministic analyses is that the consequences from any release would not cause a whole-body projected dose more than 1 rem over four days as given in the Environmental Protection Agency's Protective Action Guides Manual, "Protective Action Guides and Planning Guidance for Radiological Incidents," published January 2017.

A.6 Explosion event family analysis

The external hazard that could challenge the facility by a nearby explosion is a transportation accident.

This hazard could cause an explosion nearby the facility. Depending on the size of the explosion, the resulting pressure could pose a challenge to the facility. A large explosion could result in a full collapse of the Aurora powerhouse; that phenomenon is not evaluated in this section because a powerhouse collapse is examined as part of the seismic event family.

The parameter of interest is the ability to shut down the reactor. Reactor shutdown is achieved by the shutdown rods (see Chapter 2 of the final safety analysis report), which are normally held in place by electromagnets. On a loss of power to the magnets, the shutdown rods drop into the reactor and reactor shutdown is achieved. Because of the simple design of the shutdown rods, the system of interest in an explosion is the reactor enclosure system (see Chapter 2 of the final safety analysis report). Specifically, the module equipment housing, which protects the shutdown rod electromagnets and drive lines, as can be seen in Figure 1-19. If the integrity of the module equipment housing is maintained, the shutdown rods are not prevented from dropping into the reactor.

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Figure 1-19: Schematic showing the shutdown rod components housed within the module equipment housing

Therefore, the goal of the explosion event family is to determine the maximum explosion that could occur nearby the Aurora, without exceeding material limits of the module equipment housing. Instead of running a sensitivity study on the module equipment housing to calculate effects of differently-sized explosions, the blast capacity of the module equipment housing is found. The blast capacity, as used in this chapter, is the maximum static overpressure allowable, before material limits are exceeded.

A.6.1 Explosion event family analysis

The blast capacity of the module equipment housing was determined by applying directional pressure until material limits were exceeded. This pressure was approximated using various idealized hand calculations for plates, cylinders, and rings with uniform load conditions, which is extremely conservative. If computer modeling is performed on the same components, it is likely that the blast capacity would be significantly higher. Additionally, the pressure calculated was a static overpressure, which is extremely conservative for this type of analysis, as opposed to a dynamic overpressure, which would result in a much higher blast capacity.

The following assumptions were applied in this analysis:

- The static overpressure is assumed to be directional and only applied to half of the cylinder portion of the module equipment housing.
- The module equipment housing is assumed to be stainless steel 304.

The cylinder and top plate of the module equipment housing are evaluated separately analytically [14]. The cylinder was evaluated for the nominal static load that would reach the elastic limit of the module equipment housing in bending. Next, the cylinder was evaluated for through uniform radial pressure loading, with the ends capped. Third, the elastic stability of the cylinder was investigated and idealized as a curved panel with uniform load. Fourth, the limiting static pressure of the cylinder was evaluated by idealizing the cylinder as a plate with orthogonal dimensions accounting for the arching effects due to the shape of the cylinder. From these four steps, the static pressure capacity was found for the cylinder. Similarly to the cylinder, idealized hand calculations were performed for the top plate of the module equipment housing. { }

To validate the analytical calculations, an idealized LS-DYNA finite element analysis was performed. A simple elastic model was created to represent the module equipment housing, with a fixed boundary condition at the bottom of the module equipment housing. Two simulations were performed, linearly varying pressure-time loads. A time step was determined at which the elastic limit was exceeded for the module equipment housing. Next, a load-curve correlation was applied to determine the corresponding pressure. { } This analysis confirms the hand calculations as appropriately conservative.

A.6.2 Explosion event family analysis results

{ } Therefore, the explosion event family analysis resulted in one site commitment, discussed in Section A.6 Explosion event family analysis."

A.7 Fire event family analysis

The external hazard that could challenge the facility by a fire inside the Aurora powerhouse is a transportation accident.

A transportation accident could cause a fire nearby the powerhouse. For purposes of this deterministic analysis, the fire is assumed to enter the facility. Therefore, this external hazard is analyzed to be the initiator to an internal fire and is already addressed through an FHA. The FHA is included in Chapter 7 of the final safety analysis report. The purpose of this section of the external hazards evaluation is to summarize the FHA.

A.7.1 Fire event family analysis methodology

The goal of the FHA is to describe the Fire Protection Program (FPP) and to conduct a fire hazards analysis. The primary objectives of the FPP are to minimize both the probability of occurrence and the consequences of fire. The fire hazards analysis contains the following:

- An evaluation of the potential in situ and transient fire hazards
- A determination of the effects of a fire in any location in the plant, including the impact on the ability to achieve a safe state and minimizing the risk of radioactive release to the environment
- A determination of the appropriate measures for fire prevention, fire detection, fire suppression, and fire containment for each area containing components necessary for achieving a safe state

The goal of the FHA is to analyze the consequences of a single, credible fire. Specifically, the FHA analyzes whether such a fire could impede placing the reactor in a safe state. Each fire area in the Aurora is analyzed for the effects of a single, credible fire and summarized in this section.

Fire areas are separated by rated fire barriers capable of protecting the components necessary to achieve and maintain the safe state, and each fire area contains areawide detection. The fire areas are shown in Figure 1-20 and are the following:

- Fire area 1 – Atrium
- Fire area 2 – Power conversion system (PCS) area
- Fire area 3 – Control cabinet 1
- Fire area 4 – Control cabinet 2
- Fire area 5 – Basement

Fire hazards are characterized as a combination of ignition sources and combustible material. It is important to note that each fire area, other than fire area 1 (atrium), contains limited combustible materials; no combustible liquid is maintained inside fire areas 2-5. Therefore, the deterministic assumption of a fire in each fire area is extremely conservative, as there are no likely combustion or ignition sources to be the source of the fire.

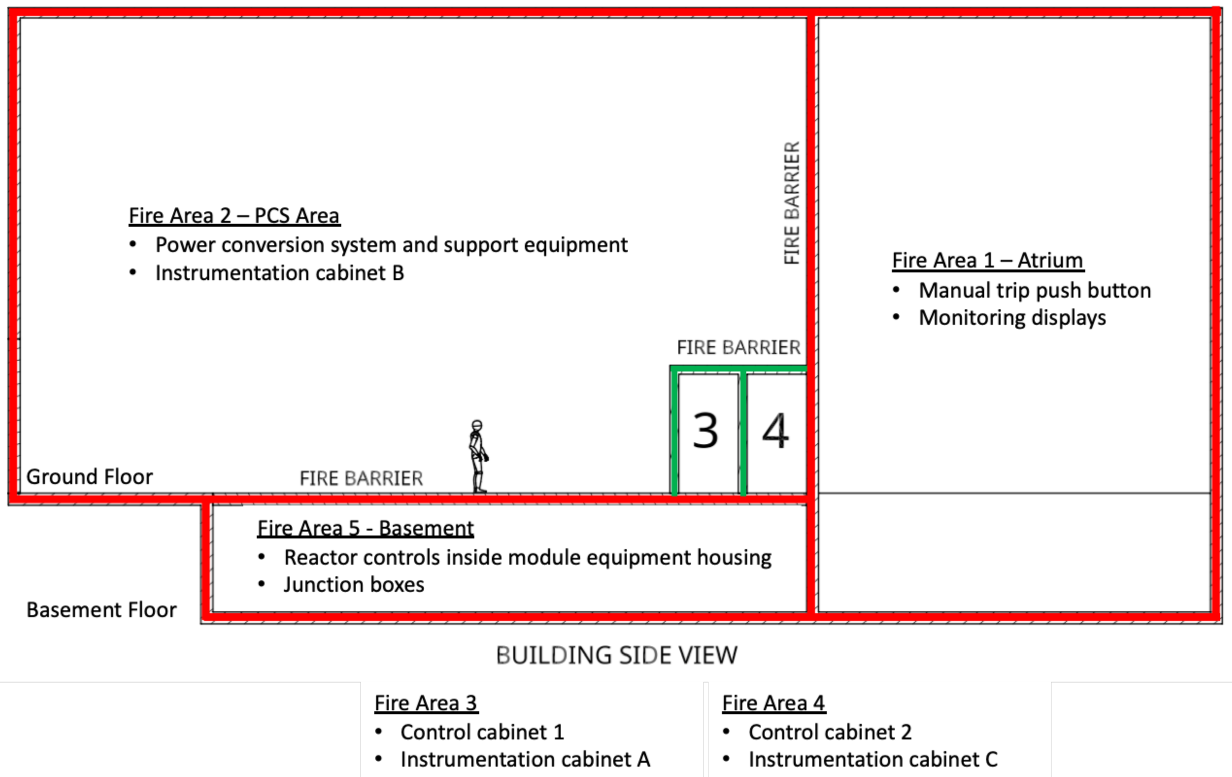

Figure 1-20: Fire areas in the Aurora facility

A fire in fire area 1 has no effect on achieving the safe state. A single, credible fire in fire area 1 does not challenge the secondary system, therefore reactor heat removal is maintained. Since reactor heat removal is maintained, there is no need to shut down the reactor. Additionally, it does not affect the capability of the automatic reactor trip, the logic for which is housed in control cabinets 1, and 2.

A fire in fire area 3 or 4 is assumed to disable the control cabinet housed in that fire area. The Aurora is designed with two redundant control cabinets, which are used to initiate an automatic reactor trip. Control cabinet 1 and 2 are located in fire area 3 and 4, respectively. Disabling one control cabinet in either fire area does not challenge achievement of the safe state because the automatic reactor trip is protected by the redundancy of the control cabinets.

A fire in fire area 2 is assumed to fully disable the secondary system, which is normally responsible for providing heat removal from the reactor module. It is important to note that it is extremely conservative to assume, in an area with little to no combustion or ignition sources, that a fire is capable of fully disabling heat removal from the reactor module via the secondary system. This scenario is similar to what is analyzed for a total loss of heat sink (i.e., loss of secondary system) in the internal events analysis in Chapter 5.1. Nevertheless, the achievement of the safe state is unchallenged, because the automatic reactor trip logic is housed in fire area 3 and 4 and remains capable of initiating a reactor trip.

A fire in fire area 5 (basement) is assumed to fully disable all control logic cables that are present in the basement. These cables are how control cabinets 1 and 2 communicate with the instrumentation located in the reactor module. It is important to note that a fire in the basement cannot cause an internal transient, namely a reactivity insertion, due to the design of the components in the reactor module. Therefore, the only consequence of a fire in the basement is the temporary loss of communication via the control logic cables due to a postulated circuit failure, with no reactor transient. Further, a fire in fire area 5 cannot disable the automatic reactor trip, because the shutdown rod electromagnets open on a loss of power and drop the shutdown rods into the reactor. This is ensured by wrapping the power cables to the electromagnets in conduit without compatible energized sources, and by ensuring that the automatic trip generated in the control cabinet removes the all power to the cables. Therefore, a fire in fire area 5 does not challenge achieving the safe state.

A.7.2 Fire event family analysis results

For purposes of the fire event family, a fire was assumed inside the Aurora facility and was deterministically analyzed through the FHA. The FHA further analyzed whether the safe state of the facility was challenged. The safe state was defined as the reactor reaching a shutdown condition following a single, credible fire. Next, a single, credible fire was assumed in each fire area and the impacts to the automatic reactor trip were analyzed. Because of the robustness of fire protection included early in the design of the Aurora, a single, credible fire is not capable of preventing the ability to reach the safe state. Therefore, the fire event family analysis did not result in any site commitments.

A.8 Flood event family analysis

External hazards that can challenge the facility by water inside the powerhouse are as follows:

- External flooding
- High tide
- Hurricane
- Ice cover (causing blockage of river)
- Precipitation, intense
- Seiche
- Snow
- Storm surge
- Tsunami
- Waves

The above external hazards are associated with large amounts of unexpected water potentially entering the powerhouse. Most events that are associated with extreme weather typically have sufficient advance notice to prepare a facility against the associated effects. External hazards that could cause standing water to enter the facility are almost always associated with extreme weather, therefore, they are rarely unanticipated. The ability to predict possible flood events provides the ability to prepare for a potential flood.

The design of the Aurora is such that the Aurora powerhouse is enclosed by a large berm. The purpose of the berm is for investment protection reasons, and it is not included for any safety reasons. The berm is simply ground around the powerhouse that has been artificially elevated as can be seen in Figure 1-21.

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Figure 1-21: Berm surrounding the Aurora powerhouse

Even if flooding was able to reach the site, the water would still have to make it through the berm to the powerhouse. Further, if the water was able to make it on the site, past the berm, and into the powerhouse, water would need to advance to the basement to be of interest in this analysis.

The Aurora powerhouse has two elevations: (1) a ground floor, and (2) a basement. The elevation of interest in this analysis is the basement since it contains the reactor module. The square footage of the basement is very large, as it compares the remainder of the building. Therefore, for enough water to accumulate in the basement such that it can reach a height great enough to enter the area around the reactor module is very unlikely.

Nevertheless, the goal of this external hazards flood event family analysis is to analyze the effects of standing water in the basement. Specifically, it is of interest to examine the effects of water in the basement as they relate to the removal of heat from the reactor and reactivity effects on the reactor.

A.8.1 Flood event family analysis methodology

There are two phenomena of interest related to standing water in the facility: (1) effects of heat transfer from the reactor module, and (2) reactivity effects to the reactor. The goal of the heat transfer analysis is to evaluate heat transfer effects from the reactor module. This parameter is specifically of interest because reactor decay heat is credited in the safety analysis to be removed only by passive means through natural convection to air. Since there are no systems in the Aurora for the removal of decay heat, it is of interest to analyze the effects of water in the basement to the capability to remove decay heat.

The goal of the reactivity analysis is to ensure that even in the most limiting shutdown state, the reactor cannot increase unexpectedly in reactivity and reach criticality. The most limiting shutdown state is at the coldest temperature, which is assumed to be room temperature.

The module is designed with a large gap, called the reactor cavity, between the outer surfaces of the shell and the inner surfaces of the reactor emplacement. The reactor module and reactor emplacement configuration can be seen in Figure 1-22. The purpose of the reactor emplacement is to provide structural support for the reactor module.

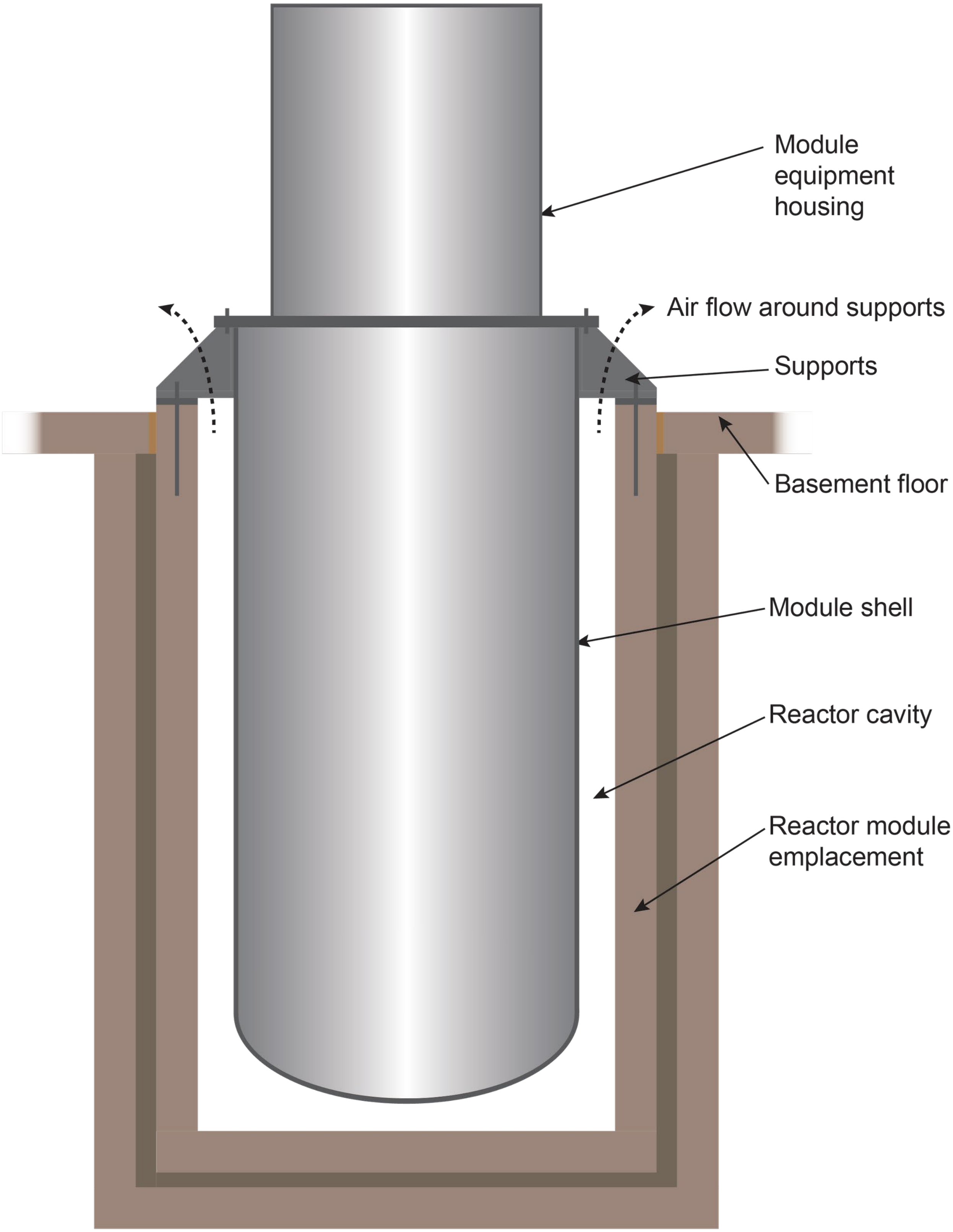

Figure 1-22: Reactor cavity between the reactor module and the reactor emplacement

The reactor cavity is designed to provide a sufficient gap such that air can remove the decay heat following a reactor shutdown through natural convection. As described in the safety analysis, in Chapter 5.1, "Transient analysis," the most challenging event to the Aurora is a complete loss of the secondary system, referred to as a loss of heat sink. The secondary system is the primary mode of heat removal from the reactor during normal operations. Following a loss of heat sink, the reactor is shutdown, and decay heat is removed passively from the reactor module to the air in the reactor cavity, and ultimately to the environment. Specifically, this decay heat removal occurs by natural convection from air circulating through the reactor cavity. Therefore, the effects of water in the basement to this heat removal are important to understand for the safety case.

The basement is the lower elevation of the Aurora powerhouse and contains the reactor module. Surrounding the reactor module is a slight elevation in the basement floor, which is standard practice for industrial facilities and is included for investment protection reasons. This slight elevation gain is described as a reactor module curb and can be seen schematically in Figure 1-23. The purpose of the reactor module curb is to maintain the cleanliness of the reactor module and reactor cavity and is mostly intended for maintenance intervals; there are no safety implications to this design feature.

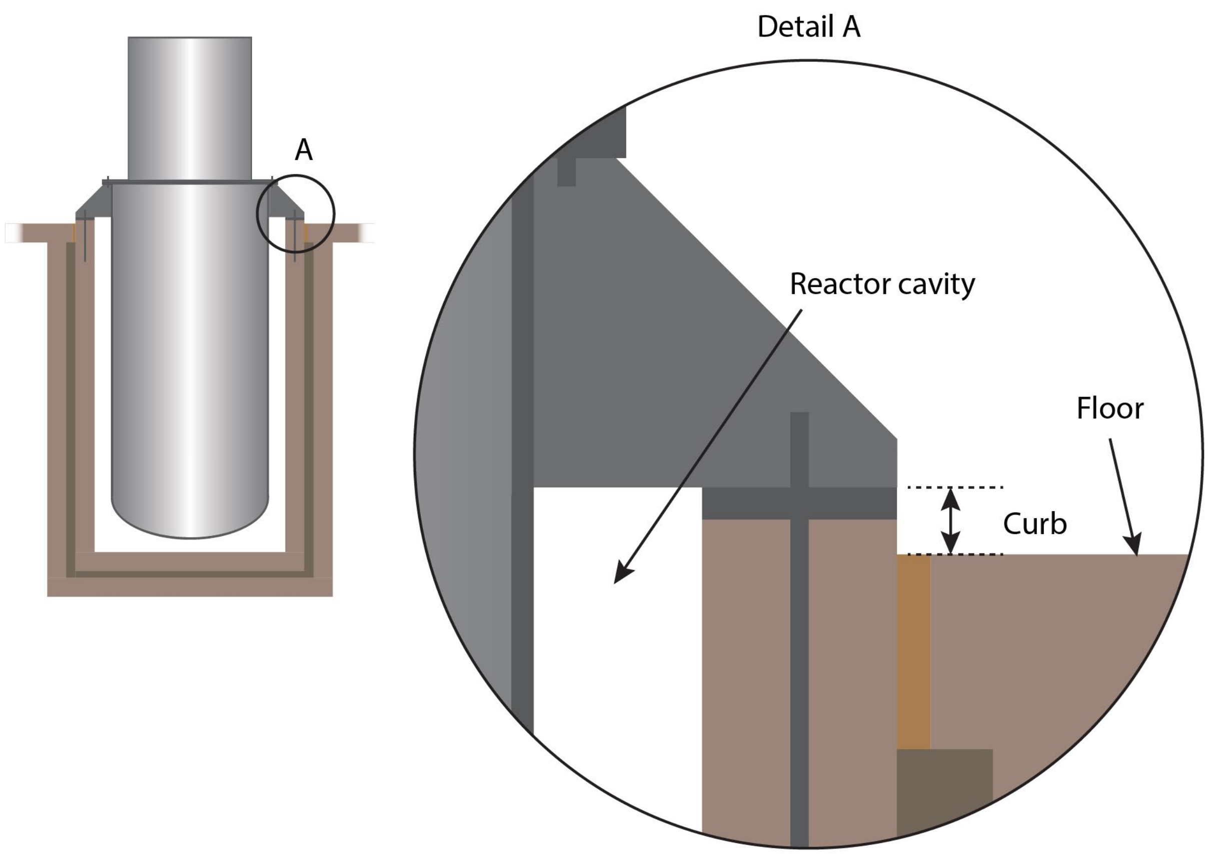

Figure 1-23: Emplacement of reactor module in basement with emphasis on reactor module curb

In the event that water reaches the site, penetrates through the berm, enters the powerhouse, and is able to reach the basement, it is extremely unlikely the water would make it over the reactor module curb. The basement area is large, as compared to the rest of the facility, such that for enough standing water to accumulate so that it is able to overcome the height of the reactor module curb, over 30,000 gallons of water would have to reach the basement. It is unlikely this much water could enter the basement suddenly, with no warning to the Aurora staff. Nevertheless, for the heat transfer and reactivity effects analyses, water is assumed to overcome the reactor module curb and enter the reactor cavity.

A.8.1.1 Flooding heat transfer methodology

During normal operations, the reactor cavity is filled with air, and air flow is driven by natural convection from the surface of the reactor module. The heat transfer coefficient from the reactor module to the air in the reactor cavity is described in the safety analysis, located in Chapter 5.1 of the final safety analysis report. This coefficient is estimated at the same value during both normal operations and following shutdown. The heat transfer from the surface of the reactor module to the fluid in the reactor cavity can be described as follows:

$$\dot{q} = hA_{shell}(T_{shell} - T_{fluid}) \tag{1}$$

Where $\dot{q}$ is the heat transfer rate, h is the heat transfer coefficient, A_{shell} is the heat transfer area on the surface of the reactor module shell, T_{shell} is the temperature of the reactor module shell, and T_{fluid} is the temperature of the fluid. In the unlikely event of a flood that allows water to exceed the height of the reactor module curb and start flowing into the reactor cavity, water would initially fill up the lower volumes of the reactor cavity. In an even more unlikely event, water would fill up the entire volume of the reactor cavity.

A.8.1.2 Flooding heat transfer results

The heat transfer coefficient from the reactor module to water, driven by natural convection and conservatively ignoring the effects of evaporation, is estimated to be about an order of magnitude higher than that of nominal conditions, from reactor module to air [15]. If boiling effects are included, the heat transfer coefficient could be as high as four times larger than nominal conditions [15]. These increased heat transfer coefficients mean that heat removal from the surface of the reactor module to the fluid in the reactor cavity, be it air, water, or a mix, is significantly increased. Since one of the goals of the safety analysis is to show that there is sufficient heat transfer from the reactor module to the fluid in the reactor cavity, increasing the heat transfer coefficient only increases the heat removal from the reactor module. Therefore, the effects of water in the reactor cavity put the reactor in a less challenging state than analyzed in the safety analysis.

A.8.1.3 Flooding reactivity effects methodology

Similar to the flooding heat transfer effects approach, the unlikely event in which water is able to partially, or fully fill, the reactor cavity is considered in this analysis. The most limiting state is assumed to be a shutdown state at the coldest temperature. Room temperature is assumed because reactivity is higher at room temperature than at hot temperatures; therefore, flooding at higher temperatures is bounded by this analysis.

The reactor cavity is substantially far (i.e., over one meter from the outermost reactor cell) from the active portions of the core and separated from the active core by a significant amount of shielding. Nonetheless, reactivity effects from water in the reactor cavity, primarily through increased neutron moderation, are considered. For this analysis, flooding is assumed to fully fill the reactor cavity with water to bound the effects of a partial flood of the reactor cavity.

Furthermore, for this analysis, the control drums, which are designed to control reactivity letdown with fuel depletion, are assumed to be in the highest reactivity positions.

A.8.1.4 Flooding reactivity effects results

The most limiting shutdown state, with all materials at room temperature, water fully filling the reactor cavity, and control drums in the highest reactivity positions, was modeled using Serpent and shows that the reactor maintains subcriticality with a margin of more than 4000 pcm. Therefore, in the highly unlikely event of a full flood of water into the reactor cavity in the most limiting state, the reactor does not unexpectedly reach criticality.

A.8.2 Flood event family analysis results

Following a flood nearby the site, it is extremely unlikely that water could enter the building and reach the reactor module. The powerhouse is surrounded by a berm, which is unlikely to be penetrated by localized flooding. Further, the basement of the powerhouse is a significantly large area, such that reaching a substantial water height is unlikely. The reactor module is located in the basement and features a reactor module curb, which is an elevated portion of the basement floor. It is unlikely that enough water could accumulate to overcome the added elevation of the reactor module curb to enter the reactor cavity. Nevertheless, water was postulated to enter the reactor cavity to partially and fully mix with the normal fluid (i.e., air).

The purpose of the heat transfer analysis was to assure that heat can be adequately removed from the reactor module, which is a primary method of decay heat removal following a worst-case accident. Since the heat transfer coefficient to water, or an air-water mix, is significantly higher than to air alone, water entering the reactor cavity only increases heat removal from the reactor module. Therefore, in the extremely unlikely event that water enters the reactor cavity, the heat removal from the reactor module is only increased, and the reactor is in a less challenging state.

The purpose of the reactivity effects analysis was to ensure that the reactor cannot unexpectedly increase in reactivity and reach criticality. The analysis was conducted in the most limiting shutdown state. All materials were assumed at room temperature, the reactor cavity was assumed to be fully filled with water, and the control drums were assumed to be in the highest reactivity positions. The analysis shows that the reactor maintains subcriticality with a margin of more than 4000 pcm, and unexpected criticality does not occur. Therefore, the flood family analysis did not result in any site commitments.

A.9 Seismic event family analysis

External hazards included in the seismic event family pose either a challenge to the Aurora powerhouse or a challenge to the reactor module and are as follows:

- Hail
- High winds
- Precipitation, intense
- Sandstorm
- Seismic event
- Snow
- Tornadoes (extreme winds)
- Transportation accident
- Tsunami

In order to fully analyze the effects of the external hazards included in the seismic event family, a large resulting force from a ground acceleration and a full powerhouse collapse are analyzed. Specifically, analysis of the resulting force from the ground acceleration on the reactor module is of interest. Since the reactor module is densely packed of nearly all metal components,[4] a structural analysis of the reactor module is an appropriate indicator of the integrity of the internals. The complement to this analysis is the powerhouse collapse analysis to also analyze the structural integrity of the reactor module, specifically the module equipment housing. Therefore, the relevant features of interest in the seismic analysis are the following:[5]

- Ability for the shutdown rods to insert into the reactor
- Protection to the shutdown rod equipment, provided by the reactor module
- Integrity of the reactor module and internals

[4] Further information of the reactor module components is located in Chapter 2.

[5] Seismic events have traditionally been considered the most bounding events for metal-fueled fast reactors, primarily due to the possibility of large positive reactivity insertions caused by control rod motion relative to the core lattice or reactor coolant sloshing. Reactivity challenges typically associated with seismic events are not of concern for the Aurora. These challenges typically include sloshing coolant, or oscillating control rods. Since the Aurora does not utilize a reactor coolant, there are no reactivity concerns associated with sloshing of the coolant. Additionally, since the Aurora does not operate with control rods, oscillation of rods is not of concern. The shutdown rods used in the Aurora are fully withdrawn a significant distance from the core during normal operation and would not pose a risk in the same manner as control rods which are inserted during normal operation. The other reactivity system is situated in the same structure as the reactor such that any oscillations would affect both the system and the reactor similarly, or would place the reactor in a less reactive configuration.

Ultimately, these three features can be analyzed through an evaluation of the reactor module integrity. If the reactor module does not reach structurally-challenging limits, then the reactor is shut down, and decay heat is passively rejected to the air and surrounding structures. These results would be within the design parameters of the Aurora systems, as described in Chapter 2, and within the assumptions and consequences of the safety analysis, in Chapter 5.

The seismic analysis included as part of this external hazards evaluation is a summary of Chapter 7, "Earthquake criteria."

A.9.1 Seismic event family analysis methodology

For purposes of this deterministic seismic analysis, this section is broken up into two sections: (1) analysis of a large ground acceleration, and (2) analysis of a full powerhouse collapse. The goal of the large ground acceleration analysis is to confirm that the reactor module integrity remains intact, which assures integrity of the internals. The goal of the powerhouse collapse is to analyze the reactor module integrity, specifically those portions that protect the shutdown rod equipment. If the reactor module integrity is upheld after an extreme ground acceleration and a full powerhouse collapse, the safety of the reactor is unchallenged.

A.9.1.3 Large ground acceleration methodology

The purpose of the ground acceleration analysis is to evaluate the structural effects on the reactor module, following a hypothetical extreme ground acceleration, as a result of a large earthquake. This analysis assumes a large earthquake occurs nearby the Aurora that disables heat removal by the secondary system and results in an automatic reactor trip. Therefore, this analysis focuses on showing that the structural limits of the reactor module are unchallenged.

The first step in the large ground acceleration analysis is to define a conservatively large, bounding, ground acceleration that would result following an extreme earthquake. Instead of assessing the largest earthquake at a specific U.S. site, this analysis assessed the largest earthquake experienced in the U.S., including areas outside of the contiguous U.S. The goal of this analysis is to define a set of conservative seismic design parameters, such that the Aurora could be sited in most U.S. locations, without seismic concerns. This approach is based upon a review of seismic ground motions of operating nuclear plants in the contiguous U.S. and evaluation of seismic parameters at other representative U.S. locations, such as Alaska, Hawaii, Puerto Rico, and the Virgin Islands. The peak ground acceleration (PGA) used was 1.75 g and the resulting response spectrum was the basis for the modal analysis of the reactor module; the response spectrum was developed in accordance with RG 1.60, "Design response spectra for seismic design of nuclear power plants," Revision 2, issued July 2014. More information on the determination of this PGA is in Chapter 7.

The following assumptions are made for the ground acceleration portion of this analysis:

- The reactor module is modeled as a single body, which is appropriate because of the similarity of materials present in the reactor module shell and internal components, as well as the rigidity of the internal components within the module.

- The reactor module is modeled with a shell model, which is appropriate because the width of the reactor module walls are less than 1/10 the diameter of the reactor module.

- The density of the model is adjusted to account for the reactor module internals.

- The reactor module is assumed to be rigidly mounted at the support flange, which is approximately at the foundation elevation of the powerhouse, such that no significant amplification of seismic accelerations is assumed.

- The analysis assumes a close coupling of the reactor module and the ground.

- The module is conservatively assumed to have the damping value of 5%, the smallest damping value outlined in RG 1.60.

- Material properties used in the analysis assume a temperature of the module shell of 300 C, which is conservatively high in comparison to the temperatures expected during operation.

Next, a three-dimensional finite element model is analyzed using modal and response spectrum analyses in ANSYS Mechanical [16]. The reactor module is modeled with a refined mesh to adequately capture localized failure modes.

It is necessary to determine the dominant modes of the reactor module so the corresponding spectral acceleration could be applied to the ANSYS model. Due to the simplicity of the model, more than 67% of the participating modal mass is within the first two mode shapes, of approximately 18.8 Hz, and correspond to cantilever, or flexural shapes applicable to the reactor module; higher mode shapes are primarily circumferential in nature.

The two parameters of interest for the reactor module for the large ground acceleration analysis are (1) the peak equivalent (von Mises) stress, and (2) the horizontal displacement of the reactor module. The stresses in the reactor module are analyzed to assure that the reactor module did not reach material limits that could result in failure and to evaluate any concerns related to deflection of the module sufficient to distort internals so that the shutdown rods could not drop. The results are summarized as follows:

- The maximum stress in the model is 40.6 MPa, which is substantially lower than the conservative yield strength of 129 MPa.

- The maximum horizontal displacement, taken at the lowest tip of the reactor module, which experiences the greatest displacement, is 2.1 mm. Since the shutdown rod tolerance is approximately 7.9mm, a 2.1 mm maximum displacement of the entire module would not compromise the ability for the shutdown rods to drop.

A.9.1.4 Large ground acceleration results

Following an extreme earthquake, the secondary system is assumed to be lost. The loss of the secondary system results in an automatic reactor trip signal being sent to the shutdown rods via the reactor trip system. Because of the large ground acceleration that results from an extreme earthquake, the reactor module is analyzed for mechanical loading and maximum displacement, to confirm no material failure is experienced, and to confirm the shutdown rods are able to insert to shutdown the reactor.

This portion of the seismic analysis concludes that the reactor module experiences mechanical loads within its material limits, confirming that the integrity of the module is maintained, and also that the ability of the shutdown rod to insert will not be compromised due to deflection.

A.9.1.1 Aurora powerhouse collapse methodology

The purpose of the Aurora powerhouse collapse analysis is to evaluate the structural effects on the module equipment housing following a hypothetical complete collapse of the powerhouse. This analysis assumes a full powerhouse collapse in order to bound all external hazards that could pose a challenge to the powerhouse. This analysis assumes a full powerhouse collapse disables the secondary system since the secondary system is located on the first floor of the powerhouse, directly underneath the powerhouse roof. The result of the secondary system being disabled is an automatic reactor trip, if not already triggered by another secondary failure. Therefore, this analysis focused on the ability of the reactor to be shut down by the shutdown rods, following an automatic reactor trip signal. To maintain shutdown functionality, the integrity of the reactor module must be upheld. Specifically, the subcomponent of the reactor module analyzed is the module equipment housing, which functions to protect equipment such as the shutdown rods. The goal of this analysis is to demonstrate the integrity of the module equipment housing through an impact analysis of the heavy powerhouse components.

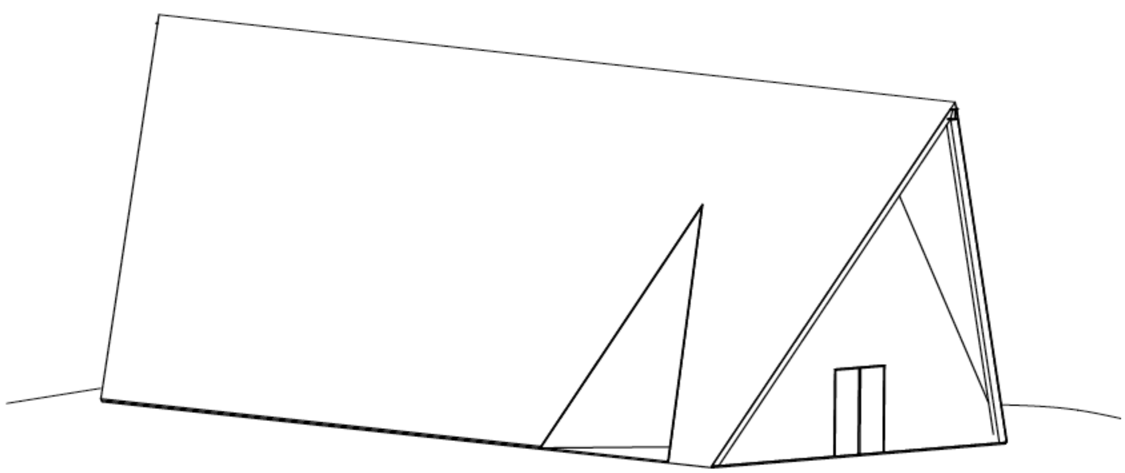

Figure 1-24: Aurora powerhouse

The Aurora powerhouse is an A-frame, as shown in Figure 7-3, which has a relatively small square footage of less than 5,000 sq. ft. The module equipment housing is a reactor module component and is located in the basement of the powerhouse as shown in Figure 7-4. One of the functions of the module equipment housing is to protect the shutdown rod equipment, which was the function of interest to this analysis. The module equipment housing, which is made of stainless-steel 304, is assumed to have an ultimate tensile strength of 517 MPa [17]. The considerations analyzed the impacts from a collapsed roof, crane, and floor to assess the deformation and penetration damage on the module equipment housing following a powerhouse collapse.

{

}

Figure 1-25: Module equipment housing location

The Aurora powerhouse includes several heavy components that could cause damage to the module equipment housing upon collapse. The falling objects considered were the heavy objects in the powerhouse that had a line of sight to the module equipment housing. These included the roof, the crane, and the floor, with the following general assumptions:

- The roof falling objects included several A-frame roof beams, which are arranged in a triangle formation and are supported at the base and top of the A-frame, and of standard steel roof construction.
- The crane falling object included a single girder crane type, which is underhung, and supported by a standard I-beam girder.

- The floor falling object included the first floor of the powerhouse, which is located directly above the module equipment housing, and was conservatively assumed to be thick concrete.

These falling objects can be seen in Figure 7-5. {

}

Figure 1-26: Schematic of falling objects considered

From these falling objects, the roof (i.e., roof beams) are determined to cause the maximum impact damage to the module equipment housing and are used in this analysis. The module equipment housing thickness required to prevent penetration by the roof beams is calculated based on idealized hand calculations from BC-TOP-9A, Revision 2, "Topical Report - Design of structures for missile impact," issued September 1974. The roof beam is assumed to land on edge for minimum effective missile diameter and maximum penetration force. Even following such an extreme collapse scenario, the thickness of the module equipment housing is found to be great enough such that no penetration was experienced from the falling roof beam.

A.9.1.2 Aurora powerhouse collapse results

Following a full Aurora powerhouse collapse, the secondary system is lost. The loss of the secondary system results in an automatic reactor trip signal being sent to the shutdown rods via the reactor trip logic. Subsequently, the shutdown rods insert and shut down the reactor. Because of several heavy powerhouse components, the goal of the powerhouse collapse analysis is to confirm that the shutdown rods would be able to insert. This analysis analyzed whether the integrity of the module equipment housing is upheld through an impact and penetration analysis of the falling of heavy powerhouse components.

The powerhouse collapse analysis found that the thickness of the module equipment housing is sufficient to withstand impact from falling objects due to a full powerhouse collapse. Therefore, the shutdown rods are able to insert into the reactor following a full powerhouse collapse.

A.9.1.4 Seismic event family analysis results

Following an extreme earthquake, it is unlikely that the Aurora powerhouse would suffer complete collapse and that the reactor module would see such extreme structural loads. Nevertheless, for the purposes of the seismic event family, the powerhouse is assumed to completely collapse, and the reactor module is assumed to be experience a large ground accelerations.

The purpose of the large ground acceleration analysis is to assure that the integrity of the reactor module is maintained, following an extreme earthquake. The first step of this analysis was to determine a ground acceleration that would bound the entire U.S., including islands and Alaska. The ground acceleration applied in this analysis was 1.75 g PGA, corresponding to a 0.50 g PGA ASCE 7 value. Next, this ground acceleration is applied to the reactor module and reactor module material properties are analyzed through ANSYS. The reactor module material properties of interest are not found to be close to their limits, so a large ground acceleration is not found to challenge the integrity of the reactor module.

The purpose of the powerhouse collapse analysis was to assure that the shutdown rods are able to fully insert into the reactor module, in order to shut down the reactor. To conduct this analysis, an impact evaluation is performed on the reactor module. Even following a collapse of the heaviest object in the most limiting orientation to the reactor module, the reactor module is found to be robust and is not penetrated. Therefore, a full powerhouse collapse is not found to impede insertion of the shutdown rods.

As a result of these analyses, the seismic event family resulted in one site commitment, discussed in Section A.10 Site commitments." This site commitment ensures that any proposed site will be verified to be within the ground acceleration analyzed in the seismic analysis.

A.10 Site commitments

The last step of the external hazards methodology is to define commitments to address limitations from the bounding deterministic analyses or to mitigate site-dependent hazards.

A.9.2.1 Event family site commitment

Sections A.6, A.7, A.8, and A.9 describe bounding analyses performed for the explosion, fire, flood, and seismic event families, respectively. The fire and flood event families assumed a bounding event and did not result in a state that challenged the safety of the reactor; therefore, no site commitments are necessary for the fire and flood event families. From the four event families, the explosion and seismic event families require site commitments, in order to ensure that any proposed site (1) does not challenge the blast capacity of the reactor module and (2) does not have a recorded PGA higher than what was analyzed for the Aurora.

A.9.2.1.2 Explosion event family

A man-made hazard nearby the site could result in an explosion, which could pose damage to key Aurora components. In order to conservatively bound these man-made hazards, a blast analysis was performed as part of the explosion event family in the external hazards methodology. The blast analysis assumed a large explosion nearby the facility and calculated the maximum pressure several key Aurora reactor components could withstand, referred to as the "blast capacity." { }A site commitment is made as a result of the deterministic analysis conducted for the explosion event family. This site commitment ensures that the blast capacity of the reactor module is not exceeded and is named the "man-made hazards commitment."

Man-made hazards commitment: The area surrounding the proposed site will be evaluated for explosive hazards. Blast hazards that are identified will be evaluated to determine if their resulting pressure exceeds the blast capacity.

A.9.2.1.2 Seismic event family

An extreme earthquake could have two effects: damage to the Aurora powerhouse and a resulting force from the ground acceleration. The seismic event family assumed a full powerhouse collapse following an extreme earthquake, so no site commitment is needed for that portion of the deterministic analysis. Since it is impractical to bound all large ground accelerations, an extreme ground acceleration was assumed as part of the deterministic analysis for the seismic event family. In order to ensure that proposed sites are within this assumed ground acceleration value, a site commitment is made and is named the "seismic event commitment."

Seismic event commitment: The largest recorded PGA for the proposed site will be determined under ASCE 7. If the PGA of the proposed site exceeds 0.50 g, additional analyses must be performed.

A.9.2.2 Site-dependent hazards commitments

The site-dependent external hazards are the remainder of the external hazards from the list of all potential external hazards in Table 1-6. These are the external hazards that do apply to the Aurora, are not associated with a challenge to the heat sink, and were not found to have a common challenge, as summarized in

Table 1-7. Site-dependent hazards can be mitigated by appropriate site selection and are as follows:

- Avalanche
- Coastal erosion
- Forest fire
- Grass fire
- Landslide
- Sinkholes

A.9.2.2.1 Avalanche site commitment

The external hazards evaluation did not analyze an avalanche hazard.

Avalanches are generally considered a mass of snow, rock, ice, soil, and other materials moving rapidly down a mountainside. The steepness of the mountainside generally needs to have a 25-45 degree slope for an avalanche to occur and is illustrated in Figure 1-27.

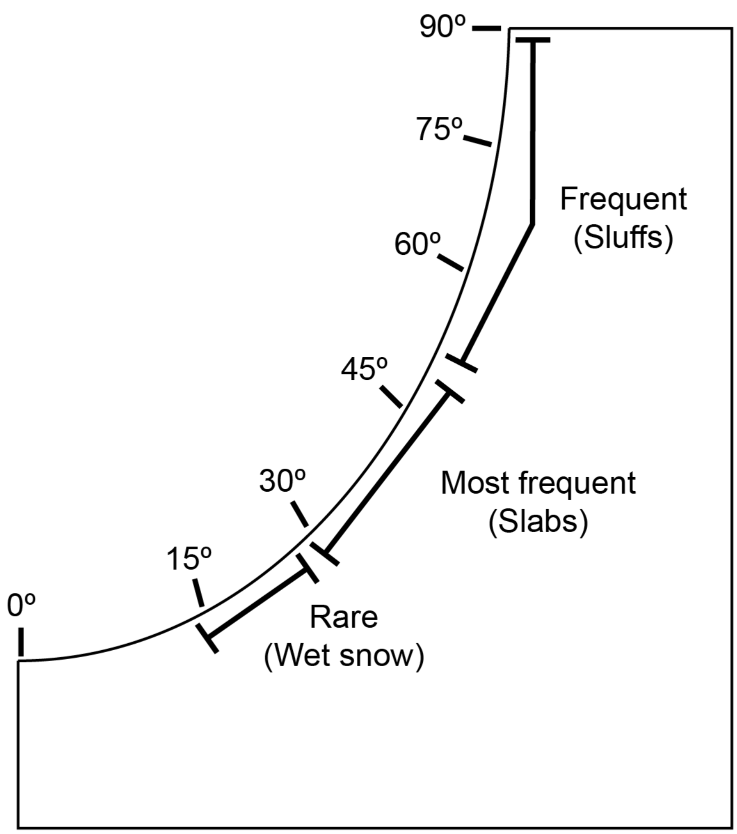

Figure 1-27: Relationship between slope degree and avalanche frequency

Terrain that lies in the fall line, or along a downhill line of trajectory, could also be considered as capable of being hit by an avalanche. Typically, the critical incline of the avalanche start zone is measured as the steepest part of a slope with a down slope length of 60 ft or more [18]. An avalanche occurring either above or below the proposed site could damage the site, depending on its characteristics.

If the proposed site has both of the following characteristics:

- Is within 1 mi of a slope greater than 25 degrees, judged by 100 ft contour lines
- Has data indicating avalanches have occurred in the region or geomorphologic indicators of avalanches

Then, the proposed site is considered to be in an avalanche-prone environment and requires further avalanche investigations.

Avalanche commitment: Information will be provided on whether the proposed site is in an avalanche-prone environment. If the proposed site is in an avalanche-prone environment, further investigations will be performed to evaluate the potential avalanche concerns.

A.9.2.2.2 Coastal erosion site commitment

The external hazards evaluation did not analyze a coastal erosion hazard.

The regions in the U.S. with highest rate of coastline retreat can lose over 50 ft per year of coast due to erosion [19]. The concern of coastal erosion impacting the proposed site within the 20 year Aurora lifetime can be eliminated by not siting within one-half of a mile of a coastline, even assuming the greatest historically recoded coastline retreat.

Coastal erosion commitment: Further investigations will be performed for possible coastal erosion concerns if the proposed site is located within one-half of a mile of a coastline.

A.9.2.2.3 External fire site commitment

The external hazards evaluation did not evaluate external fires that could occur nearby the Aurora facility, although it did evaluate hazards that resulted in an internal fire.

External fire hazards considered are the following site-dependent external hazards:

- Forest fires
- Grass fires

In order to conservatively bound these external fire hazards, the area around the Aurora will be cleared from vegetation. Oklo Power will follow the guidance in NFPA 1144. Specifically, vegetation will be modified to mitigate hazardous conditions within 30 ft of the Aurora powerhouse foundation. Additionally, all slash from vegetation modification and construction debris will be treated or removed prior to or immediately upon completion of construction. By removing all vegetation around the Aurora foundation, the potential for damage due to external fires is ameliorated.

External fire commitment: The area directly surrounding the powerhouse will be cleared during site preparation in accordance with NFPA 1144.

A.9.2.2.4 Landslide site commitment

The external hazards evaluation did not analyze a landslide hazard.

A landslide is broadly defined as the downslope movement of a mass of regolith or bedrock under the influence of gravity. For this discussion, the term landslide incorporates earthflows, rockslides, and slump blocks. Earthflows and rockslides can be fast moving events, whereas a slump block tends to be slower moving. Landslides could cause damage to the proposed site if it is located at the top of the land mass or near the bottom.

The USGS and local geologic organizations inventory landslide data and suggest areas of increased probability of experiencing a historic landslide. Contributing site characteristics that indicate the potential of a landslide include the slope, the soil strength, and ground water. The likelihood of a landslide occurring is also impacted by the seismic activity, fire activity, precipitation, flood activity, and volcanic activity of a region. Generally, landslides are more likely to occur on slopes greater than 15 degrees, though they can occur on smaller slopes [20]. It is appropriate to investigate for historic landslides within a mile of the proposed site.

If the proposed site has either of the following characteristics:

- Is within 2 mi of a slope greater than 15 degrees, judged by 100 ft contour lines
- Has data indicating a landslide has occurred within 2 miles of the site

Then, the proposed site is considered to be in landslide-prone environment and requires further landslide investigations.

Landslide commitment: Additional information will be provided on whether the proposed site is in a landslide-prone environment. If the proposed site is in a landslide-prone environment, further investigations are necessary to evaluate the potential landslide concerns.

A.9.2.2.5 Sinkhole site commitment

The external hazards evaluation did not analyze a sinkhole hazard.

Sinkholes are a type of underground void that form in karst terrains, i.e., terrain where water can drain below the ground and dissolve water-soluble evaporate rock such as salt, gypsum, or carbonate rocks. The largest recent land collapse due to a sinkhole in the U.S. was about 325 ft long [21].

Sinkhole commitment: Additional information will be provided on the proposed site, to show whether it is within one-fourth of a mile of karst terrain. If the proposed site is found to be in an area that has karst terrain, further investigations will be performed to determine if there are underlying voids within a radius of 300 feet outside of the site boundary that could result in a significant ground collapse.

A.9.3 Conclusion

One the design objectives of the Aurora is the ability to be sited in the majority of the U.S. In order to develop a set of site suitability parameters that are bounding for most U.S. sites, a broad external hazards evaluation was conducted. Although external hazards have been traditionally evaluated on a site-specific basis through PRA, the simple design of the Aurora allows for the evaluation of most extreme external hazards on a deterministic basis.

A total of 36 external hazards were evaluated as part of this methodology. Most external hazards were grouped together, in event families, due to a common challenge to the Aurora. The event families were further evaluated under extreme deterministic analyses to determine if the safety of the Aurora was adversely affected. The Aurora was found to be resilient for most of the event families, requiring only two site commitments to ensure the safety of the Aurora. For those external hazards that were not grouped into event families (i.e., site-dependent hazards), specific site commitments were made to ensure that the safety of the Aurora is maintained through appropriate site selection. The collection of these site commitments comprise the set of parameters that any proposed site must be evaluated against to assess the safety suitability of the site. This set of parameters is called the generic site envelope.

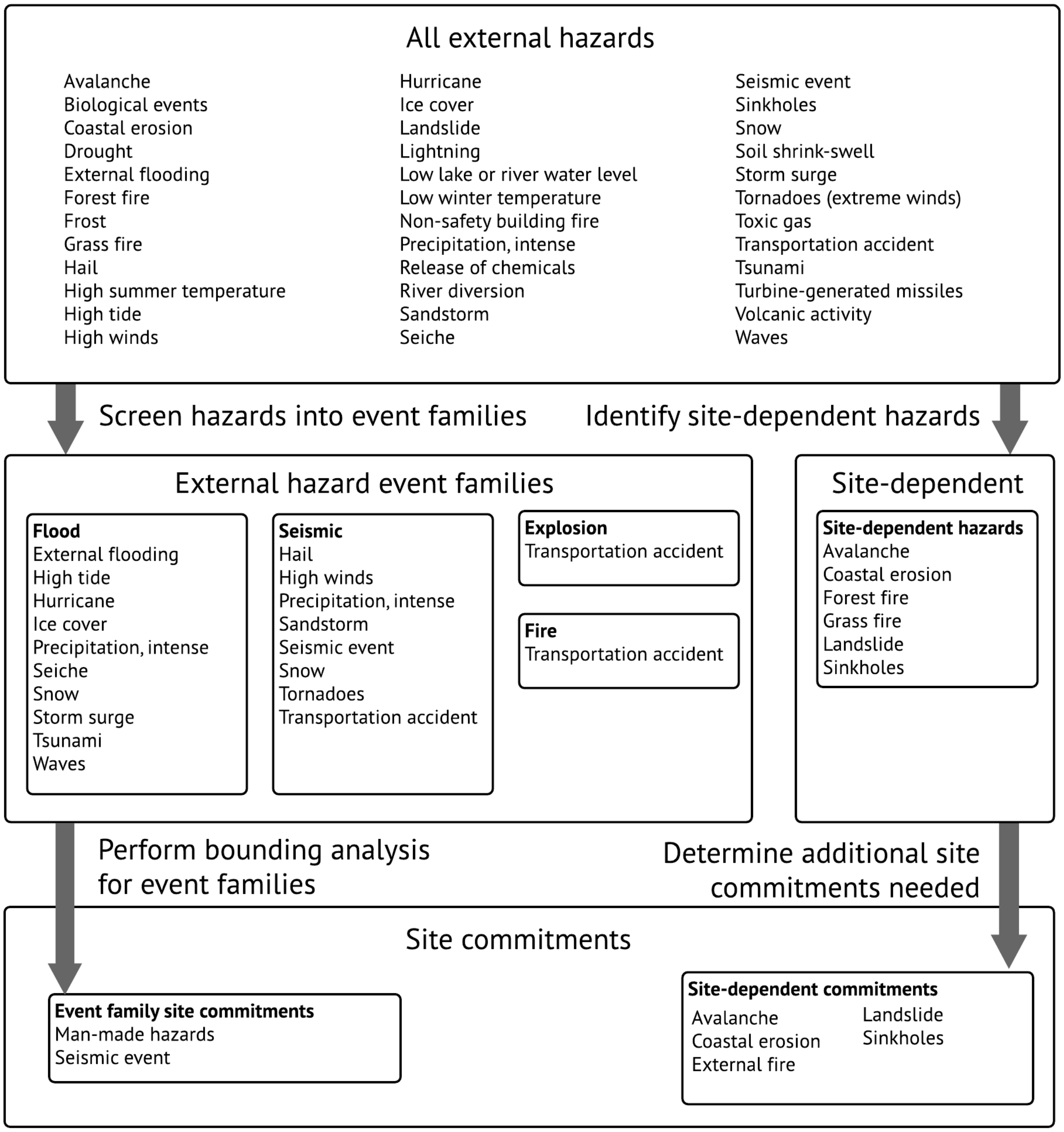

Figure 1-28: External hazard methodology summary

Appendix B: Generic site envelope

The external hazards evaluation resulted in several site commitments. Site commitments are specific actions to be performed during site selection or site preparation. These site commitments are grouped under site bases, which are siting principles that assure that the safety of the Aurora is not adversely affected by the natural features of the proposed site. The collection of site bases comprise the generic site envelope. Ultimately, the generic site envelope is the set of parameters the site must meet to ensure the site does not adversely impact the safety of the Aurora. This generic site envelope development process is shown in Figure 1-29.

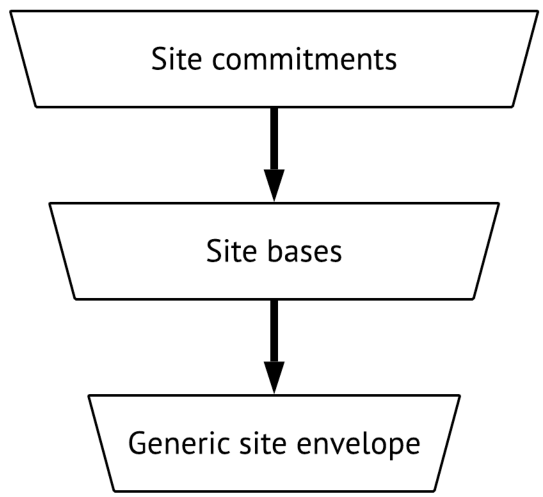

Figure 1-29: Generic site envelope development process

The generic site envelope consists the coastal site, external fire, geologic, man-made hazards, and seismic bases and consider the following siting principles:

- **Coastal site basis:** The proposed site will not damage the Aurora reactor by coastal hazards.
- **External fire basis:** The proposed site will not damage the Aurora facility due to an external fire.
- **Geologic basis:** The proposed site will not damage the Aurora facility due to soil or topographic characteristics.
- **Man-made hazards basis:** The proposed site will not damage the Aurora reactor by an explosion.
- **Seismic basis:** The proposed site will not damage the Aurora reactor by a large ground acceleration.

The bases and the respective site commitments are depicted in Figure 1-30.

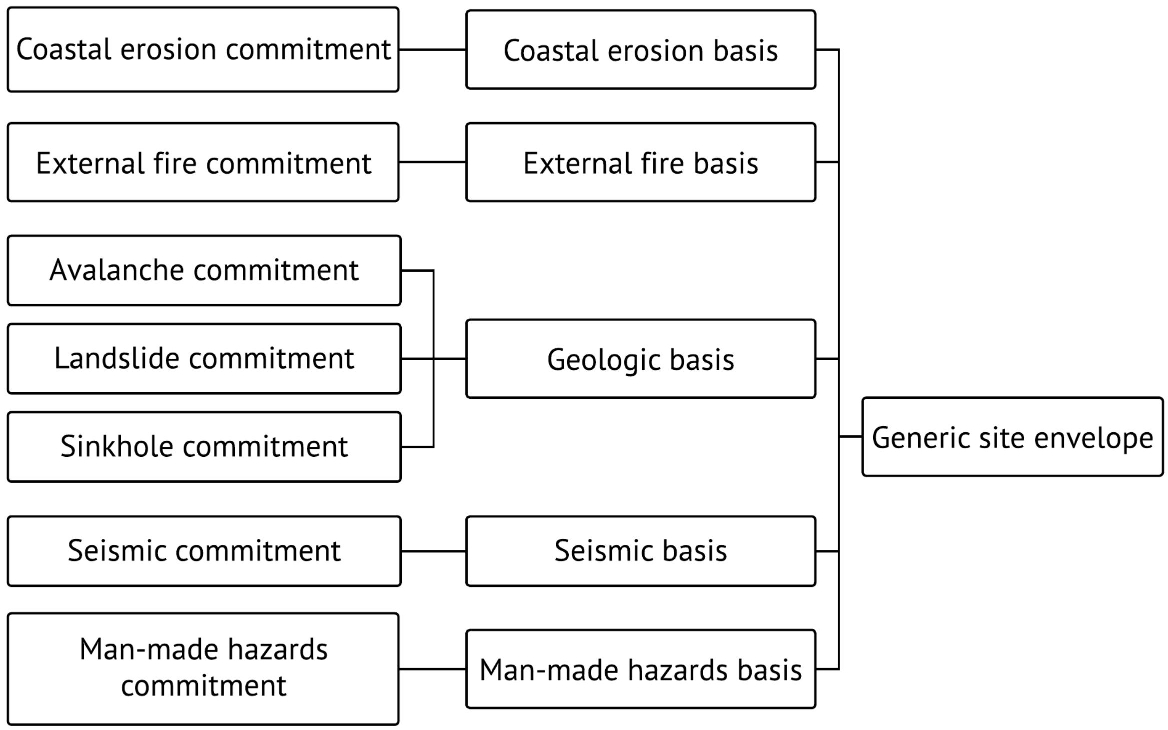

Figure 1-30: Generic site envelope hierarchy

As sites are considered, the first step is to make sure the site conditions fit within the generic site envelope, shown in Table 1-9. The proposed site is evaluated for each site commitment parameter. For proposed sites that have parameters which exceed the allowable values, further investigations must be conducted to evaluate the effect on the safety of the Aurora.

Table 1-9: Generic site envelope

Site commitment parameter	Value
Coastal site basis	
Distance to a coast (mi)	0.5
External fire basis	
Brush clearing (ft)	30
Geologic basis	
Avalanche site commitment	
Distance to slope greater than 25 degrees (mi)	1
Historic avalanche data or geomorphologic indicators of avalanches (Y/N)	N
Landslide site commitment	
Distance to slope greater than 15 degrees (mi)	0.25
Historic landslide data (Y/N)	N
Sinkhole site commitment	
Distance to karst terrain (mi)	0.25
Man-made hazards basis	
Blast hazards investigation (required for all sites)	-

II.02 Design and analysis of structures, systems and components

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2 DESCRIPTION AND ANALYSIS OF STRUCTURES, SYSTEMS, AND COMPONENTS

2.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(2) requires, in part:

> A description and analysis of the structures, systems, and components 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 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 [7] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated;

The purpose of this chapter is to provide an overview of the structures, systems, and components (SSCs) that are part of the Aurora design. The level of detail provided for each SSC is scoped to correspond directly to the safety functions required of the SSC, with additional detail provided only when required for basic understanding of the system's function. This chapter focuses on the expected functions of the Aurora SSCs and addresses the majority of the requirements in 10 CFR 52.79(a)(2). Functions of SSCs during abnormal conditions is covered by Chapter 5.1, “Transient analysis.”

2.1 Introduction

2.1.1 Design bases, design commitments, and programmatic controls

Design bases are the characteristics of a system that ensure the safe operation of the reactor. Most major systems in the reactor have at least one design basis, but some systems that are not relied on for safe operation do not have any design bases. Each design basis has one or more design commitments, which are the specific commitments made to ensure that the design basis is met. Each design commitment has one or more programmatic controls that are used to verify that the commitment is met.

The central importance of the design bases drives the structure of this chapter. Each major system of the Aurora design is described in its own section, and each section is organized around the design bases of the system. The level of detail of the section is scoped to provide the required information to evaluate the sufficiency of the design bases in ensuring safe operation.

Programmatic controls are used to verify that design commitments are met, and therefore that design bases are satisfied. These controls include preoperational tests (POTs), inspections, tests, and analysis acceptance criteria (ITAAC), startup tests (SUTs), and technical specifications (TS).

The assumptions and inputs modeled in the safety analysis in Chapter 5.1 are chosen to ensure that the transient analysis model reflects the characteristics described in the design bases and resultant design commitments. The programmatic controls function not only to verify that the design commitments are met (i.e., that the as-built system is as described in this chapter), but to provide assurance that the assumptions in the safety analysis are valid (i.e., that the modeled system is representative of the as-built system).

Some additional information about each system is provided for the purpose of improving overall understanding of the system. In particular, performance bases are provided for each system as a means of describing functions of the system that are not relied on for safe operation of the reactor. Because the performance bases are not relied on for safety, they do not require design commitments and programmatic controls. As a result, analyses related to the performance bases are generally not included in this application. Systems that have no functions that are relied on for safe operation of the reactor only have performance bases, and their sections have a correspondingly limited level of detail.

2.1.2 Design basis summaries

Gray summary boxes are used throughout this chapter to summarize each design basis at the end of the section describing the applicable system. These boxes contain the design basis, a summary of the evaluation that explains how the design basis is met, and a listing of the design commitments and programmatic controls that ensure the design basis is met.

The following abbreviations are used in the summaries:

- Design basis (DB)
- Design commitment (DC)
- Preoperational test (POT) (see Chapter 14)
- Startup test (SUT) (see Chapter 14)
- Inspections, tests, and analysis acceptance criteria (ITAAC) (see Part VI)
- Technical specification (TS) (see Part IV)

For example: a design basis (DB) for the shutdown rod system (SRS), the resulting design commitment (DC), and the required programmatic controls, would be listed as follows in the summary box:

DB.SRS.01 The shutdown rod system provides sufficient negative reactivity to achieve cold shutdown with insertion of one rod.

DC.SRS.01.A The worth of each shutdown rod will be greater than 1400 pcm, where 1400 pcm is greater than the total of: the reactivity worth associated with the temperature decrease from hot full power conditions to cold zero power conditions, and an additional margin of 500 pcm.

SUT.SRS.01.A1 and A2 (see Chapter 14)

2.1.3 Design parameters

For the purpose of understanding the function and layout of systems, this chapter provides schematics drawn to scale, and nominal dimensions in tabular form. In contrast to large reactors, the Aurora has very few dimensions that affect the design bases and, ultimately, the safety of the plant. The dimensions that are fundamental in the description and analysis of SSCs and their design bases are referred to as “key dimensions.” The few key dimensions are the result of intentional design decisions and are an important aspect to the philosophy of the Aurora design.

2.1.3.1 Key dimensions

The fundamental building block of the active core of the Aurora is the reactor cell, described in this chapter and shown schematically in Figure 2-3. The reactor cell forms the simplest self-contained unit of core reactivity and heat flow. As described by the principal design criteria of the Aurora, core reactivity and thermal performance are the two driving phenomena to ensure robust and safe operation. The means by which the reactor cell dimensions impact the neutronic performance and the thermal characteristics of the Aurora are further described in this chapter and Chapter 5.

The reactor cell dimensions presented here form an important component of the reactor design, as the reactivity of the system is most sensitive to these parameters. This is because, in a fast spectrum reactor like the Aurora, no moderating material is present in the core, and as such, neutrons are not expected to slow down significantly after being generated in fission. As a result, the probability of a neutron causing fission is most dependent on the probability that a neutron encounters fuel before being absorbed in another material or escaping from the core. The probability of a neutron encountering fuel is first-order dependent on the dimensions of the fuel relative to the immediately surrounding structural materials, namely the heat pipe and the reactor cell can, making these dimensions of primary importance for evaluating the reactivity of the system.

The dimensions of the reactor cell are also of primary importance for evaluating the heat transfer characteristics of the core at steady-state. The heat pipe in each reactor cell is the primary heat transfer pathway for removing heat generated by fission in the fuel and transferring it to the power conversion system via the heat exchanger system. This means that the heat pipe, the reactor cell inner cylinder wall, and the fuel form the primary resistances in the thermal conduction network through which the heat flows from the fuel to the heat exchanger system. Thus, the radial dimensions of these components, together with their thermal conductivities, determine to first order the temperature gradient that exists between the fuel and the heat pipe. Likewise, in transient thermal analyses, these dimensions, together with the thermal conductivity and heat capacity of the materials, determine the change in the temperature distribution over time.

The dimensions with first order importance to the core reactivity and thermal characteristics of the system are the key dimensions of the Aurora. The key dimensions are summarized in

Table 2-1 and depicted in Figure 2-1. Other dimensions, such as the thicknesses of the surrounding reflector and structural components outside of the active core, do have some effect on the core reactivity and thermal characteristics. However, since these components are at a distance from the fuel and the primary heat transfer pathway of the heat pipes, the neutronic and thermal response of the system is only second- or third-order dependent on these dimensions, making their exact values less relevant than those of the reactor cells. As such, only the reactor cell dimensions presented here are considered key dimensions of the Aurora.

Table 2-1: Key dimensions for the Aurora

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Figure 2-1: Schematic showing location of key dimensions

2.1.4 Chapter structure

The goal of this chapter is to describe the Aurora SSCs and to provide an analytical basis for the design bases selected. This chapter is organized by the Aurora major systems, which are the following:

- Reactor system (Section 2.2)
- Control drum system (Section 2.3)
- Shutdown rod system (Section 2.4)
- Reactor enclosure system (Section 2.5)
- Heat exchanger system (Section 2.6)
- Instrumentation and control system (Section 2.7)
- Power conversion system (Section 2.8)
- Electric power system (Section 2.9)
- Building and auxiliary systems (Section 2.10)

Each major system is composed of subsystems, as needed. Each of the sections for these major systems or subsystems includes a description and the bases of the system. The bases of the system are further divided into design bases and performance bases. Design bases, as discussed in Section 2.1.1, are the characteristics of a system that ensure the safe operation of the reactor. Performance bases are intended functions of a system but do not relate to the safe operation of the reactor.

2.2 Reactor system

2.2.1 Summary description

The Aurora reactor is a compact fast reactor, using metal uranium zirconium (UZr) fuel to generate heat, and heat pipes (rather than flowing coolant) to transport the heat from the reactor core. It has a rated thermal output of 4 megawatts (MW), and a core power density of 3.91 W/cm^3. The fast spectrum, small size, and lack of flowing coolant in the core enable a simple design with few moving parts.

The reactor system is the fundamental system of the Aurora. It is designed to generate heat, which is converted to the appropriate end product by the secondary system. This section describes the design of the mechanical and nuclear components of the reactor system, which is composed of the following subsystems:

- Reactor core system (Section 2.2.2)
- Reflector system (Section 2.2.3)
- Shielding system (Section 2.2.4)

The reactor core subsystem is responsible for generating heat and transporting it to the heat exchanger system (see Section 2.6). The reflector subsystem surrounds the reactor core system and functions to improve fuel utilization by reflecting neutrons back into the core. The shielding subsystem functions to limit onsite and offsite radiation exposure and keep radiation fluence to equipment and enclosures below a level at which embrittlement or other significant radiation damage may occur.

The thermal design of the reactor system provides cooling for the fuel and the core components during steady-state, full-power conditions via heat pipes. Steady state analyses at full power are shown in Section 2.2.2.5.1. As the reactor system does not include a flowing coolant traveling through the core, many of the typical concerns of thermal and hydraulic behavior need not be considered (e.g., critical heat flux, flow velocities, coolant and moderator voids).

The low power level of the core leads to low decay heat production. After shutdown, adequate cooling of the fuel and core components is achieved through heat conduction from the fuel to the surrounding systems, and ultimately transferred to the environment. The very low power density, and the high thermal conductivity of the materials in the reactor system, allow for effective radial and axial conduction of heat out of the core and ensure that decay heat does not present a challenge to safety. Analysis of important transients is presented in Chapter 5.1.

2.2.2 Reactor core system

2.2.2.1 *Introduction to the reactor core system*

The reactor core system consists of a matrix of hexagonal "reactor cells." The reactor cells are integrated structural, nuclear, and thermal units. Collectively they function to achieve criticality, generate heat, and transport the heat to the heat exchanger system (see Section 2.6). The reactor cells operate entirely passively.

2.2.2.2 *Bases of the reactor core system*

2.2.2.2.1 Design bases of the reactor core system

The design bases (DB) for the reactor core system, a subsystem of the reactor system (RXS), are as follows:

DB.RXS.01 The reactor core system uses metal fuel with well characterized properties.

DB.RXS.02 The reactor core system is operated at steady state thermal power levels that prevent damage to the system during transients.

DB.RXS.03 The reactor core system has inherently negative reactivity feedback.

DB.RXS.04 The reactor core system provides a pathway to conduct heat from the fuel to the surrounding systems and ultimately to reject it to the environment.

2.2.2.2.2 Performance bases of the reactor core system

The reactor core system is also designed to meet both of the following performance bases:

- The reactor core system generates heat through fission in the nuclear fuel and transfers it to the heat exchanger system.
- The reactor core system is robustly designed, such that it may reliably meet the energy generation needs of its deployment.

2.2.2.3 *Description of the reactor core system*

2.2.2.3.1 Reactor cells

The reactor core system consists entirely of reactor cells, which are integrated structural, nuclear, and thermal units. Each reactor cell is composed of the following components:

- Can
- Lower axial reflector
- Metal fuel
- Upper axial reflector,
- Gas plenum
- Axial shielding

- Sodium bond
- Heat pipe

The stainless steel can encloses the fuel, upper and lower axial reflectors, a gas plenum, and axial shielding. The fuel, reflectors, and shielding are annular, and are fully enclosed by the hexagonal outer can wall and the cylindrical inner can wall. Each reactor cell also contains a heat pipe, which is inserted into the cylindrical "socket" formed by the inner can wall, and extends from the base of the can, through the annular components, and into the heat exchanger. Nominal reactor cell dimensions are summarized in Table 2-2, and a reactor cell and its components are shown in Figure 2-2.

Table 2-2: Nominal reactor cell dimensions

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The reactor cell can provides structural support for the reactor core and is one of the passive physical barriers described in the safety analysis in Chapter 5.1. The lower and upper axial reflectors function to reflect neutrons back to the fuel region of the reactor cell. The metal fuel generates heat for the reactor core system. The gas plenum exists in part for packaging reasons but also provides volume to accommodate fission gases released from the fuel during operation[6]. The axial shielding reduces fluence to the heat exchanger system (see Section 2.6). Finally, the heat pipe functions to transport the heat generated in the fuel out of cell and into the heat exchanger. The cell can contains a small amount of sodium, referred as the bond, that occupies the gaps between the fuel, reflector blocks, and cell can to improve heat transfer.

A cross-sectional view of a reactor cell is shown in Figure 2-3. Sections A, B,C and D provide cross-sectional schematics of different portions of the reactor cell. Section A shows the heat pipe region. Section B shows the axial shielding region. Section C shows the fuel region, including the annular fuel element and the sodium bond. Section D shows the lower reflector region (with the same cross section as the upper reflector region), including the can, reflector, and sodium bond.

[6] Since the Aurora core operates at a very low burnup, little to no fission gas is expected to be released from the fuel. The gas plenum is very conservatively oversized given this expectation.

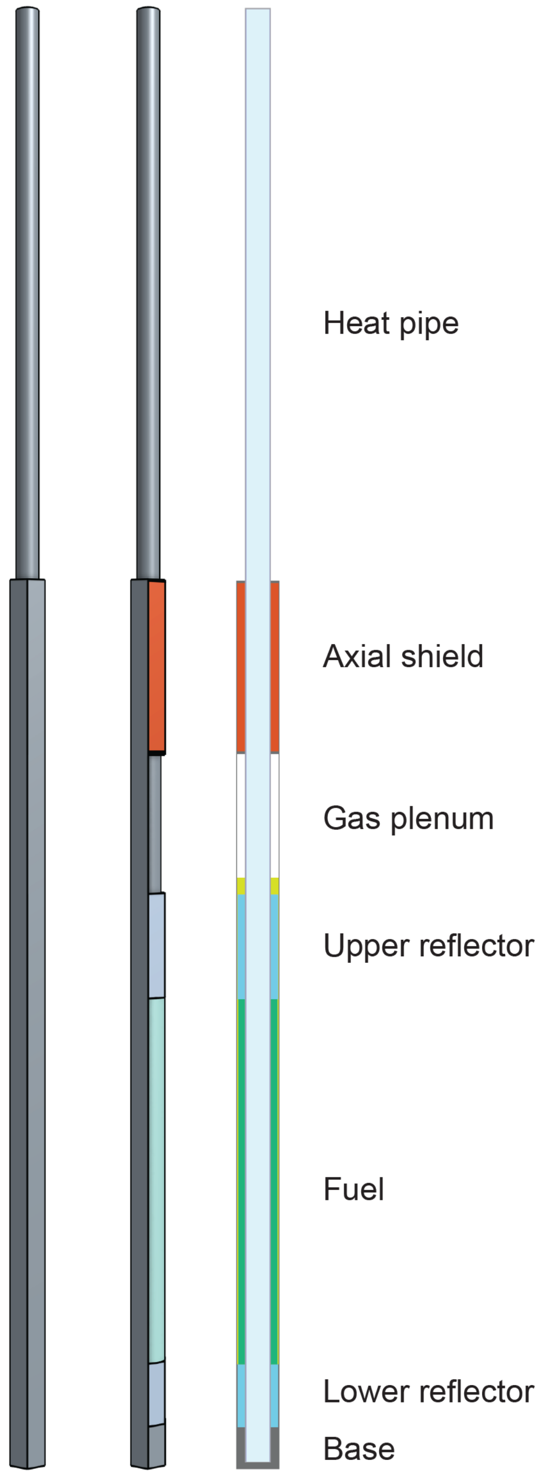

Figure 2-2: Reactor cell isometric and schematic views

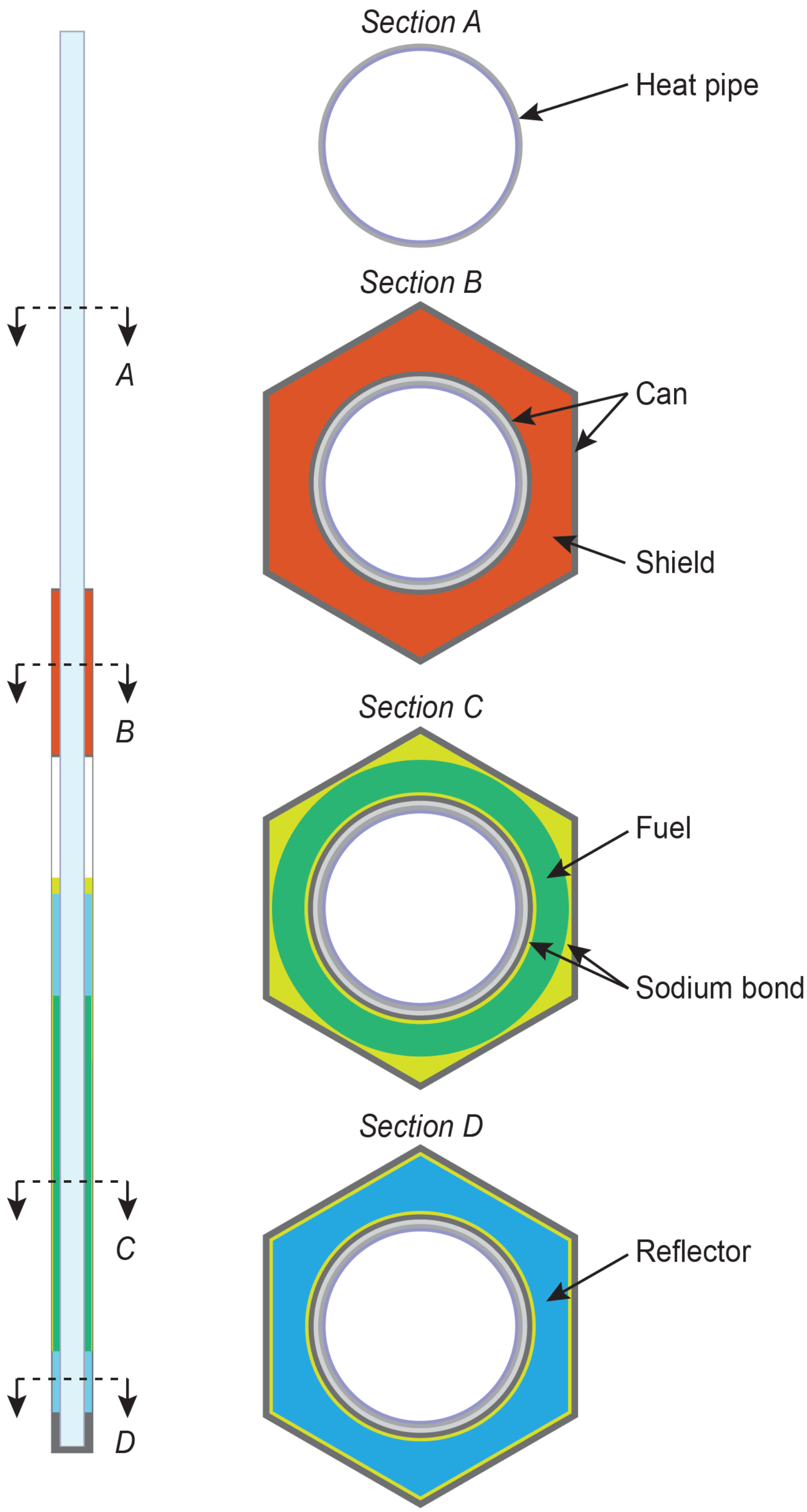

Figure 2-3: Cross-sectional views of a reactor cell

2.2.2.3.2 Heat pipes

The heat pipe in the center of each reactor cell is a sealed tube with a small amount of potassium working fluid. The heat pipes operate passively, requiring no pumps or external piping. They also operate at sub-atmospheric pressure. Because each heat pipe is a sealed, independent heat transport device they offer redundant and reliable cooling, increasing defense-in-depth.

Heat pipes are nearly isothermal, with an equivalent thermal conductivity orders of magnitude greater than high-conductivity metals like copper; as a result, they are often referred to as thermal superconductors. More description of equivalent thermal conductivity is provided in 5.6.2.12.2.

Heat pipes can operate at a wide range of temperatures, and the operational temperature range depends on heat pipe characteristics, including size, and materials. The maximum power throughput of a heat pipe is dependent on its operating temperature. When operated within a specific operational temperature range, heat pipe performance increases with temperature, automatically maintaining proper power-flow ratios in the event of transients. This inherent maintenance of power-flow ratios is beneficial for behavior in transients including failure of neighboring heat pipes. The heat pipes in the Aurora are designed to operate within this range.

The nominal dimensions of heat pipes in the Aurora design are shown in Table 2-3 and the corresponding regions of the heat pipe are shown in Figure 2-4. The base of each heat pipe is located below the bottom of the active core (i.e., the fuel).

Table 2-3: Nominal heat pipe dimensions

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Figure 2-4: Heat pipe regions within the reactor cell

2.2.2.3.3 Core layout

The reactor core consists of { } reactor cells, arranged in a hexagonal lattice on the base plate. The base plate is part of the reactor enclosure system and is located at the bottom of the capsule (see Section 2.5). {

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Figure 2-5: Radial configuration of the reactor core

{ } Absorber cells do not contain fuel, and they are considered part of the reflector system (see Section 2.2.3). The centermost position in the core is an instrumentation position, which is used primarily for the startup source.

Figure 2-5 also shows the positions of the control drums (see Section 2.3), and the shutdown rods (see Section 2.4). The three control drums are located at vertices of the core, within the reflector system. The three shutdown rods are in locations of high reactivity worth in the active core, occupying lattice positions that would otherwise contain reactor cells.

The reactor uses various instrumentation systems to monitor core performance; neutron flux monitors, cell temperature sensors, and other instrumentation systems are used to provide the operating parameters, trips, and alarms. More information on instrumentation and control systems is provided in Section 2.7.

Important core design characteristics and operating conditions of the Aurora are summarized in Table 2-4.

Table 2-4: Reference core design characteristics and operating conditions

Characteristic	Value
Core thermal output (MWth)	4
Number of core enrichment zones	1
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{	}
Instrumentation positions	1
Control drums	3
Shutdown rods	3

2.2.2.4 Materials of the reactor core system

The reactor cell can is constructed using stainless steel 316L (SS316L), chosen for its low neutron absorption in the fast spectrum, significant operating and irradiation experience, and strength at operating conditions.

The reflector material used in the reactor cells is zirconium. Zirconium has well-characterized behavior in reactors and is chosen for its high neutron scattering cross-section and low neutron absorption.

The reactor core uses UZr metal fuel, often referred to as binary metal fuel. UZr also has significant operating experience in fast reactors, which operated over decades, and has many favorable characteristics for the Aurora design. Section 2.2.2.4.1 describes the fuel material in more detail.

The heat pipes consist of a SS316L wall, with a porous metal wick thermally bonded to the inner surface. They also contain a small amount of potassium working fluid. These materials are described further in Section 2.2.2.4.3.

2.2.2.4.1 Fuel material

2.2.2.4.1.1 Fuel type

The reactor core system employs a metal alloy in the form of uranium alloyed with 10% zirconium (U-10Zr). Metal fuel has a long history of use in U.S. fast reactors, beginning with the Experimental Breeder Reactor I (EBR-I) in 1951 and employed extensively in the Experimental Breeder Reactor II (EBR-II), which operated between 1964 and 1994 [22]. Over 130,000 metal fuel pins were irradiated in the experiments over decades [23].

The thermophysical properties of metal fuel provide favorable performance in fast reactor operating conditions [24]. U-10Zr has a very high theoretical density of 15.5 g/cm^3, which enables significant heavy metal core loadings for enhanced reactivity. The thermal conductivity of metal fuel is high, at 32 W/m-K, which helps to reduce peaking of fuel temperatures and precludes local hot spots. In addition, the low heat capacity of U-10Zr, at 260 J/kg-K, limits the amount of stored heat in the fuel while operating at temperature, enabling easier cooling of the fuel when shutting down or progressing through an abnormal event. Table 2-5 provides a summary of relevant U-10Zr thermophysical property values at a reference temperature.

Table 2-5: Selected thermophysical properties of U-10Zr at a reference temperature [24]

Material property	Value at 650 C
Density (g/cc)	15.5
Thermal conductivity (W/m-K)	31.7
Specific heat (J/kg-K)	248

2.2.2.4.1.2 Fission gas generation and fuel swell

During irradiation, fission gases form void pores in the fuel, which in turn causes metal fuel to swell. The phenomenology of fuel at higher irradiation levels is that once the fuel swells by approximately a third of its volume, the fission gas voids interconnect and the fission gasses are released to the upper plenum, with very little additional swelling.[7] This interconnection of fission gas voids typically occurs at a burnup of 2-3 atom per cent (at.%),[8] which means that at lower burnups, the vast majority of fission gases are retained in the fuel [25], [26].

Since the Aurora core's peak burnup never exceeds 1 at.%, pore interconnection and substantive fission gas release to the plenum is not expected. Although the burnup is not high enough to reach the maximum extent of swelling, the reactor cells are nevertheless designed with enough internal free volume to accommodate such swelling.

2.2.2.4.2 Eutectic formation considerations

Although the melting (solidus) temperature for the fuel is high, at 1,230 C, a more relevant limit for the Aurora fuel and the surrounding reactor cell can arises from considerations relating to eutectic formation. Eutectic effects between steel and fuel have been analyzed at length historically and were typically referred to as fuel-clad chemical interactions. The Aurora design does not employ clad fuel, so the eutectic considerations are referred to as fuel-steel chemical interaction. Effects caused by fuel-steel chemical interaction occur at elevated temperatures where interdiffusion occurs between the uranium component of the U-10Zr fuel and the stainless steel. This interdiffusion begins to form a lower melting-point eutectic.

A correlation developed by Argonne National Laboratory (ANL) and applied to the safety analyses in Chapter 5.1 shows that this process begins around 720 to 725 C but proceeds very slowly at this temperature (< 0.01 μm/s), increasing with temperature until reaching a rate of 0.1 μm/s at 830 C [27]. This correlation was developed with a series of dipping tests, where solid iron samples were dipped into a series of molten U-Fe baths, and the eutectic penetration rate was measured [28]. These tests represent bounding behavior of uranium and iron at high temperatures. In other words, the eutectic penetration rates experienced with actual fuel designs would be less than the rates observed in these tests.

Several irradiated fuel pins, irradiated to 2-10 at.% burnup, were studied using out-of-pile tests and showed eutectic formation rates that were bounded by the ANL correlation [27]. Lanthanide fission products and fuel alloying element redistribution enhance fuel-steel

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[7] Accordingly, metal fuel designed for reaching high burnups includes enough volume to accommodate swelling of the fuel, which helps to limit the stress applied to the fuel enclosure by the fuel itself.

[8] Fuel burnup can be expressed as percent of heavy metal atoms that have fissioned (at.%) or in units of fission energy produced per unit mass of heavy metal (GWd/MTHM or MWd/kgHM). A burnup of 1 at.% of burnup corresponds to roughly 9.4 GWd/MTHM. The nuclear fleet of large LWRs in the U.S. operates at around 50 MWd/kgHM.

chemical interaction. These effects occur in high burnup fuel and are not expected to be relevant for the Aurora design. Furthermore, the steel dimensions of interest in this experimental work were significantly smaller than the steel in the Aurora design. Thicker steel may lead to self-arresting eutectic penetration behavior due to the dilution of eutectic constituents as the eutectic progresses through the steel. For these reasons, the ANL correlation is substantially conservative for the Aurora design.

In light of this, eutectic penetration occurring between 720 C and 830 C progresses slowly, if at all in the Aurora design. Therefore, 720 C is considered a conservative operating limit, and there is considerable margin above that temperature before eutectic penetration occurs at rates that may lead to steel breach in timescales of interest to the Aurora design.

2.2.2.4.3 Heat pipe materials

The heat pipe wall is a SS316L tube. The wick is composed of high-porosity metal that provides capillary pressure to help drive the flow of the heat pipe working fluid. The wick is thermally bonded to the inner surface of the heat pipe wall. These materials, along with the potassium working fluid, were chosen because of their suitability at the operating temperature, heat flux, and irradiation, as well as their compatibility with one another.

2.2.2.4.3.1 Behavior in radiation

The materials used in the heat pipes are common materials in fast reactors with well understood behavior during irradiation. The capture cross-sections in the fast spectrum are very low; thus, the amount of activation is minimal. More information on activation of materials is in Chapter 3, "Radioactive materials to be produced in operation."

2.2.2.4.3.2 Impurity induced corrosion

Impurity induced corrosion was identified as the only significant life-limiting factor for heat pipe operation. Nonmetallic impurities, especially carbon and oxygen, located in the working fluid, container, wick, and surrounding structures (including the fuel and heat exchanger) diffuse into the working fluid and are carried toward the evaporator, where they concentrate. These impurities can precipitate and reduce flow in the wick, form low melting point eutectics with the container, or form ternary compounds with the container and working fluid. These problems can be avoided entirely by reducing the potential for contamination via proper material selection, and by removing contaminants with thorough cleaning and high-temperature bakeoff during the fabrication process [29]. High temperature baking is standard in heat pipe production, and visual inspection of the cold end of the heat pipe is sufficient to identify impurities. As such, standard production produces heat pipes of sufficient quality for long lifetimes, and there is no mechanism for producing heat pipes with significant impurities with standard production.

2.2.2.5 Design evaluation of the reactor core system

As described in Chapter 5.1, the top-level safety goal of the Aurora is to minimize the risk to the public and the environment by controlling dose. This top-level safety goal is accomplished by meeting two safety subgoals: (1) maintain fuel integrity and (2) maintain reactor cell can integrity. The primary challenge to both of these subgoals is temperature.

Eutectic formation, a phenomenon described in Section 2.2.2.4.2, can result in localized melting at temperatures lower than the melting temperature of fuel or steel components alone. Therefore, the reactor cell is designed and operated in such a way as to avoid exceeding the eutectic formation temperature for durations that would cause the cell can wall integrity to be compromised. For practical purposes, this is most easily accomplished by showing that the peak fuel temperature remains below 720C, which is the conservatively defined temperature at which eutectic formation onset may occur. This is achieved primarily by operating with sufficient temperature margins during normal steady state operation. These margins rely on inherent properties of the system such as high thermal conductivity (DB.RXS.01) and on setting the appropriate operating limits (DB.RXS.02).

The inherent negative reactivity feedback of the reactor core system (DB.RXS.03) and passive conduction of heat away from the core (DB.RXS.04) contribute to maintaining acceptable temperatures during transients.

The performance of the reactor core system is analyzed using the neutron transport code Serpent, and the thermal analysis code ANSYS.

2.2.2.5.1 Steady state operating condition

2.2.2.5.1.1 Fast neutron spectrum

The Aurora reactor operates in the fast spectrum, where neutrons born at fission energies of 2-3 MeV slow down only to about 1 keV to 1 MeV. Fast spectrum reactors are generally less sensitive to material selection because more materials are transparent to neutrons at those energies than at the thermal energy range. Materials such as stainless steel, for structural support, and zirconium, for neutron reflection, are suitable, and they are heavy enough to largely avoid neutron moderation.

Fast spectrum reactors also do not experience significant sensitivity to fission product poisoning effects, since most strong thermal-spectrum absorbers like xenon-135 have very small cross-sections at high energies. Figure 2-6 shows spectrum plots for the Aurora reactor in comparison to a pool-type, metal-fueled sodium fast reactor (SFR) and a pressurized water reactor (PWR). The Aurora core operates with a spectrum very similar to the SFR.

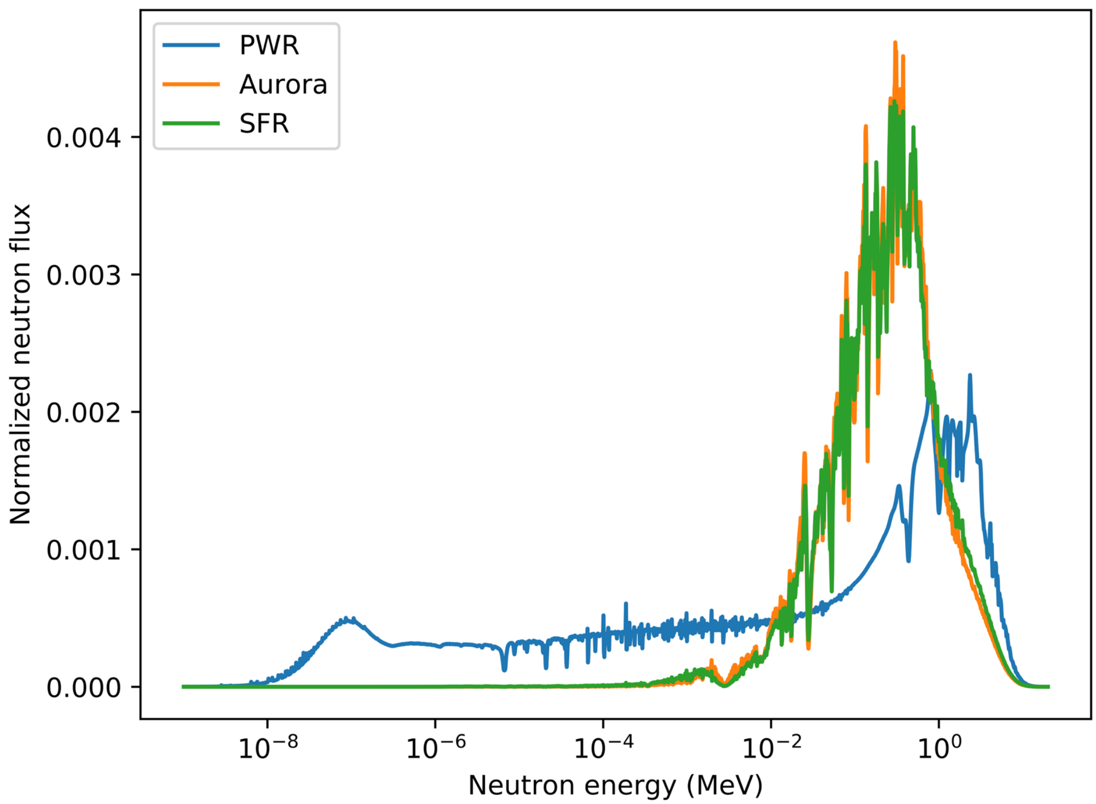

Figure 2-6: Comparison of the neutron spectra of the Aurora reactor to other designs

2.2.2.5.1.2 Core power distribution

The Aurora core operates with relatively low radial and axial power peaking factors, despite employing a constant core-wide U-235 enrichment. Small fast reactor cores, like the Aurora core, operate with relatively low radial and axial power peaking factors compared to most other reactor types due in large part to the unusually long mean free path of fast neutrons in the core and the small size of the core. Since the total neutron cross-section decreases with increasing incident neutron energy, a faster spectrum contributes to a large mean free path. A large neutron mean free path reduces core power peaking, helps the core to react to transients in a unified manner, and limits susceptibility to localized effects.

The maximum radial peaking factor in the Aurora core is less than 1.20 and the maximum axial peaking factor in the peak radial cell is 1.18, for a combined total maximum local peaking of 1.42. The maximum radial peaking decreases less than 1% from core beginning-of-life (BOL) to core end-of-life (EOL), and the location of peak power stays the same throughout fuel cycle life. Therefore, analysis of the peak temperatures conservatively uses the peaking factor at BOL. The radial reactor cell power peaking distribution at BOL is shown in Figure 2-7.

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Figure 2-7: Beginning of life (BOL) radial reactor cell power peaking distribution

The axial power distribution is nearly symmetric, with a maximum axial offset of -1.5% in the peak reactor cell, and nearly follows the idealized cosine shape often used as an approximation for axial power distributions in other reactor systems. The near-ideal axial power shape is due to the absence of an axial coolant temperature rise since the heat pipe maintains near-constant temperature along its entire length, as well as nearly symmetric reflector configurations above and below the active core region. The axial power distribution for the peak cell is shown in Figure 2-8.

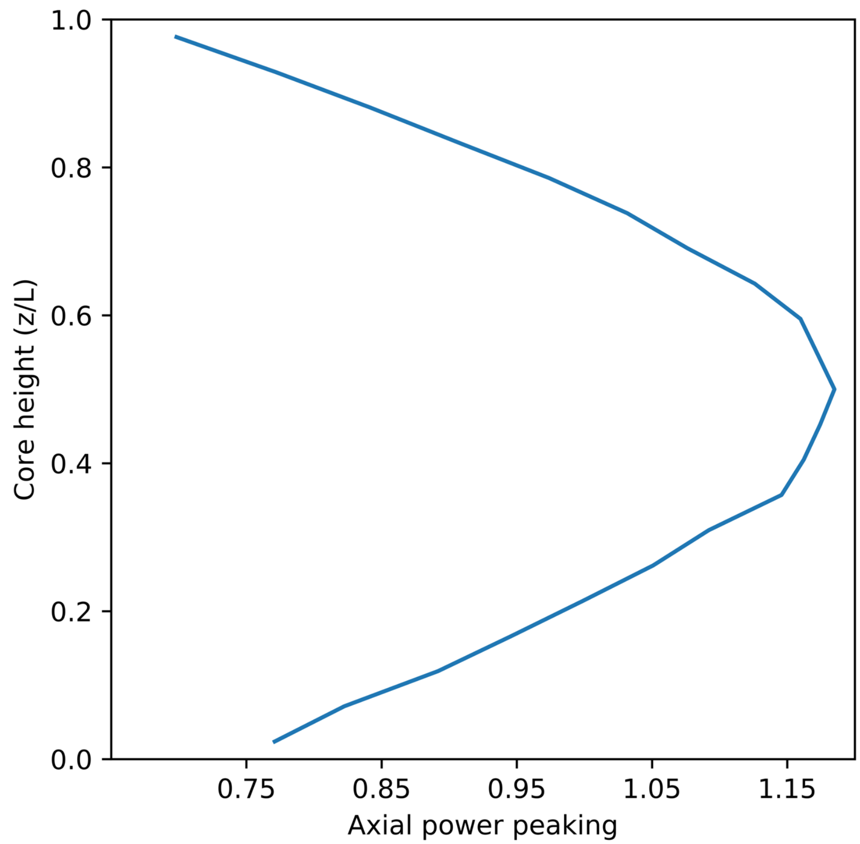

Figure 2-8: Axial power peaking profile in the peak reactor cell

The power peaking results in this section correspond to the core configuration with the absorber cells (see Section 2.2.3) in their maximum reactivity configuration (i.e., no absorber rods inserted). This is not an expected operating state for the Aurora; however, it does result in the highest radial power peaking values, which means that this radial power distribution is the bounding distribution for any potential configuration of the absorber cells. As such, the transient analysis presented in Chapter 5.1 is not sensitive to the configuration of the absorber cells, as the presence of absorber rods will only serve to reduce the power peaking below that which was modeled.

2.2.2.5.1.3 Peak fuel temperature

Both temperature and axial power peak at the core midplane due to the cosine-shaped axial power distribution and the near axially isothermal heat pipe working fluid temperature. The peak fuel temperature therefore occurs at the axial midplane of the peak reactor cell, within the innermost ring of reactor cells. Aside from the core power level, the most important driver of the peak fuel temperature during steady-state is the thermal conductivity of the fuel.

A conservative assumption was made in the calculation of fuel temperature: the thermal conductivity of the fuel was taken at 70% of its nominal value. This was done to account for the degradation in thermal conductivity associated with increasing porosity generated during irradiation [24]. Oklo analysis showed that an assumption of 30% degradation in thermal conductivity for the entire cycle is conservative because at low burnups the actual degradation would be significantly less than 30%.

Using this conservative approach, operating at the core maximum power level of 4 MWth results in a steady-state maximum fuel temperature of 640 C at the core midplane in the reactor cell at the peak location. Figure 2-9 and Figure 2-10 show the radial temperature distribution in the peak reactor cell at the core mid-plane, for the full reactor cell and for the just the fuel, respectively. Figure 2-11 shows the axial temperature distribution of the peak reactor cell at multiple radial locations. These temperature distributions were calculated using the ANSYS code. {

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Figure 2-9: Radial temperature distribution in the peak reactor cell at the peak axial position

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Figure 2-10: Radial temperature distribution in the fuel of the peak reactor cell at the peak axial position

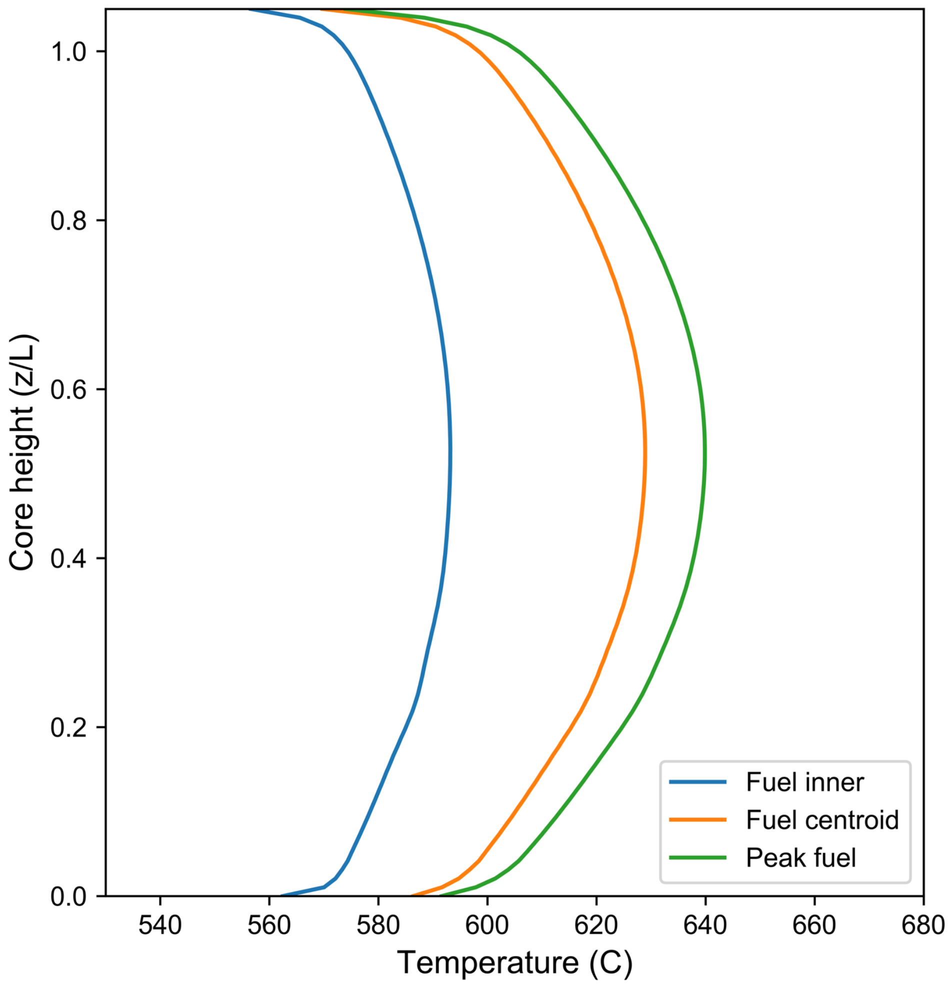

Figure 2-11: Axial temperature profile of the peak reactor cell

The peak fuel temperature of 640 C during steady state operation at full power provides approximately an 80 C margin to the conservatively defined operating limit of 720 C. As described in the safety analysis in Chapter 5.1, this margin is large enough to prevent eutectic formation during the maximum credible accident. To ensure that proper margins are maintained during normal operation, the appropriate limit will be placed on the core power level (DC.RXS.02.A).

2.2.2.5.2 Inherent feedback mechanisms

The Aurora core possesses a net negative power coefficient of reactivity, which contributes to safe behavior during transient conditions. The primary contributor to the net negative power coefficient of reactivity in the Aurora reactor core is the temperature feedback coefficient of reactivity. The temperature feedback coefficient components of interest in the Aurora reactor are the fuel thermal expansion coefficient of reactivity, the fuel Doppler coefficient of reactivity, and the structural material thermal expansion coefficient of reactivity.

The fuel thermal expansion temperature coefficient of reactivity is the reactivity change in the core associated with the material expansion with increasing temperature of the metal fuel. The fuel Doppler coefficient is the reactivity change associated with the broadening of the fuel's reaction cross-sections with increased temperature. The reactor cell thermal expansion coefficient includes the effects of material expansion of the reactor cell can wall and the heat pipe wall. The fuel thermal expansion coefficient is the most impactful driver of negative reactivity feedback since it is both relatively large (as shown in Table 2-6) and is the quickest to respond to power changes.

Table 2-6: Temperature coefficients of reactivity at full-power conditions

Component	Reactivity coefficient (pcm/K)
Fuel thermal expansion	-0.50
Fuel doppler	-0.15
Reactor cell thermal expansion	-0.07
Baseplate thermal expansion	-1.40
Net	**-2.12**

Other temperature coefficients of reactivity contribute to the net reactivity coefficient beyond those displayed in Table 2-6, such as the radial reflector temperature coefficient. However, these additional components will operate on slightly slower timescales since they rely on heat generated in the fuel to be conducted radially outward through the core. Their contribution to the net coefficient is expected to be large and negative, but less ultimately meaningful due to their slower timescale of response.

A design commitment (DC.RXS.03.A) is taken to demonstrate a net negative power coefficient of reactivity of the system. This commitment ensures that all components of the reactivity feedback discussed here, and any other less impactful components, collectively contribute to safe behavior during transient conditions. The safety analysis in Chapter 5.1 conservatively neglects reactivity feedback entirely, and therefore does not rely on the magnitude of the negative feedback coefficient.

2.2.2.5.3 Passive conduction of decay heat

As described in the safety analysis in Chapter 5.1, the Aurora reactor does not rely on active cooling during the decay heat phase following shutdown during the maximum credible accident. The reactor passively conducts heat between the systems within the reactor module (the reactor core system, reflector system, shielding system, enclosure system, and heat exchanger system), distributing heat throughout the substantial thermal mass presented by the module. The heat is then removed from the module via natural convection at the surface of the module shell. The safety analysis shows that this passive heat removal is sufficient to maintain acceptable fuel temperatures and meet the top-level safety goal of the Aurora in the event of the maximum credible accident.

Design commitments are taken for each system to ensure that it is properly configured to provide conduction between systems (including DC.RXS.04.A), and a commitment is taken to test the ability of the module to passively remove heat at decay heat levels during startup testing (DC.RXS.04.B). In addition, a design commitment is made for the building system to ensure that the reactor emplacement supports passive cooling of the module shell (DC.BAS.01.A).

2.2.2.6 *Summary of the reactor core system*

Design basis:

DB.RXS.01 The reactor core system uses metal fuel with well characterized properties.

Design evaluation summary:

The analysis in this section has shown that the steady state operating temperature of the reactor core system provides substantial margin to fuel-steel eutectic formation temperatures at full power. The safety analysis in Chapter 5 shows that this margin is sufficient to maintain the safety and operational goals of the Aurora in the event of the maximum credible accident. A design commitment is taken to the ensure that the fuel used in the Aurora meets the critical characteristics required to maintain the safety and operational goals, and the appropriate programmatic controls are in place to ensure the commitments are met.

Design commitments and programmatic controls:

DC.RXS.01.A The fuel in the reactor system is procured according to 10 CFR Part 50 Appendix B, with a critical characteristic of thermal conductivity.

(see Oklo Quality Assurance Program Description)

Design basis:

DB.RXS.02 The reactor core system is operated at steady state thermal power levels that prevent damage to the system during transients.

Design evaluation summary:

The analysis in this section has shown that the steady state operating temperature of the reactor core system provides substantial margin to fuel-steel eutectic formation temperatures at full power. The safety analysis in Chapter 5 shows that this margin is sufficient to maintain the safety and operational goals of the Aurora in the event of the maximum credible accident. A design commitment is taken to the steady state operating power level to ensure that this margin is maintained, and the appropriate programmatic controls are in place to ensure the commitments is met.

Design commitments and programmatic controls:

DC.RXS.02.A The power level of the reactor system is limited to 4 MWth.

See also DC.ICS.01.A through D

Design basis:

DB.RXS.03 The reactor core system has inherently negative reactivity feedback.

Design evaluation summary:

The analysis in this section has shown that the reactivity feedback of the reactor core system is dominated by the large negative temperature reactivity feedback due to thermal expansion of metal fuel. Although reactivity feedback is not relied on in the safety analysis presented in Chapter 5, a design commitment is taken for a negative power coefficient of reactivity and the appropriate programmatic controls are in place to ensure that this commitment is met.

Design commitments and programmatic controls:

DC.RXS.03.A The net power coefficient of reactivity of the reactor core system is negative.

SUT.RXS.03 (see Chapter 14)

Design basis:

DB.RXS.04 The reactor core system provides a pathway to conduct heat from the fuel to the surrounding systems and ultimately to reject it to the environment.

Design evaluation summary:

This section described the layout of the reactor core system, a matrix of hexagonal reactor cells. Decay heat is conducted away from the fuel in both the axial and radial directions both within and among the reactor cells and outward toward surrounding systems. The transient analysis in Chapter 5 shows that, when configured as designed, the reactor core system provides adequate heat conduction to maintain fuel temperatures below their required limits during the decay heat phase of the maximum credible accident (without active cooling). Design commitments are made to ensure proper as-built configuration of the core prior to operation and to demonstrate that the reactor core system can be cooled via conduction. The appropriate programmatic controls are in place to verify them.

Design commitments and programmatic controls:

DC.RXS.04.A The critical components of the reactor core system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.

SUT.RXS.04.A (see Chapter 14)

DC.RXS.04.B The reactor core system can be cooled by conduction through the surrounding systems (reflector system, shielding system, heat exchanger system, and reactor enclosure system) and subsequent convection from the module shell after shutdown.

SUT.RXS.04.B

2.2.3 Reflector system

2.2.3.1 Introduction to the reflector system

The reflector system functions primarily to reflect neutrons back into the core, improving fuel utilization, increasing neutron economy, and flattening the power profile of the core. It consists of absorber cells, and solid reflector blocks. The reflector system is configurable during initial startup to adjust for the initial core reactivity, then remains fixed during normal operations.

2.2.3.2 Bases of the reflector system

2.2.3.2.1 Design basis of the reflector system

The design basis for the reflector system, a subsystem of the reactor system (RXS), is as follows:

DB.RXS.05 The reflector system provides a pathway to conduct heat from the reactor core system to the surrounding systems and ultimately to reject it to the environment.

2.2.3.2.2 Performance bases of the reflector system

The reflector system is also designed to meet all of the following performance bases:

- The reflector system is designed to improve fuel utilization.
- The reflector system is designed to be configurable during startup testing to adjust for the initial core reactivity and remains fixed during normal operation.
- The reflector system is designed to reduce neutron fluence to the capsule and module shell.

2.2.3.3 Description of the reflector system

The reflector system functions to enhance fuel utilization and adjusts for uncertainties in the core initial reactivity. It consists of three major components:

1. Axial reflector blocks integrated into the reactor cells of the reactor core system (see Section 2.2.2)
2. A reconfigurable ring of six absorber cells, located in the center of the reactor core system
3. A fixed reflector region surrounding the outermost ring of cells, made up of zirconium and stainless steel blocks

Absorber cells are similar in structure to reactor cells, but they do not contain fuel. Instead of fuel, the absorber cells contain a reflector block that spans the full height of the core region. They also contain a smaller heat pipe than the reactor cells, which functions to distribute heat axially throughout the absorber cell but is not actively cooled by the heat exchanger system.

The reflector block in the absorber cells is either a solid monolithic block or contains six holes that can be configured to contain absorber rods or reflector rods. The absorber rods consist of a hollow steel rod containing boron carbide absorber, and the reflector rods are solid steel. Absorber cells can be configured with any combination of absorber and reflector rods. Figure 2-12 shows a schematic of an absorber cell.

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Figure 2-12: Absorber cell schematic

Absorber cells are configured during startup testing to adjust the initial reactivity of the core. This initial adjustment ensures the core has the desired excess reactivity, and that the design commitments for the reactivity worth of (1) the control drum system (Section 2.3) and (2) the shutdown rod system (Section 2.4) can be met. If the initial reactivity is too low, absorber rods can be replaced with reflector rods in the absorber cells. If the initial reactivity is too high, reflector rods can be replaced with absorber rods. After this initial adjustment, the configuration is fixed during normal operation. A representative configuration of the absorber cells is shown in Figure 2-13.

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Figure 2-13: Schematic showing a representative configuration of absorber cells in the center of the core

The fixed reflector region consists of hexagonal blocks of zirconium and stainless steel that surround the last ring of reactor cells and extend out to the capsule, as shown in Figure 2-5. These blocks provide additional neutron reflection. The control drum system is located within the fixed reflector region.

2.2.3.4 Materials of the reflector system

The reflector materials used in the reactor are zirconium and SS316L, which have well-characterized behavior in reactors. Zirconium is chosen for its high neutron scattering cross-section and low neutron absorption. SS316L is durable, inexpensive, has an acceptable neutron scatter-to-capture cross-section, and possesses high strength at operating temperatures.

Boron carbide is used as the neutron absorber in the absorber rods. Boron carbide was chosen as the neutron poison material due to its relatively high neutron absorption cross-section in the fast spectrum, at 1 barn. Other benefits of boron carbide include a high melting temperature, chemical stability at high temperatures, and low production of gamma radiation upon neutron absorption.

The maximum operating temperatures of the reflector system is bounded by the peak fuel temperature of 640 C, which is much lower than the melting point for the reflector system materials, and well below their thermal design limits.

2.2.3.5 Design evaluation of the reflector system

2.2.3.5.1 Steady state operating condition

The reflector system is completely passive. The only adjustments made to it are during startup testing, and the configuration remains fixed during operation. The primary function of the reflector system is to reflect neutrons back into the core, which has the following benefits:

- maximizing fuel usage, which increases neutron economy, and
- flattening the power profile of the core

The performance of the reflector system is analyzed using the neutron transport code Serpent. Because the neutronic performance of the reflector system relates to its performance bases only, the related analysis is not presented.

2.2.3.5.2 Passive conduction of decay heat

As described in the safety analysis in Chapter 5.1, the Aurora reactor does not rely on active cooling for decay heat removal, following shutdown after the maximum credible accident. Decay heat is passively conducted from the fuel to the substantial amount of available thermal mass of the other systems within the reactor module. This conduction and thermal mass availability applies to the systems within the reactor module, which include: (1) the reactor core system, (2) the reflector system, (3) the shielding system, (4) the enclosure system, and (5) the heat exchanger system. The heat is then removed from the reactor module via natural convection of air at the surface of the module shell. The safety analysis shows that this passive heat removal is sufficient to maintain acceptable fuel temperatures and meet the top-level safety goal of the Aurora in the event of the maximum credible accident.

Design commitments are taken for each system to ensure that it is properly configured to provide a conduction pathway between systems (including DC.RXS.05.A for the reflector system), and a commitment is taken to test the ability of the reactor module to passively remove heat at decay heat levels during startup testing (DC.RXS.04.B). In addition, a design commitment is made for the building system to ensure that the reactor emplacement supports passive cooling of the module shell (DC.BAS.01.A).

2.2.3.6 Summary of reflector system

Design basis:

DB.RXS.05 The reflector system provides a pathway to conduct heat from the reactor core system to the surrounding systems and ultimately to reject it to the environment.

Design evaluation summary:

This section describes the reflector system, a matrix of hexagonal reflector and absorber cells, and blocks of fixed reflector material that surround the reactor core system. Decay heat is conducted away from the reactor core system and outward toward surrounding systems. The transient analysis in Chapter 5 shows that, when configured as designed, the reflector system provides adequate heat conduction to maintain fuel temperatures below their required limits during the decay heat phase of the maximum credible accident (without active cooling). A design commitment is made to ensure proper as-built configuration of the reflector system prior to operation and the appropriate programmatic controls are in place to verify it.

Design commitments and programmatic controls:

DC.RXS.05.A The critical components of the reflector system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.

POT.RXS.05.A (see Chapter 14)

2.2.4 Shielding system

2.2.4.1 Introduction to the shielding system

The shielding system, a subsystem of the reactor system, functions to (1) limit occupational radiation exposures, and (2) keep radiation fluence to equipment and reactor enclosures below a level at which embrittlement or other significant radiation damage could occur. It is made up of components located throughout the reactor module, including components within the reactor cells of the reactor core system, functioning collectively to meet shielding design and performance goals.

2.2.4.2 Bases of the shielding system

2.2.4.2.1 Design basis of the shielding system

The design basis for the shielding system, a subsystem of the reactor system (RXS), is as follows:

DB.RXS.06 The shielding system provides a pathway to conduct heat from the reactor core system and reflector system to the surrounding systems and ultimately to reject it to the environment.

2.2.4.2.2 Performance bases of the shielding system

The shielding system is also designed to meet all of the following performance bases:

- The shielding system minimizes occupational dose rates and exposures.
- The shielding system is designed to minimize radiation fluence, and activation, to the power conversion system fluid in the heat exchanger system.
- The shielding system is designed to reduce the total fluence to the module shell.
- The shielding system is designed to minimize the radiation exposure to equipment in the module equipment housing.

2.2.4.3 Description of the shielding system

The shielding system is composed of five major regions:

1. The heat exchanger shield, integrated into the reactor cells of the reactor core system (see Section 2.2.2)
2. The capsule backfill shield, located above the fixed reflector region
3. The top shield, located near the top of the capsule
4. The radial shield, located in the radial gap between the capsule and the module shell
5. The bottom shield, located in the axial gap between the capsule and module shell below the capsule

Collectively, these components function to meet shielding design and performance bases. Figure 2-14 shows the axial layout, and Figure 2-15 shows the radial layout of each major component of the shielding system. All of the shields utilize boron carbide.

The heat exchanger shield is made up of individual axial shielding components integrated into each reactor cell, as shown in Figure 2-3 in Section 2.2.2. The purpose of the heat exchanger shield is to attenuate the axial core radiation flux, in order to limit activation of the power conversion system fluid in the heat exchanger.

The capsule backfill shield is located outside the core lattice, above the fixed reflector and below the capsule lid. Its purpose is to reduce neutron pathways outside the core lattice.

The purpose of the top shield is to attenuate neutron pathways out of the core into the module equipment housing. The top shield rests above all reactor module internal structures save the upper insulator, which is positioned directly below the capsule lid. The top shield is removable to allow access to the reactor internals for maintenance, and has penetrations for shutdown rod drive lines, control drum drive shafts, power conversion system piping, and instrumentation cabling.

The radial shield is located in the radial space between the capsule and the module shell. The purpose of the radial shield is twofold. First to minimize the radiation exposure to the module shell and second to conduct heat radially from the capsule to the module shell and which allows the heat ultimately to reject to the environment.

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Figure 2-14: Axial schematic showing regions of the shielding system

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Figure 2-15: Radial schematic showing regions of the shielding system

2.2.4.4 Materials of the shielding system

All components of the shielding system are made of stainless steel, boron carbide and zirconium hydride. Stainless steel 304 (SS304) and SS316L are used for structural material, and boron carbide is used as the primary absorber material. The behavior of these materials is well-characterized under irradiation. Boron carbide and SS316L are described in more detail in Sections 2.2.2.4 and 2.2.3.4. Metal hydrides, like Zirconium hydride, may be added to enhance shielding performance in some locations.

2.2.4.5 Design evaluation of the shielding system

2.2.4.5.1 Steady state operating condition

The shielding system is a structure and therefore completely passive. The only function of the shielding system is to limit occupational radiation exposures and keep radiation fluence to equipment and enclosures below a level at which embrittlement or other significant radiation damage may occur.

The performance of the shielding system is analyzed using the neutron transport code Serpent. Because the neutronic performance of the shielding system relates to its performance bases only, the related analysis is not presented.

2.2.4.5.2 Passive conduction of decay heat

As described in the safety analysis in Chapter 5.1, the Aurora reactor does not rely on active cooling for decay heat removal, following shutdown after the maximum credible accident. Decay heat is passively conducted from the fuel to the substantial amount of available thermal mass of the other systems within the reactor module. This conduction and thermal mass availability applies to the systems within the reactor module, which include: (1) the reactor core system, (2) the reflector system, (3) the shielding system, (4) the enclosure system, and (5) the heat exchanger system. The heat is then removed from the reactor module via natural convection of air at the surface of the module shell. The safety analysis shows that this passive heat removal is sufficient to maintain acceptable fuel temperatures and meet the top-level safety goal of the Aurora in the event of the maximum credible accident.

Design commitments are taken for each system to ensure that it is properly configured to provide a conduction pathway between systems (including DC.RXS.06.A for the shielding system), and a commitment is taken to test the ability of the reactor module to passively remove heat at decay heat levels during startup testing (DC.RXS.04.B). In addition, a design commitment is made for the building system to ensure that the reactor emplacement supports passive cooling of the module shell (DC.BAS.01.A).

2.2.4.6 Summary of the shielding system

Design basis:

DB.RXS.06 The shielding system provides a pathway to conduct heat from the reactor core system and reflector system to the surrounding systems and ultimately to reject it to the environment.

Design evaluation summary:

This section describes the shielding system, which primarily functions to limit occupational radiation exposures and keep radiation fluence to equipment and enclosures below a level at which embrittlement or other significant radiation damage may occur. The shielding system also functions to conduct heat from the reactor core system and reflector system outward toward the module shell. The transient analysis in Chapter 5 shows that, when configured as designed, the shielding system provides adequate heat conduction to maintain fuel temperatures below their required limits during the decay heat phase of the maximum credible accident (without active cooling). A design commitment is made to ensure proper as-built configuration of the shielding system prior to operation and the appropriate programmatic controls are in place to verify it.

Design commitments and programmatic controls:

DC.RXS.06.A The critical components of the shielding system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.

POT.RXS.06.A (see Chapter 14)

SUT.RXS.06.A1 and A2

2.3 Control drum system

2.3.1 Introduction to the control drum system

The control drum system functions to control reactivity letdown with fuel depletion. It has no reactor shutdown function, as shutdown is achieved using the shutdown rod system (see Section 2.4). The control drum system consists of three control drums of equivalent worth, located at three vertices of the active core lattice, within the reflector region (see Section 2.2.3). Each drum (i.e., control drum) contains a half-cylinder of absorber, and a half-cylinder of reflector. The control function is achieved by rotating the drums to adjust the relative positions of absorber and reflector, inserting positive reactivity by rotating the absorber portion out of the core, and inserting negative reactivity by rotating the absorber portion into the core. Generally, the control drums are designed to rotate the absorber portion very slowly out of the core to add reactivity to compensate for reactivity loss with fuel depletion, over fuel cycle life.

2.3.2 Bases of the control drum system

2.3.2.1 Design basis of the control drum system

The design basis for the control drum system (CDS) is as follows:

DB.CDS.01 The control drum system is designed to limit both the rate and magnitude of reactivity insertion that the system can achieve so as to minimize the effect of an unintended reactivity insertion.

2.3.2.2 Performance basis of the control drum system

The control drum system is also designed to meet the following performance basis:

- The control drum system controls reactivity letdown with fuel depletion.

2.3.3 Description of the control drum system

2.3.3.1 Configuration and location of the control drum system

Each control drum consists of a SS316L cylindrical enclosure, containing a half-cylinder of boron carbide absorber material and a half-cylinder of zirconium reflector. The components of the control drum are shown schematically in Figure 2-16.

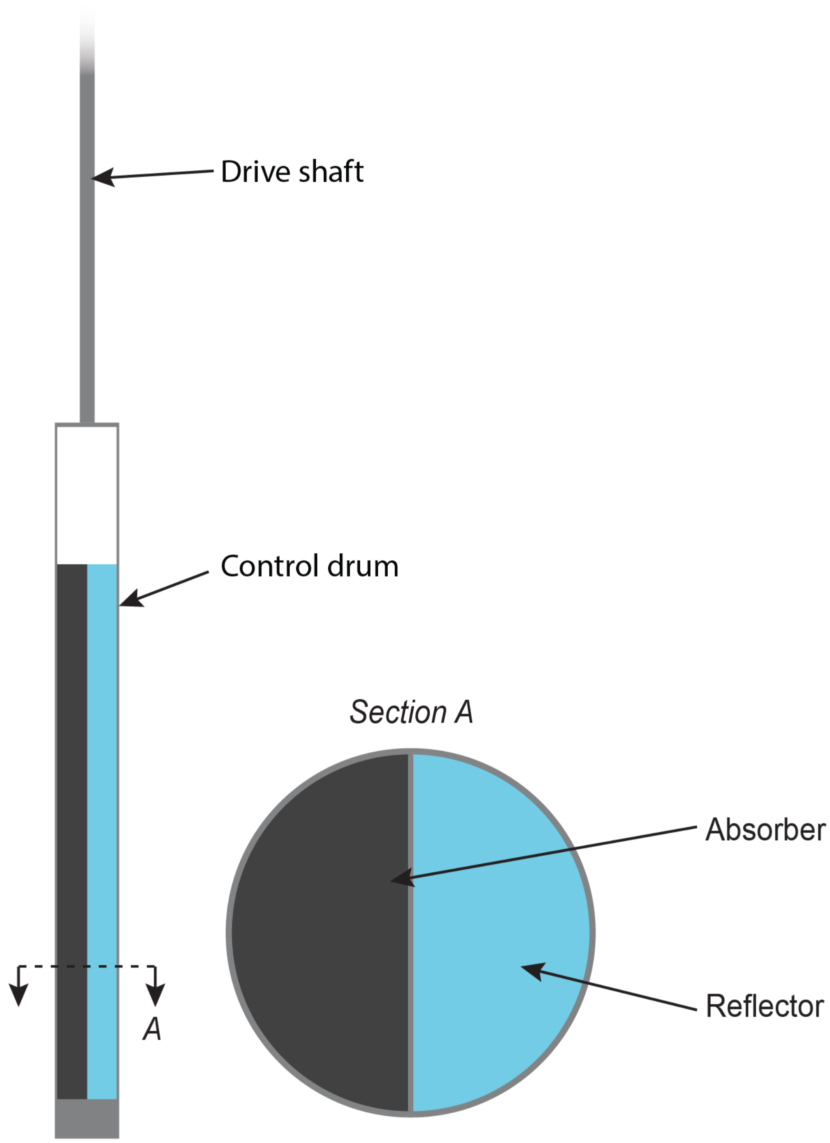

Figure 2-16: Control drum axial schematic and section view

The reflector and absorber regions of the control drum are the same height as the core region. The drums are placed at three of the vertices of the active core cell lattice, spaced evenly with rotational symmetry around the outer perimeter as shown in Figure 2-5. The control function is achieved by rotating the drums. The boron carbide absorber is rotated into the core to reduce core reactivity and rotated out of the core to increase core reactivity, as shown in Figure 2-17.

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Figure 2-17: Control drum rotation schematic showing absorber rotated fully in and fully out

2.3.3.2 Operation of the control drum system

The control drum rotation is driven by a shaft that extends into the module equipment housing (see Section 2.5), where it is coupled to the control drum drive motor via the appropriate gearing. The drums are not rotated continuously, rather the drive motor and gearing are designed to rotate the control drums a very small amount at regular intervals to compensate for reactivity letdown associated with fuel depletion. The maximum total rotation of the control drums is 180 degrees over the course of operation.

The control drums are not relied upon in any safety analysis. The drums are designed to stop in place on a reactor trip signal or loss of control signal. This function is achieved using an electromagnetic brake that allows drum rotation when energized and prevents rotation when de-energized.

To reduce the impact of a spurious drum rotation, the drive motors and gearing are designed to limit the maximum rotation speed of the drum. This limit is achieved by specifying the maximum speed of the motor, and the transmission ratio of the gearing. The maximum rotation speed is chosen to limit the rate of reactivity insertion, as described in Section 2.3.5.

To further reduce the impact of spurious drum rotation, the total reactivity worth of the drums is also limited. This limits the total reactivity insertion possible during a single spurious movement, as described in Section 2.3.5.

Each control drum is instrumented and controlled as described in Section 2.7.

2.3.4 Materials of the control drum system

The reflector and absorber materials, zirconium and boron carbide, are the same as those used in the reflector system and are used for the same reasons described in Section 2.2.3. The drum cylinder and internal separator are made of SS316L, which is chemically compatible with both boron carbide and zirconium.

2.3.5 Design evaluation of the control drum system

The control drums cannot insert sufficient reactivity to challenge the safety of the core, because the system is limited in multiple ways. First, the total reactivity worth of the drums is low, which is enabled by a very small reactivity letdown over fuel cycle life. Second, the speed of rotation of the drums is limited physically by the motor and gearing specified.

The reactivity worth of the control drum system is analyzed using the neutron transport code Serpent.

2.3.5.1 Reactivity letdown over fuel cycle life and drum worth limit

The combined worth of the drums is limited to 700 pcm, to compensate for the reactivity letdown over fuel life with some additional margin. By design, the control drums are only required to compensate for the reactivity loss associated with fuel depletion, as they have no shutdown function. This means that from absorber fully rotated in, to absorber fully rotated out, the drums can provide only 700 pcm of reactivity insertion.

2.3.5.2 Control drum rotation limits

The control drums rotate at regular intervals, rather than continuously during operation, to offset the reactivity letdown with fuel burnup. The low rate of reactivity letdown requires an insertion rate of { }. At times, including during startup, larger drum rotation speeds is required to make larger reactivity adjustments. As a result, a drum speed rotation limit of 1×10-2 deg/sec is set. At this drum speed, it takes roughly five hours to fully insert the drum absorbers. This maximum speed is set to provide an acceptable maximum rotation time while limiting the potential challenge to core safety, as analyzed in Chapter 5.1. The control drum motor and gearing are designed to be incapable of exceeding this rotational speed limit, and this limit is verified using the appropriate programmatic controls, as summarized in Section 2.3.6.

In addition, an electromagnetic brake prevents drum rotation in the direction of positive reactivity insertion when the drums are not being actively rotated and whenever a reactor trip signal is received (see Section 2.7.3). This provides an additional layer of protection against spurious drum rotation, such that if a reactor trip signal is received, the electromagnetic brake engages, stopping the drums in place.

2.3.6 Summary of the control drum system

Design basis:

DB.CDS.01 The control drum system is designed to limit both the rate and magnitude of reactivity insertion that the system can achieve so as to minimize the effect of an unintended reactivity insertion.

Design evaluation summary:

The analysis in this section shows that the control drum system has a low reactivity worth and a low maximum drum rotation speed. These design features limit the rate of reactivity insertion that the control drum system can provide. Further, stepper motors are used to prevent unintended rotation of the control drums. The safety analysis in Chapter 5 shows that these limits provide sufficient protection against spurious reactivity insertion. Design commitments are taken for these limits, and the appropriate programmatic controls are in place to ensure the design commitments are met.

Design commitments and programmatic controls

DC.CDS.01.A The maximum rotation speed of the drums is limited to 1×10^{-2} deg/sec.

POT.CDS.01.A (see Chapter 14)

DC.CDS.01.B The total reactivity worth of the drums is less than 700 pcm at all operating conditions.

SUT.CDS.01.B

DC.CDS.01.C The control drum actuators use stepper motors to eliminate the possibility of unintentional rotation.

POT.CDS.01.C

2.4 Shutdown rod system

2.4.1 Introduction to the shutdown rod system

The shutdown rod system functions to shut down the reactor core by putting it in a sub-critical state. During normal operation, the shutdown rods are fully withdrawn from the core and reactivity control is achieved using negative temperature feedback and the control drum system (see Section 2.3). The shutdown rod system consists of three redundant shutdown rods of equal worth, such that each rod can independently provide sufficient negative reactivity to shut down the reactor. The rods are suspended above the core by electromagnets. The rods insert via gravity when the electromagnets are de-energized, and each rod has an independent electromagnet to ensure redundancy. A loss of power results in rod (i.e., shutdown rod) release and insertion, and subsequent reactor shutdown.

2.4.2 Bases of the shutdown rod system

2.4.2.1 Design bases of the shutdown rod system

The design bases for the shutdown rod system (SRS) are as follows:

DB.SRS.01 The shutdown rod system provides sufficient negative reactivity to achieve cold shutdown with insertion of one rod.

DB.SRS.02 The shutdown rod system fully inserts the shutdown rods within a sufficient time after receiving a trip signal to prevent damage to the reactor.

2.4.2.2 Performance bases of the shutdown rod system

The shutdown rod system is also designed to meet all of the following performance bases:

- The shutdown rod system is designed to shut down the reactor and hold it shutdown, under conditions which allow for activities such as fuel loading, inspection, and repair.
- The shutdown rod system is designed to prevent a stuck rod.
- The shutdown rod system fails in the tripped (inserted) condition.
- The shutdown rod system requires deliberate action to initiate removal of the shutdown rods from the core.

2.4.3 Description of the shutdown rod system

2.4.3.1 Configuration and location of the shutdown rod system

The shutdown rods consist of an absorber portion and a drive line. The absorber portion of the shutdown rod is a SS316L cylinder filled with boron carbide. The drive line is a SS316L rod that extends into the module equipment housing (see Section 2.5), where it is connected to the shutdown rod retrieval mechanism via an electromagnet. A schematic of a shutdown rod is shown in Figure 2-18.

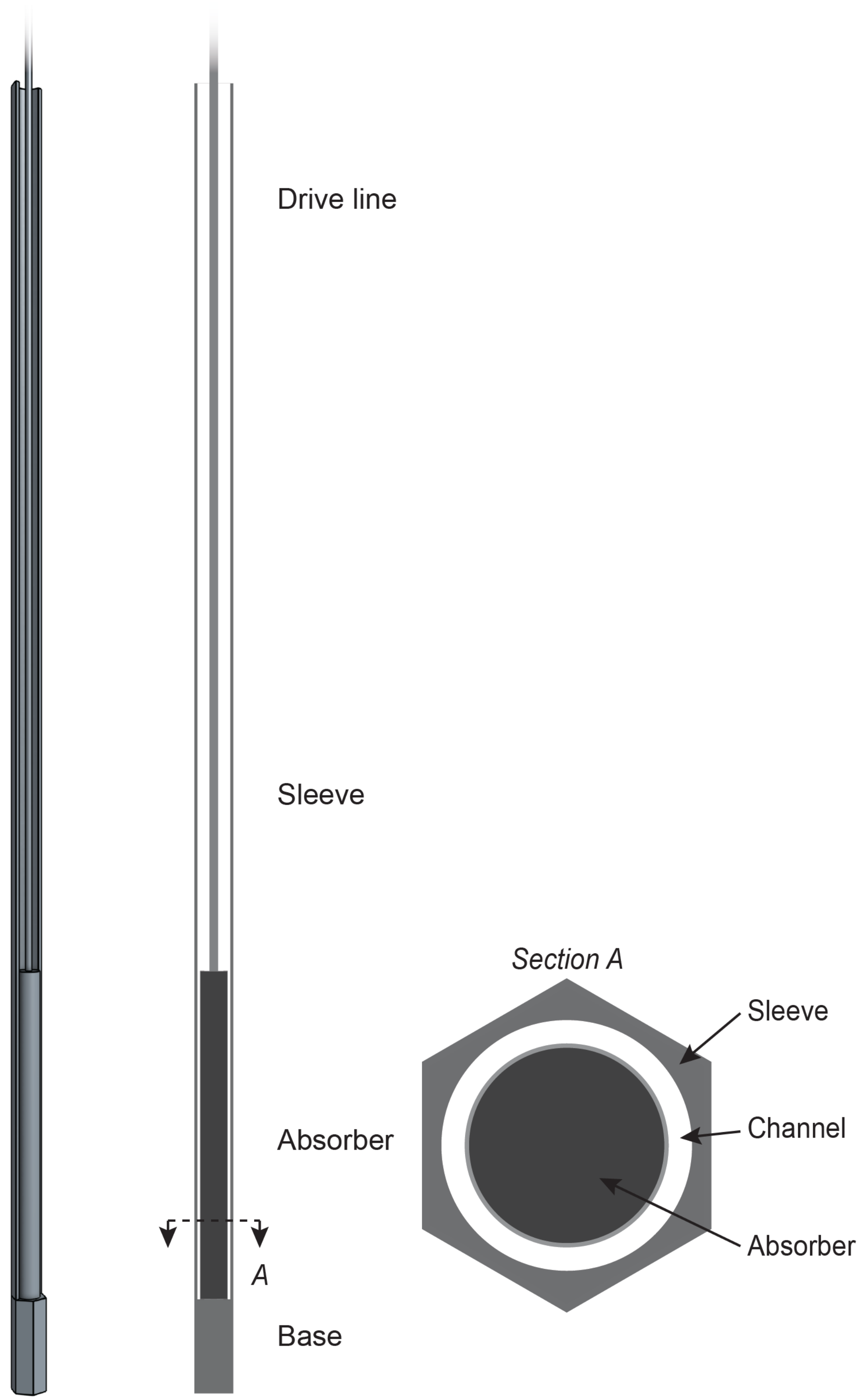

Figure 2-18: Shutdown rod schematic

Figure 2-18, Section A shows a cross-sectional view of an inserted shutdown rod. The shutdown rod travels vertically within the shutdown rod sleeve, and the space between the rod and the sleeve is referred to as the channel. During normal operation the rod is held outside of the core and the portion of the sleeve in the active core region is empty.

2.4.3.2 Operation of the shutdown rod system

During normal operation, the shutdown rods are fully withdrawn from the core and the only function of the shutdown rod system is to shut down the reactor. During reactor startup, the shutdown rods are slowly withdrawn as the core temperature increases to nominal operating temperature. The shutdown rod system consists of three redundant shutdown rods and their associated electromagnets and retrieval mechanisms. The electromagnets and withdrawal mechanisms are mechanically and electrically independent for redundancy. The release of the shutdown rods is actuated by the reactor trip system, which is further described in Section 2.7.3.

Each rod is independently withdrawn by its own shutdown rod withdrawal mechanism. The shutdown rod withdrawal mechanisms are used to withdraw the shutdown rods from the core and suspend them in the withdrawn position. The shutdown rods are fully withdrawn from the reactor core during normal operation as shown in Figure 2-19.

When a trip signal is received from the reactor trip system, the electromagnet suspending each rod de-energizes, and all three of the rods are inserted via gravity. A loss of power to a shutdown rod electromagnet, for any reason, also causes a rod release. The released rod falls by gravity into the reactor core and is guided by the shutdown rod sleeve, which reduces the probability of a shutdown rod misalignment. A hard stop with a spring and damper is used to prevent damage to the shutdown rod from impact when it inserts.

The mass of the shutdown rods can be suspended by an electromagnet, and do not require a mechanical latching or clamping mechanism to hold the rods. This substantially reduces the complexity of the system and improves the reliability of rod release. The electromagnet is attached to a linear actuator that is used to withdraw the rod from the core during startup. To withdraw a rod, the linear actuator lowers the electromagnet into contact with the top of the drive line. After the electromagnet is engaged, the linear actuator raises the electromagnet and the attached rod out of the core. The components of the shutdown rod mechanism are shown schematically in Figure 2-20.

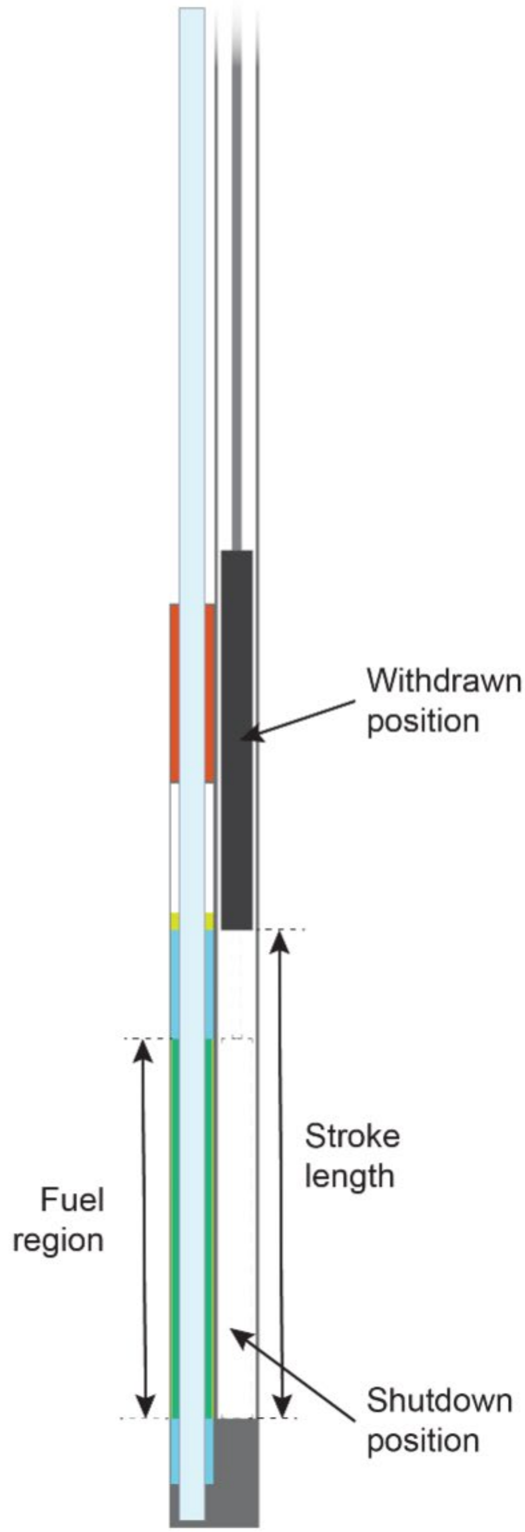

Figure 2-19: Schematic of shutdown rod in fully withdrawn position

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Figure 2-20: Schematic of shutdown rod mechanisms

2.4.4 Materials of the shutdown rod system

The absorber material used for the shutdown rods is boron carbide. Boron carbide is chosen for its high neutron absorption cross-section, its stability at high temperatures, and its lack of gamma ray production upon neutron absorption.

SS316L is used for the casing of the shutdown rod and the drive line. SS316L was chosen for its high strength at operating temperatures and low chemical interaction with the absorber material and other core materials. Because SS316L is not magnetic, a small plain steel disc is attached to the top of the drive line to allow the electromagnet to retrieve the shutdown rod.

2.4.5 Design evaluation of the shutdown rod system

To meet the required design bases the rods must have sufficient reactivity worth to achieve their shutdown function. They must also insert sufficiently quickly and reliably.

The worth of the shutdown rods is analyzed using the neutron transport code Serpent.

2.4.5.1 Reactivity worth

The design basis DB.SRS.01 requires that the shutdown rod system be capable of achieving shutdown with insertion of any single rod. To achieve this requirement, the worth of any single shutdown rod must account for the reactivity changes associated with temperature (as the core cools, reactivity feedbacks result in increased reactivity of the fuel). An additional margin of 500 pcm excess reactivity is incorporated to account for any uncertainties.

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Table 2-7: Shutdown rod worth spanning the range of temperature operating conditions

} Note that no significant reactivity worth is associated from moving from the hot zero power temperature condition of all core materials at 600 C to the hot full power temperature condition (with peak fuel temp at 640 C) due to the small increase in peak fuel temperature (only 40 C).

Design commitments are taken that each rod will have sufficient reactivity worth to meet this design basis. The programmatic controls used to ensure that the design commitments are met involve testing during startup to verify that the worth of each rod is sufficient to meet the commitments.

2.4.5.2 Shutdown rod insertion time

The safety analysis in Chapter 5.1 shows that the peak fuel temperature is not exceeded when modeling a rod insertion delay time of 10 seconds. This does not mean that 10 seconds is a limit, rather that this time meets the acceptance criteria in Chapter 5.

There are two steps that contribute to the shutdown rod insertion time. The first is the detection and signaling time, or the time it takes for the instrumentation and control system to detect that a reactor trip setpoint has been exceeded (or that a manual trip signal has been initiated) and to send a trip signal to the shutdown rod system. This step is the focus of DB.ICS.01 (see Section 2.7). The second step is the rod release and drop into the core. This step is the focus of DB.SRS.02 in this section.

Design commitments are made to limit the shutdown rod insertion time to 10 seconds or less, the insertion time assumed for the safety analysis. Specifically, commitments are made to limit the detection and signaling time to 6 seconds (DC.ICS.02.A) and to limit the rod release and drop time to 4 seconds (DC.SRS.02.A).

2.4.6 Summary of the shutdown rod system

Design basis:

DB.SRS.01 The shutdown rod system provides sufficient negative reactivity to achieve cold shutdown with insertion of one rod.

Design evaluation summary:

The analysis in this section has shown that the shutdown rods have sufficient reactivity worth to shut down the reactor using only a single rod. A design commitment is taken for this rod worth, and the appropriate programmatic controls are in place to verify that the commitment is met prior to beginning normal operation.

Design commitments and programmatic controls:

DC.SRS.01.A The worth of each shutdown rod will be greater than 1400 pcm, where 1400 pcm is greater than the total of: the reactivity worth associated with the temperature decrease from hot full power conditions to cold zero power conditions, and an additional margin of 500 pcm.

SUT.SRS.01.A1 and A2 (see Chapter 14)

Design basis:

DB.SRS.02 The shutdown rod system fully inserts the shutdown rods within a sufficient time after receiving a trip signal to prevent damage to the reactor.

Design evaluation summary:

This section describes the system responsible for shutdown rod insertion. This system receives a trip signal from the reactor trip system (see Section 2.6.3), the electromagnets de-energize, and then the rods drop by gravity into the core. The safety analysis in Chapter 5 showed that an assumed rod insertion time of 10 seconds upholds the safety goal of the Aurora in the event of the maximum credible accident. A design commitment is taken to a maximum time for each of the two components of the rod insertion time such that the total rod insertion time is less than 10 seconds, and the appropriate programmatic controls are in place to ensure that this limit is not exceeded.

Design commitments and programmatic controls:

DC.SRS.02.A The shutdown rod system fully inserts shutdown rods within 4 seconds of receiving a trip signal.

POT.SRS.02.A (see Chapter 14)

SUT.SRS.02.A

TS.LCO.01 (see Part IV)

(See also DC.ICS.02)

2.5 Reactor enclosure system

2.5.1 Introduction to the reactor enclosure system

The reactor enclosure system functions to enclose and provide structural support to the systems within the reactor module, and to maintain acceptable boundary integrity. The system consists of two main components: the capsule and the module shell. Collectively these two components contain: (1) the reactor system (see Section 2.2), (2) the control drum system (see Section 2.3), (3) the shutdown rod system (see Section 2.4), (4) the heat exchanger system (see Section 2.6) and (5) portions of the instrumentation and control system (see Section 2.7). The bulk of these systems are contained in the capsule, which is contained in the module shell. The module equipment housing is located on top of the module shell lid, and acts as an extension of the module shell. The module equipment housing contains portions of the control drum system and shutdown rod system.

2.5.2 Bases of the reactor enclosure system

2.5.2.1 Design basis of the reactor enclosure system

The design basis for the reactor enclosure system (RES) is as follows:

DB.RES.01 The reactor enclosure system provides a pathway to conduct heat away from the systems inside it and to reject it to the environment.

2.5.2.2 Performance bases of the reactor enclosure system

The reactor enclosure system is also designed to meet all of the following performance bases:

- The reactor enclosure system provides two passive physical barriers to fission product release: the capsule and the module shell.
- The reactor enclosure system supports, protects, and properly locates the reactor system, the shutdown rod system, the control drum system, the heat exchanger system, and portions of the instrumentation and control system.

2.5.3 Description of the reactor enclosure system

The capsule and module shell are both made of SS304. They do not serve as pressure boundaries. They operate at near-atmospheric pressure and they are backfilled with an inert gas. Each is a passive physical barrier to radiation release, but they are not credited to perform this function. Figure 2-21 shows a schematic of the reactor enclosure system, and each component is further described in this section.

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Figure 2-21: Reactor enclosure schematic

2.5.3.1 Capsule

The capsule is cylindrical with { } thick walls. It contains a thick SS316L base plate which is used to locate the cells and reflector blocks that make up the reactor core system and reflector system, as well as the in-core components of the shutdown rod and control drum systems. It also contains mounting structures for the heat exchanger system to properly support and locate the heat exchangers. The capsule is sealed with a gasket and { } thick lid, both made of SS304. The capsule lid contains penetrations for instrumentation pathways, heat exchanger piping, control drum drive shafts, and shutdown rod drive lines. These penetrations are sealed to restrict gas flow, and the capsule is backfilled with inert gas.

2.5.3.2 Module shell

The module shell is also a cylinder with { } thick walls. The capsule is entirely contained within the module shell, along with a layer of shielding between the two enclosures. This layer of shielding is part of the shielding system (see Section 2.2.4), and functions to reduce the fluence to both the module shell and the surrounding environment. Further discussion of the Radiation Protection Program can be found in Chapter 20. The shell is also sealed with a gasket and { } thick lid, both made of SS304. The shell lid contains penetrations for instrumentation pathways, heat exchanger piping, control drum drive shafts, and shutdown rod drive lines. These penetrations are sealed to restrict gas flow, and the shell is backfilled with inert gas.

The module equipment housing is another sealed volume that acts as an extension of the module shell. It is bolted on top of the module shell lid and contains the actuators for the control drum system and shutdown rod system. When the shutdown rods are withdrawn from the core the shutdown rod drive lines remain fully enclosed by the module equipment housing, protecting them from damage. The module equipment housing has penetrations for instrumentation pathways, and heat exchanger piping. These penetrations are sealed to restrict gas flow and the housing is backfilled with inert gas. The gas backfill is discussed further in 3.3.1.2.

2.5.4 Materials of the reactor enclosure system

All major components of the reactor enclosure system are constructed from SS304, except for the base plate at the bottom of the capsule, which is SS316L. Both of these alloys are durable, readily available, have an acceptable neutron scatter-to-capture cross-section, and possess high strength at operating temperatures.

2.5.5 Design evaluation of the reactor enclosure system

2.5.5.1 Passive conduction of decay heat

As described in the safety analysis in Chapter 5.1, the reactor does not rely on active cooling during the decay heat phase following shutdown during the maximum credible accident. The reactor passively conducts heat between the systems within the reactor module (the reactor core system, reflector system, shielding system, enclosure system, and heat exchanger system), distributing heat throughout the substantial thermal mass presented by the module. The heat is then removed from the module via natural convection of air at the surface of the module shell. The safety analysis shows that this passive heat removal is sufficient to maintain acceptable fuel temperatures and to meet the top-level safety goal of the Aurora in the event of the maximum credible accident.

Design commitments are taken for each system to ensure that it is properly configured to provide conduction between systems (including DC.RES.01.A), and a commitment is taken to test the ability of the module to passively remove heat at decay heat levels during startup testing (DC.RXS.04.B). In addition, a design commitment is made for the building system to ensure that the reactor emplacement supports passive cooling of the module shell (DC.BAS.01.A).

2.5.6 Summary of the reactor enclosure system

Design basis:

DB.RES.01 The reactor enclosure system provides a pathway to conduct heat away from the systems inside it and to reject it to the environment.

Design evaluation summary:

This section describes the layout of the reactor enclosure system, which consists of two cylindrical steel containers that surround the reactor system and the other major systems within the reactor module. Decay heat is conducted away from the reactor system and other surrounding systems to the reactor enclosure system. The transient analysis in Chapter 5 shows that, when configured as designed, the reactor enclosure system provides adequate heat conduction to maintain fuel temperatures below their required limits during the decay heat phase of the maximum credible accident (without active cooling). A design commitment is made to ensure proper as-built configuration of the reactor enclosure system prior to operation and the appropriate programmatic controls are in place to verify it.

Design commitments and programmatic controls:

DC.RES.01.A The critical components of the reactor enclosure system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.

POT.RES.01.A1 and A2 (see Chapter 14)

SUT.RES.01.A

2.6 Heat exchanger system

2.6.1 Introduction to the heat exchanger system

Heat is generated in the reactor core system (see Section 2.2.2) and transported by the heat pipes to the heat exchanger system. The heat exchanger system functions to transfer heat from the heat pipes to the power conversion system (see Section 2.8).

2.6.2 Bases of the heat exchanger system

2.6.2.1 Design basis of the heat exchanger system

The design basis for the heat exchanger system (HXS) is as follows:

DB.HXS.01 The heat exchanger system provides a pathway to conduct heat from the heat pipes of the reactor core system to the surrounding systems and ultimately to reject it to the environment.

2.6.2.2 Performance basis of the heat exchanger system

The heat exchanger system is also designed to meet the following performance basis:

- The heat exchanger system directs coolant to transfer heat from the heat pipes of the reactor core system to the power conversion system during normal operation.

2.6.3 Description of the heat exchanger system

The heat exchanger system is made up of six heat exchanger units, each servicing one sixth of the heat pipes in the reactor core system. Each heat exchanger unit is served by a cold leg and a hot leg from the header system. The units are installed by lowering them onto the heat pipes, which penetrate the heat exchanger units and extend beyond the top of the units.

Inside each heat exchanger unit, the power conversion system coolant removes heat from the heat pipes. During normal operation the primary function of the heat exchanger system is to ensure that each heat pipe is able to transfer heat to the power conversion system coolant. The detailed design of the heat exchanger units is outside the scope of this application, because the safety analysis shows it is not necessary to function in order to provide a conduction pathway for decay heat.

Low-enthalpy supercritical carbon dioxide (sCO_2) enters the heat exchanger unit via the cold leg and exits the heat exchanger via the hot leg at high enthalpy. Each group of six hot and cold legs combine into a single main leg, such that a single hot leg and a single cold leg exit the reactor enclosure to service the power conversion system. A schematic of the heat exchanger system and its associated piping, as well as the connection to the power conversion system can be seen in Figure 2-22.

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Figure 2-22: Schematic of the heat exchanger system

The heat exchanger units are optimized based on the particular power conversion system that is installed (see Section 2.8). For a representative sCO_2 system, with an approximate flowrate of 25 kg/s is sufficient to transfer the heat generated during full power operation. For the same representative system, the approximate average coolant flowrate per heat pipe is 0.22 kg/s and the nominal system pressure is on the order of 20 MPa. The maximum sCO_2 working fluid temperature is below 550 C.

2.6.4 Materials of the heat exchanger system

The heat exchanger system is constructed primarily of SS316. Stainless steel 316 was chosen because of its suitability at the operating temperature, pressure, and heat flux, as well as its compatibility with sCO_2, the power conversion system coolant. Compatibility between SS316 and sCO_2 has been shown to 550 °C, and SS316 is commonly used in supercritical CO_2 heat exchangers [30]. Both stainless steel and sCO_2 have relatively low cross-sections in the fast spectrum, and the heat exchanger system is exposed to a greatly reduced fluence compared to in-core materials.

2.6.5 Design evaluation of the heat exchanger system

2.6.5.1 Steady state operation

During steady state operation the heat exchanger system functions to remove heat from the reactor core system by convectively cooling the heat pipes. The system is optimized to account for the radial power peaking in the core and to remove the appropriate amount of heat from each heat pipe. This steady state heat removal function relates only to the performance basis of the system, so further details of operation during steady state are not presented.

2.6.5.2 Passive conduction of decay heat

As described in the safety analysis in Chapter 5.1, the Aurora reactor does not rely on active cooling during the decay heat phase following shutdown during the maximum credible accident. The reactor passively conducts heat between the systems within the reactor module (the reactor core system, reflector system, shielding system, enclosure system, and heat exchanger system), distributing heat throughout the substantial thermal mass presented by the module. The heat is then removed from the module via natural convection of air at the surface of the module shell. The safety analysis shows that this passive heat removal is sufficient to maintain acceptable fuel temperatures and meet the top-level safety goal of the Aurora in the event of the maximum credible accident.

Design commitments are taken for each system to ensure that it is properly configured to provide conduction between systems (including DC.HXS.01.A), and a commitment is taken to test the ability of the module to passively remove heat at decay heat levels during startup testing (DC.RXS.04.B). In addition, a design commitment is made for the building system to ensure that the reactor emplacement supports passive cooling of the module shell (DC.BAS.01.A).

2.6.6 Summary of the heat exchanger system

Design basis:

DB.HXS.01 The heat exchanger system provides a pathway to conduct heat from the heat pipes of the reactor core system to the surrounding systems and ultimately to reject it to the environment.

Design evaluation summary:

This section describes the design of the heat exchanger system, which is made up of six heat exchanger units. Decay heat is conducted away from the heat pipes of the reactor core system and outward toward surrounding systems through the heat exchanger system. The transient analysis in Chapter 5 shows that, when configured as designed, the heat exchanger system provides adequate heat conduction to maintain fuel temperatures below their required limits during the decay heat phase of the maximum credible accident (without active cooling). A design commitment is made to ensure proper as-built configuration of the heat exchanger system prior to operation and the appropriate programmatic controls are in place to verify it.

Design commitments and programmatic controls:

DC.HXS.01.A The critical components of the heat exchanger system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.

SUT.HXS.01.A (see Chapter 14)

2.7 Instrumentation and control system

2.7.1 Summary description

The instrumentation and control system includes the components and systems required to monitor and control the Aurora reactor. These systems are used for control of the plant and include the following subsystems:

- Reactor trip system (Section 2.7.3)
- Reactivity management system
- Plant control system
- Information display system

Each of the instrumentation and control subsystems rely on a common set of sensors and other components. These components are described in Section 2.7.2. Because Section 2.7.2 describes components common to multiple subsystems, it is not structured in the same way as the other system descriptions in this chapter. Design bases related to these components are contained in the subsystem descriptions for the subsystems that rely on the components.

The reactor trip system is the only subsystem credited in the safety analysis in Chapter 5, and therefore contains the only design bases for the instrumentation and control system. The reactor trip system is described in detail in Section 2.7.3. The other subsystems of the instrumentation and control system are briefly summarized in Section 2.7.4.

2.7.2 Instrumentation and control system components

This section describes components that are common to multiple subsystems of the instrumentation and control system. It is not an exhaustive description of the instrumentation and control system components, but it describes all of the components that are relied on to meet design bases of the reactor trip system, as described in Section 2.7.3.

2.7.2.1 Definitions

The following terms are specific to the Aurora and are used throughout this section.

channel: A channel is the combination of components including a sensor, lines, amplifiers, output devices, and a limit monitor which are connected for the purpose of measuring the value of a parameter and enforcing operational limits.

direct temperature measurement: A temperature channel with a sensor on the heat pipe, located in the specific reactor cell that is the subject of the temperature measurement. The direct temperature channel provides a direct temperature measurement of the reactor cell being measured.

failed sensor: A sensor with a sensing element that has stopped measuring the intended process variable. A thermocouple, for example, becomes a failed sensor when the thermocouple junction electrically separates and creates an open-circuit.

fault signal: A fail-safe digital signal to the control logic indicating a process variable has exceeded a limit setpoint, power to the limit monitor or sensor was lost, or the sensor was disconnected.

indirect temperature measurement: A temperature channel with a sensor on a heat pipe located in a neighboring reactor cell to the specific reactor cell that is the subject of a temperature measurement. The indirect temperature channel provides an indirect temperature measurement of the reactor cell being measured and can be used in conjunction with direct temperature channels to evaluate the temperature of the cell.

limit: A quantity and a designation that defines a maximum or minimum allowed value for a parameter. A limit can be “exceeded” when a parameter is above the maximum allowed value, or below the minimum allowed value.

limit setpoint: A process limit monitor configuration that specifies a maximum or minimum limit value.

operating limits: A range of allowed values for a parameter bounded by a maximum limit and a minimum limit. If only a maximum limit is defined for a parameter, then values less than or equal to the maximum limit are allowed. If only a minimum limit is defined for a parameter, then values greater than or equal to the minimum limit are allowed.

power operation: State when the reactor is at operating temperature, the reactor power is generally at steady-state but may be increased or decreased, and significant power is being produced.

process limit monitor: A device that measures an analog signal from a sensor, compares the measured value to one or more limit setpoints, and sets the state of one or more digital outputs based on the result of the comparison(s).

process variable: The measured value of a particular part of a process.

reactor trip signal: A fail-safe digital output from a reactor trip circuit to the shutdown rod system that causes the shutdown rods to be released.

rod insertion time: The elapsed time between the initiation of a reactor trip and the instant the shutdown rod reaches its fully inserted position. Rods are inserted into the core by gravity.

startup: State when the reactor temperature is being increased, and the reactor power is being increased, but the reactor is not producing significant power.

2.7.2.2 Introduction

Sensors are located as needed to measure specific process variables. Heat pipe temperature sensors, neutron flux detectors, and control drum position sensors are located in the reactor module. Process limit monitors, control logic, motor controllers, and relay logic are contained in the instrumentation enclosures and control enclosures.

Redundant sensors are included to reduce maintenance activities by obviating the need for immediate replacement of failed sensors during operation. For example, the temperature of each heat pipe is measured by three thermocouple junctions and the reactor can continue operating with one failed thermocouple junction at a particular heat pipe.

Enclosures are the physical containers that contain the components for the instrumentation and control system. There are three independent instrumentation enclosures and two independent control enclosures, all located on the first floor of the powerhouse. Specifically, both of these enclosures are located in the same room as the power conversion system. The enclosures provide for physical separation of the instrumentation and control components as well as other functions needed for the continuous operation of the plant.

The reactor trip system, reactivity management system, plant control system, and the information display system all utilize data measured by the sensors and utilize components in the instrumentation and control enclosures. The reactor trip system includes a hard-wired fail-safe circuit that does not rely on digital computers or custom software for reactor trips. The reactivity management system, plant control system, and information display system rely on stored data, custom software, and computational resources which are located in the computational enclosure (which is part of the plant control system).

Figure 2-23 provides a simplified overview of the instrumentation and control system that identifies major components and interconnects.

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Figure 2-23: Instrumentation and control system block diagram overview

2.7.2.3 Instrumentation enclosures

Three functionally identical instrumentation enclosures provide redundancy. The analog signals from the redundant sensors are routed to one of the three instrumentation enclosures. The instrumentation enclosures include signal conditioning, process variable limit monitors, and analog-to-digital conversion for each of the analog signals. The limit monitors provide independent fault signals to each of the control enclosures for each of the monitored process variables. The analog signals are also digitized, and the data is provided to a computer in the computational enclosure for analysis, recording, and display.

2.7.2.4 Control enclosures

Two functionally identical and independent control enclosures are included for redundancy. The control enclosures are used to aggregate fault signals to generate a reactor trip signal, control the shutdown rod release mechanisms, control the shutdown rod withdrawal and control drum motor drivers, receive analog feedback from sensors monitoring the actuators, and to control the power conversion system.

The control enclosures independently aggregate the fault signals received from the three instrumentation racks to determine the reactor trip state. Aggregation is performed to allow the reactor to continue operating, for example, with one failed thermocouple junction at a heat pipe when the other two thermocouple junctions at that heat pipe are measuring temperatures that are within operating limits. The aggregation is performed with discrete logic components.

The control enclosures receive analog feedback from drum position sensors and shutdown rod position sensors. Duplicate actuator sensor feedback is included to prevent a failure in one control enclosure from corrupting the sensor data in the other control enclosure. The control enclosures include motor drivers to interface with the control drum system motors and the shutdown rod system motors.

The two control enclosures independently aggregate the independent fault signals to generate a reactor trip signal from each control enclosure. The two reactor trip signals are combined such that if either control enclosure signals a shutdown, a reactor shutdown occurs. Only one of the control enclosures actively controls components such as the control drum system motors, shutdown rod system withdrawal motors, and the power conversion system at a time. A switch is used to determine which control enclosure is actively controlling these components.

Two control enclosures are included for redundancy and both control enclosures are typically operational. This redundancy allows the reactor system to continue operating, for example, if one of the control enclosures needs to be temporarily bypassed for maintenance.

2.7.2.5 Process variable limit monitors

Process variable limits are monitored by commercially available limit monitors installed in the instrumentation and control enclosures. Each limit monitor is capable of monitoring one process variable and enforcing an upper limit and a lower limit. Each limit monitor has several normally-open mechanical relay contacts. The coils for the normally-open contacts are energized when the monitored process variable is within the operating limits. If the process variable exceeds the operating limits, the associated coil is de-energized causing a normally-open relay contact in the limit monitor to open, sending a fault signal to the control enclosure. A user must physically interact with the limit monitor to configure the limit setpoint(s). These limit monitors are normally locked out and not normally accessible to onsite personnel. The limit monitors are externally powered. If external power to a limit monitor is lost, the normally-open relay contacts in the limit monitor open and send a fault signal to the control enclosure. Limit monitors are configured to send a fault signal when the input sensor is disconnected from the limit monitor. Limit monitors are configured to detect some sensor failure modes and to send a fault signal when a failed sensor is detected.

2.7.2.6 Manual reactor trip buttons

Manual reactor trip buttons are hard-wired in the reactor trip circuit to provide a user-operable reactor shutdown. The signal from the reactor trip buttons directly results in a reactor shutdown. Manual reactor trip buttons are installed in several locations in the powerhouse.

2.7.2.7 Reactor instrumentation

2.7.2.7.1 Neutron flux detectors

Neutron flux in the reactor core is monitored by wide range fission chambers located on the periphery of the reactor core. These fission chambers are operated in different modes to measure several decades of neutron flux and span flux levels from startup to normal power operation. Neutron flux detectors provide continuous measurement of core parameters such as power and period.

The neutron flux detectors are located in the zirconium reflector on the periphery of the reactor core in the three corners that are not occupied by control drums. The detectors and associated wiring are located in a protected chamber with a zirconium plug directly above that can be removed for inspection and maintenance.

2.7.2.7.2 Heat pipe temperature sensors

The temperature limits in the Aurora design are based on fuel temperature. Because heat pipes are nearly isothermal in operation, fuel temperatures can be inferred from heat pipe temperatures. Each heat pipe is instrumented with three thermocouples to provide redundancy. The thermocouples are located above the top of the reactor core in the heat exchanger region to reduce exposure to radiation.

2.7.2.7.3 Control drum absolute position sensors

2.3Position sensors are used to monitor the absolute position of each control drum. The control drum position sensors are located in the module equipment housing.

2.7.3 Reactor trip system

2.7.3.1 Introduction to the reactor trip system

The purpose of the reactor trip system is to trigger a reactor shutdown to protect the personnel, reactor, and facility. Reactor trip signals can be generated automatically, in response to the detection of abnormal plant operating conditions, or manually by onsite personnel. Signals from multiple sources are aggregated by the control logic and, when appropriate, a reactor trip signal is sent to the shutdown rod system (see Section 2.4), resulting in the insertion of the shutdown rods into the reactor core.

2.7.3.2 Bases of the reactor trip system

2.7.3.2.1 Design bases of the reactor trip system

The design bases for the reactor trip system, a subsystem of the instrumentation and control system (ICS), are as follows:

DB.ICS.01 The reactor trip system monitors reactor process variables and sends a reactor trip signal when a process variable exceeds a limit setpoint.

DB.ICS.02 The reactor trip system sends a reactor trip signal to the shutdown rod system within a sufficient time of exceeding a limit to prevent damage to the reactor.

DB.ICS.03 The reactor trip system provides the means for a reactor trip signal to be sent manually.

DB.ICS.04 The reactor trip system requires deliberate action to reset a reactor trip signal and return the system to normal operation.

DB.ICS.05 The reactor trip system is protected against unauthorized configuration changes.

DB.ICS.06 The reactor trip system is fail-safe.

2.7.3.3 Performance bases of the reactor trip system

The reactor trip system is also designed to meet all of the following performance bases:

- The reactor trip system allows periodic in-service testing when the reactor is in operation.
- The reactor trip system can receive a trip signal from the plant control system.

2.7.3.4 Description of the reactor trip system

The reactor trip system aggregates fault signals and reactor trip signals to trigger a reactor shutdown. Fault signals and reactor trip signals can be initiated automatically when an operating limit for a process variable is exceeded or manually when personnel press a shutdown button. Fault signals from multiple sources are aggregated by the control logic to determine if a reactor trip signal is sent.

Process variables are monitored by sensors inside and outside the reactor module. Operating limits for process variables are defined to protect the reactor and equipment. Each operating limit is enforced by a limit monitor or sensor that sends a fault signal to the control logic when the process variable exceeds the defined operating limits. A fault signal is sent when the operating limits of any of the following process variables are exceeded:

- Reactor over-temperature
- Reactor under-temperature
- Reactor over-power
- Reactor period too short

The steady state fuel temperature limits are set such that the safety limit is not challenged by the increase in temperature that would result from the maximum credible accident or any other off-normal event discussed in Chapter 5. The reactor trip system thermocouples do not directly measure fuel temperature, rather they infer the fuel temperature from measurements of the heat pipe temperature in the heat exchanger region. Fault signals generated by these temperature measurements are described in Section 2.7.3.4.2.1.

The other operating limits listed are not relied on in the safety analysis but are included for investment protection purposes and defense-in-depth. Fault signals generated for reactor over-power and reactor period are described in Section 2.7.3.4.2.4 and Section 2.7.3.4.2.5, respectively.

Redundant sensors allow the reactor to continue operating with some failed sensors. Fault signals from redundant sensors are aggregated with discrete logic to allow the reactor to continue operating as long as the required number of functional sensors are within the operating limits for the process variable. The limit monitors are configured to detect values outside of operating limits, disconnected sensors, and some sensor failures and automatically send a fault signal.

2.7.3.4.1 Reactor trip circuit

The reactor trip circuit is a hard-wired fail-safe circuit that does not rely on digital computers or custom software. Devices that can initiate a reactor trip are connected in series in the reactor trip logic circuit. If none of the devices that can initiate a trip are in the trip state, the output of the trip logic is high, the shutdown rod electromagnets remain energized, and the reactor can operate. If one or more of the devices in the trip logic are in the trip state, the output of the trip logic is low, a reactor trip signal is sent, the shutdown rod electromagnets de-energize, and the reactor is shut down. Figure 2-24 shows a simplified overview of the reactor trip circuit.

During startup and power operation, two redundant reactor trip circuits must be functional, and each reactor trip circuit is independently able to shut down the reactor. The results of the redundant reactor trip circuits are combined such that if either trip circuit signals a shutdown, a reactor shutdown occurs. Each reactor trip circuit includes a bypass switch that can be enabled to maintain the output of the bypassed reactor trip circuit in the high state. The reactor trip circuit that is not bypassed functions normally. Enabling both reactor trip circuit bypass switches at the same time causes a reactor trip. Enabling a bypass switch starts a count-down timer that trips the reactor if the bypass is not removed in the required time.

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Figure 2-24: Simplified reactor trip circuit overview

During startup and power operation, one reactor trip circuit may be bypassed, and therefore be nonfunctional, for up to 7 days. If a reactor trip circuit bypass is not removed within 7 days, the reactor trip system automatically initiates a reactor trip within 1 minute. If one reactor trip circuit is nonfunctional and is not bypassed, the reactor trip system automatically initiates a reactor trip within 5 seconds. If two reactor trip circuits are nonfunctional or bypassed, the reactor trip system automatically initiates a reactor trip within 5 seconds.

2.7.3.4.2 Fault signals and reactor trip signals

Fault signals generated by process variable limit monitors and other devices are sent to the control logic. Based on sensor redundancy and the aggregation logic, multiple fault signals may be sent without generating a reactor trip signal. A reactor trip signal causes a reactor shutdown. Signals from some devices, such as manual reactor trip buttons, are not aggregated and directly cause a reactor trip. The functionality of the individual fault signals and reactor trip signals that are used during normal operation is described in this section. Limit setpoints used by specific channels are shown in Table 2-8, and described in more detail in the corresponding sub-sections for each channel type.

Table 2-8: Reactor trip system limit setpoints

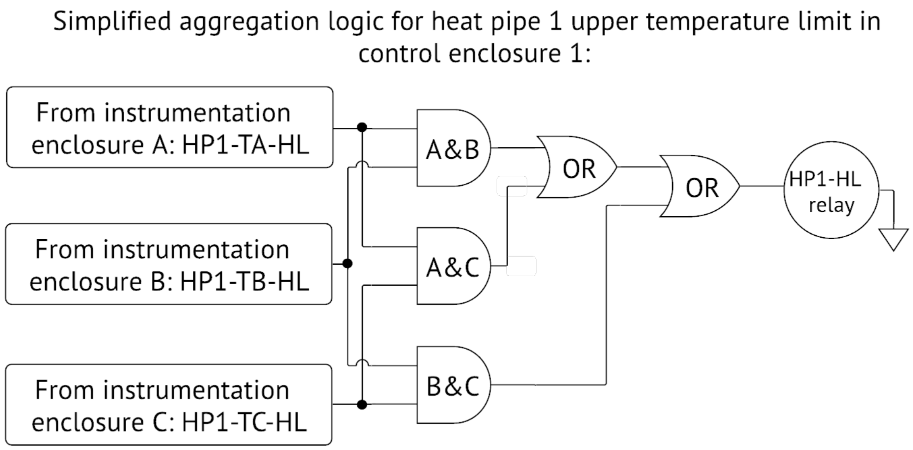
Channel	Setpoint type	Setpoint value	Number of channels monitored
Heat pipe temperature channel	upper limit (Δ from nominal)	+15 C	342
Heat pipe temperature channel	lower limit (Δ from nominal)	-35 C	342
Reactor thermal power	upper limit	4.2 MW	2
Reactor period	lower limit	5 s	2

2.7.3.4.2.1 Heat pipe temperature fault signal

Upper and lower limits are established for heat pipe temperatures at the measured locations, as shown in Table 2-8. The upper and lower limits establish the operating limits for the reactor cell heat pipe temperature channels, and each reactor cell has limits that are defined based on its nominal temperature at steady state full power. The upper limits are set at a delta of +15 C above each nominal temperature channel value at steady state full power. The lower limits are set at a delta of 35 C below each nominal temperature channel value at steady state full power. Reactor cell heat pipe temperature channel operating limits are dependent on reactor state. During power operation, the upper and lower temperature limits are enforced. During startup, the upper temperature limits are enforced but the lower temperature limits are not enforced.

Three independent thermocouple junctions are used to measure the temperature of each heat pipe in approximately the same location. Independent process limit monitors for each thermocouple can send fault signals to the aggregation logic based on the measured temperature. Measured temperatures may be outside of operating limits for 2 seconds before the process limit monitor sends a fault signal to the aggregation logic.

The fault signals are aggregated such that at least one of the following criteria must be met for each reactor cell heat pipe: (1) two or more direct temperature channels shall not be sending a fault signal (schematic shown in Figure 2-25), or (2) one direct temperature channel and nine or more indirect temperature channels for a heat pipe with two failed sensors shall not be sending a fault signal. If any of the reactor cell heat pipe temperature channels do not meet these criteria, the reactor trip system automatically initiates a reactor trip within 5 seconds.



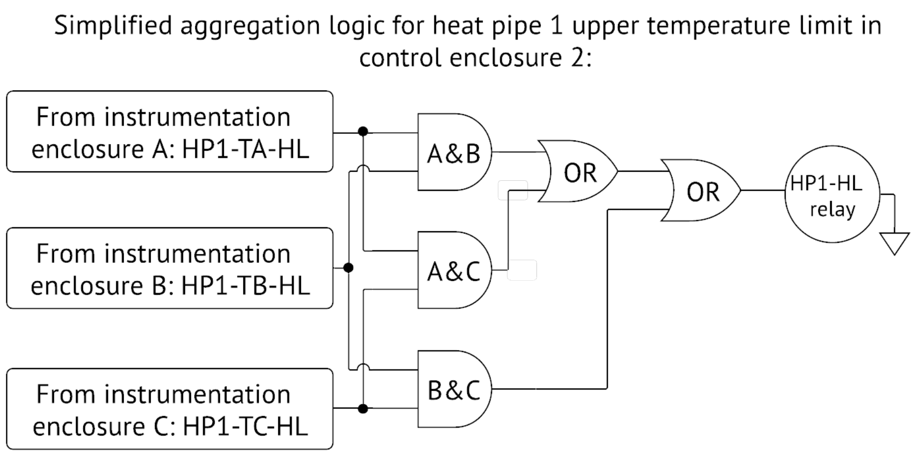


Figure 2-25: Simplified heat pipe fault signal aggregation logic

2.7.3.4.2.2 Manual reactor trip signal

Reactor trip signals can be initiated manually with onsite reactor trip buttons. Onsite manual reactor trip signals are not aggregated by control logic and directly cause a reactor shutdown.

2.7.3.4.2.3 Time delay AC power loss reactor trip signal

The control enclosures are powered by independent electrical circuits from a power distribution panel. The control enclosures also include independent battery backup power. If the power from the distribution panel to a control enclosure is interrupted, the battery backup will keep the affected enclosure operational. If the power to a control enclosure is interrupted, a count-down timer in a time delay relay is activated and will send a reactor trip signal if power is not restored to the control enclosure prior to the expiration of the count-down timer. The affected control enclosure continues to operate using the backup battery after the reactor trip to monitor the status of the reactor.

2.7.3.4.2.4 Over-power reactor trip signal

The neutron flux detectors are calibrated to measure reactor thermal power. As shown in Table 2-8, the limit setpoint for reactor over-power is 4.2 MWth, or 5% over the licensed thermal power. If the reactor power exceeds the limit setpoint, the reactor trip system automatically initiates a reactor trip within 5 seconds. The over-power limit is always enforced.

The reactor over-power trip relies on one or more functional reactor thermal power channels. The reactor thermal power channels must be calibrated on an appropriate interval. During startup and power operation, if zero reactor thermal power channels are functional for 5 minutes, the reactor will automatically shut down.

The reactor must not be intentionally operated at a thermal power greater than the licensed thermal power. Although the reactor trip system limit setpoints prevent the reactor from exceeding the licensed thermal power by more than 5%, additional measures are in place to prevent any operation in excess of the licensed thermal power. The plant control system actively controls plant parameters to produce power up to the licensed thermal power. If the reactor power exceeds the licensed thermal power, the plant control system automatically takes action to reduce the thermal power [31][32]. If the reactor power has not been reduced below the licensed thermal power within 60 minutes, the plant control system initiates a reactor trip within 1 minute.

2.7.3.4.2.5 Reactor period reactor trip signal

The reactor is not be operated with a period shorter than 5 seconds, which is consistent with the safety analysis of Chapter 5. If the reactor period is less than this limit, the reactor trip system automatically initiates a reactor trip within 5 seconds.

The reactor period trip relies on one or more functional reactor period channels. During startup, if zero reactor period channels are functional for 30 seconds, the reactor will automatically shut down. During power operation, if zero reactor period channels are functional for 5 minutes, the reactor will automatically shut down.

2.7.3.4.3 Anticipated user actions

There are several manual actions that can be performed by a user. Manual user actions are minimal during normal operation and most manual user actions are expected to occur outside of normal operation.

2.7.3.4.3.1 User actions during normal operation

Manual reactor trip buttons, as described in Section 2.7.2.6, allow the user to press a button to immediately trigger a reactor shutdown. Users should rarely need to perform this action.

2.7.3.4.3.2 User actions outside of normal operation

When the reactor is shutdown, two control system features will prevent the reactor from restarting. First, the lower operational temperature limits and the lower operational power limits will prevent shutdown rod removal when the reactor is shutdown. To start the reactor, the control system needs to be put into a startup mode to bypass the lower operating limits. Second, after a reactor trip, the shutdown logic latches in the trip state. After the condition that initiated the trip has been resolved, a user must unlatch the reactor trip by pressing a button.

If one of the two control enclosures require maintenance, the enclosure needing maintenance must be bypassed before the maintenance activities are started. A manual switch is provided for the user to bypass one control enclosure at a time.

The upper limits for the control drum angular positions must be periodically updated to allow additional reactivity to be added (see Section 2.3). Changing limit setpoints requires a user to manually interact with the appropriate limit monitor and is not possible during normal operations. The control enclosure containing the limit monitor should be bypassed prior to changing the limit setpoint.

2.7.3.5 Materials of the reactor trip system

The materials and components in the reactor trip system are chosen to withstand normal and abnormal conditions. Materials and components are also chosen to limit the quantity of flammable material and to meet applicable fire testing standards.

2.7.3.6 Design evaluation of the reactor trip system

To meet the required design bases the reactor trip system must automatically detect conditions requiring a reactor trip and send a reactor trip signal to the shutdown rod system. It must accomplish both the detection and signaling sufficiently quickly to prevent damage to the reactor. It must also allow for a manual triggering of a reactor trip signal. Finally, it must be robust against failure and ensure that reactor trip signals require deliberate manual action to reverse.

2.7.3.6.1 Automatic detection of conditions requiring trip

The reactor trip system must reliably detect and respond to the exceedance of limit setpoints that challenge the safety goals of the system (DB.ICS.01). The reactor over-temperature limit is the only condition that requires a trip to meet the safety goals, as shown in the transient analysis in Chapter 5. The transient analysis further shows that the setpoints chosen are sufficiently conservative to prevent damage to the reactor, provided the resulting reactor trip occurs within the assumed time interval as described in Section 2.7.3.6.2.

For defense-in-depth purposes and investment protection, a variety of other conditions will result in a reactor trip when detected. An extended list of conditions that result in an automatic trip signal can be found in Section 2.7.3.4.2.

Multiple steps must be taken to ensure the reliable detection of conditions requiring a reactor trip, including all of the following:

- Sensors must be installed in the correct locations
- Process limit monitors must be configured with the correct sensor scaling (or calibration) values and limit setpoints
- Sensors must be connected to the correct process limit monitors, and the process limit monitors must be connected to the reactor trip logic
- The process limit monitors send reactor trip signals

Design commitments are made to ensure that each of these steps have the appropriate programmatic controls in place to ensure the reactor trip system automatically detects and responds to conditions requiring a reactor trip (DC.ICS.01.A-D).

2.7.3.6.2 Trip signaling time

The safety analysis in Chapter 5.1 shows that the peak fuel temperature is not exceeded when modeling a rod insertion delay time of 10 seconds. This does not mean that 10 seconds is a limit, rather that this time has been demonstrated to be acceptable.

There are two steps that contribute to the shutdown rod insertion time. The first is the detection and signaling time, or the time it takes for the reactor trip system to detect that a reactor trip setpoint has been exceeded (or that a manual trip signal has been initiated) and to send a trip signal to the shutdown rod system. This step is the focus of DB.ICS.02. The second is the rod release and drop into the core, which is the focus of the shutdown rod system design basis DB.SRS.02 (see Section 2.4).

Design commitments are made to limit the shutdown rod insertion time to 10 seconds or less, the insertion time assumed for the safety analysis. Specifically, commitments are made to limit the detection and signaling time to 6 seconds (DC.ICS.02.A) and to limit the rod release and drop time to 4 seconds (DC.SRS.02.A).

2.7.3.6.3 Manual reactor trip

The ability to send a manual reactor trip signal is provided so that onsite personnel can respond to events that do not trigger an automatic shutdown.

2.7.3.6.4 Protection of reactor trip function

The reactor trip system is designed to be robust against numerous failure modes. DB.ICS.04 makes a commitment to ensuring that once a reactor trip signal is sent, the tripped state is latched in, so that deliberate user action is required to reset the tripped state as described in 2.7.3.4.3.2.

DB.ICS.05 makes commitments to ensure that no unauthorized configuration changes are made to the reactor trip system to ensure that no changes to the system prevent it from achieving the functions described in DB.ICS.01-03. To prevent remote access, the reactor trip system uses no digital computers or custom software, and it is isolated from computer networks. To limit physical access, the system is located in an access-controlled area and configurable components are password protected. Access controls are discussed in the Physical Security Plan.

The reactor trip system is configured to be fail-safe, such that no malfunction within the system, caused solely by the variations of external conditions within the ranges in the design basis, will result in unsafe failure (DB.ICS.06). Loss of AC power to the control cabinets activates a time-delay relay that sends a reactor trip signal if power is not restored promptly, as described in Section 2.7.3.4.2.3. The failure of a power supply in an instrumentation cabinet also results in a reactor trip signal. Additionally, disconnected sensors and most sensor failures result in fault signals, which are aggregated by control logic as described in Section 2.7.3.4, and result in a reactor trip signal if the required minimum redundancy of functional sensors is not maintained.

2.7.3.7 *Summary of the reactor trip system*

Design basis:

DB.ICS.01 The reactor trip system monitors reactor process variables and sends a reactor trip signal when a process variable exceeds a limit setpoint.

Design evaluation summary:

This section describes the design of the reactor trip system, which provides the ability to detect and respond to multiple trip conditions. The transient analysis in Chapter 5 shows that if reactor trip signals are sent in response to the chosen setpoints, and the shutdown rods insert within the appropriate time interval, then fuel temperatures will be maintained below the required limits. Design commitments are made to ensure that each of the trip conditions will be reliably detected, and will result in a reactor trip signal, and the appropriate programmatic controls are in place to verify it.

Design commitments and programmatic controls:

DC.ICS.01.A The reactor trip system sensors are installed in the correct locations.

POT.ICS.01.A1 and A2 (see Chapter 14)

SUT.ICS.01.A1

DC.ICS.01.B The reactor trip system process limit monitors are connected to the correct locations, and are configured with the correct sensor scaling information and limit setpoints.

POT.ICS.01.B1 and B2

TS.LCO.02 (see Part IV)

DC.ICS.01.C The reactor trip system sensors are connected to the correct process limit monitors.

POT.ICS.01.C1 and C2

SUT.ICS.01.C

DC.ICS.01.D The reactor trip system process limit monitors send a fault signal when a process variable exceeds a limit.

POT.ICS.01.D

TS.LCO.02

Design basis:

DB.ICS.02 The reactor trip system sends a reactor trip signal to the shutdown rod system within a sufficient time of exceeding a limit to prevent damage to the reactor.

Design evaluation summary:

This section describes the design of the reactor trip system, which provides the ability to detect and respond to multiple trip conditions. The transient analysis in Chapter 5 shows that if reactor trip signals are sent in response to the chosen setpoints, and the shutdown rods insert within the assumed time interval, fuel temperatures will be maintained below the required limits. A design commitment is made to ensure that the trip signal is sent sufficiently quickly, and the appropriate programmatic controls are in place to verify it.

Design commitments and programmatic controls:

DC.ICS.02.A The reactor trip system detects the exceedance of a limit setpoint and sends a reactor trip signal within 6 seconds.

POT.ICS.02.A (see Chapter 14)

TS.LCO.02 (see Part IV)

(See also DC.SRS.02)

Design basis:

DB.ICS.03 The reactor trip system provides the means for a reactor trip signal to be sent manually.

Design evaluation summary:

This section describes the design of the reactor trip system, which provides the ability to automatically detect and respond to multiple trip conditions. The system is also designed to provide a means for manually initiating a reactor trip. A design commitment is made to ensure that the manual reactor trip buttons send a reactor trip signal, and the appropriate programmatic controls are in place to verify it.

Design commitments and programmatic controls:

DC.ICS.03.A Manual reactor trip buttons send a reactor trip signal when pushed.

POT.ICS.03.A (see Chapter 14)

TS.LCO.02 (see Part IV)

Design basis:

DB.ICS.04 The reactor trip system requires deliberate action to reverse a reactor trip signal and return the system to normal operation.

Design evaluation summary:

This section describes the design of the reactor trip system, which provides the ability to automatically detect and respond to multiple trip conditions and provide a means for manually initiating a reactor trip. As described, once a reactor trip signal is sent, the reactor trip is latched in and cannot be reversed without manual action. This ensures that reactor shutdown occurs so that the condition that caused the reactor trip can be addressed. A design commitment is made to ensure that the reactor trip system is latched in the tripped state following a reactor trip signal, and the appropriate programmatic controls are in place to verify it.

Design commitments and programmatic controls:

DC.ICS.04.A A reactor trip signal causes the reactor trip system to latch in the tripped state. After the condition that caused the reactor trip has been resolved, a control must be toggled to reset the trip system from the tripped state.

POT.ICS.04.A (see Chapter 14)

TS.LCO.02 (see Part IV)

Design basis:

DB.ICS.05 The reactor trip system is protected against unauthorized configuration changes.

Design evaluation summary:

This section describes the design of the reactor trip system, which provides the ability to automatically detect and respond to multiple trip conditions and provide a means for manually initiating a reactor trip. This section also describes how the system is designed to prevent unauthorized configuration changes, through component selection, isolation, and other protections, and the appropriate programmatic controls are in place to verify it.

Design commitments and programmatic controls:

DC.ICS.05.A The reactor trip system does not use any digital computers or custom software.

POT.ICS.05.A (see Chapter 14)

DC.ICS.05.B The reactor trip system is isolated from computer networks to prevent changes to limit setpoints, scaling information, or other configuration by unauthorized personnel.

POT.ICS.05.B

DC.ICS.05.C The process limit monitors are installed in an access-controlled area to prevent changes to limit setpoints, scaling information, or other configuration by unauthorized personnel.

POT.ICS.05.C

DC.ICS.05.D The process limit monitors are password protected to prevent changes to limit setpoints, scaling information, or other configuration by unauthorized personnel.

POT.ICS.05.D

Design basis:

DB.ICS.06 The reactor trip system is fail-safe.

Design evaluation summary:

This section describes the design of the reactor trip system, which provides the ability to automatically detect and respond to multiple trip conditions and provide a means for manually initiating a reactor trip. As described, the system is designed such that failures are detected and result in a safe condition. Loss of power to control cabinets results in a reactor trip signal, and the disconnection of individual sensors results in fault signals. The reactor trip logic aggregates these fault signals and sends a reactor trip signal if the required minimum redundancy in sensors is no longer met. A design commitment is made to ensure that these failures result in safe conditions, and the appropriate programmatic controls are in place to verify it.

Design commitments and programmatic controls:

DC.ICS.06.A Loss of AC power to one or both control cabinets activates a time-delay that results in a reactor trip signal if power is not restored within five minutes.

POT.ICS.06.A (see Chapter 14)

DC.ICS.06.B Loss of DC power to the reactor trip circuit or the aggregation logic in one or both control cabinets causes a reactor trip signal.

POT.ICS.06.B

DC.ICS.06.C Detection of a disconnected sensor causes the associated process limit monitor to send a fault signal.

POT.ICS.06.C

DC.ICS.06.D Redundant reactor trip system logic is installed in separate fire areas to prevent fire-induced failure of the reactor trip system.

POT.ICS.06.D

2.7.4 Other instrumentation and control systems

The reactor trip system is designed to protect against spurious operation or failure of the other instrumentation and control systems from harming the reactor. The other instrumentation and control systems have no design bases and are therefore only described briefly.

2.7.4.1 Reactivity management system

The reactivity management system monitors performance parameters in the reactor and can adjust core reactivity. The primary function of the reactivity management system is to control rotation of the three control drums (see Section 2.3) to maintain core criticality. During normal operation, actuation of the control drums is controlled automatically based on measurements of reactor parameters.

During normal operation, the reactivity management system automatically maintains core criticality. The automatic control components of the reactivity management system take inputs of several variables including neutron flux, temperature, and control drum position. These measurements are used to determine the signals that are sent to the control drum actuators to maintain core criticality.

2.7.4.2 Plant control system

The plant control system performs plant-wide process monitoring and control, including plant automation and alarm indication. Alarms are displayed onsite in the monitoring room and may be displayed in other offsite locations.

2.7.4.2.1 Startup and shutdown capability

The plant control system cannot over-ride or prevent the operation of the upper limit enforcement in the automatic reactor trip system or change process limit monitor setpoints. The plant control system can bypass lower-limit enforcement for the purpose of reactor startup. The plant control system cannot startup the reactor without a deliberate user action to initiate the startup. The user controls to initiate the startup are part of the plant control system.

The plant control system can initiate automatic reactor trips that provide defense-in-depth to the reactor trips initiated by the reactor trip system. In addition to other trips not described here that are used for investment protection, the following types of reactor trips are implemented by the plant control system:

- Secondary loop trips
- Shutdown rod insertion time trips

Secondary loop trips are used to ensure that the power conversion system is providing a sufficient heat sink for the reactor. They are initiated by pressure switches and thermocouples in the secondary loop, and by the power conversion system. These trips are for defense-in-depth and investment protection purposes because the reactor trip system would already initiate a reactor trip on over-temperature as described in Section 2.7.3.4.2.1 in the case of a loss of heat sink. The safety analysis in Chapter 5.1 credits only the over-temperature trip.

Shutdown rod insertion time trips are used to ensure the reactor is in compliance with the conditions analyzed in Chapter 5. If the plant control system detects that one shutdown rod exceeds the allowed insertion time, a 90 day countdown timer is automatically started. The reactor can be restarted, and power operation can continue until the 90 day countdown timer expires, at which time an automatic reactor trip is initiated. If the shutdown rod system is restored to meet the allowed insertion time within the 90 day limit, the countdown timer can be stopped. If the rod insertion time for two shutdown rods exceeds the allowed rod insertion time, the plant control system disables the ability to restart the reactor. After the shutdown rods are repaired, the ability to restart the reactor can be reenabled for the purpose of testing the rod insertion times. Routine power operation cannot resume until the shutdown rod insertion time is restored to meet the allowed insertion time.

2.7.4.2.2 Monitoring and recording capability

The plant control system monitors plant-wide process variables and stores the data for data retention, analysis, use by the reactivity management system, and use by the information display system. The plant-wide process data is also used to trigger audible and visual alarms. Data collected from components outside of the reactor include area radiation monitor data, secondary loop data, power conversion system status information, facility information, and grid demand. The plant control system monitors the status of many components in the instrumentation and control enclosures, including the status of the limit monitor digital outputs, individual logic aggregators, DC power supplies, and uninterruptable power supplies. The plant control system monitors heat pipe thermocouple data, reactor power data, and reactor period data.

2.7.4.2.3 Control function capability

The plant control system controls the motor drivers for the motors in the shutdown rod system and the control drum system. After the user action to initiate startup, the plant control system automatically performs a sequence of preprogrammed actions to bring the Aurora to full power operation. Control drum and shutdown rod position process variables from the control drum system and the shutdown rod system are measured by the plant control system.

During normal operation, the plant control system acts as an intermediary between the reactivity management system and the control drum system. Drum position setpoints determined by the reactivity management system are communicated to the plant control system. The plant control system verifies that the requested drum position value is within the drum position operating limit, and then commands the control drum to the requested position. Figure 2-26 is a block diagram showing a simplified data flow between plant control system components.

The plant control system transmits data to the power conversion system, receives power conversion system status data, and can control components in the power conversion system to ensure optimal operation of the Aurora reactor.

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Figure 2-26: Simplified overview of data flow

2.7.4.3 Information display system

The information display system presents the status of parameters in the Aurora in the onsite monitoring room and remotely as needed. The primary function of the information display system is to show the current state of the plant. Important parameters are displayed in real-time on fixed displays or indicators that cannot be reconfigured by the monitor personnel. Other parameters are displayed on user-configurable displays or indicators that can display real-time data and past data.

The data displayed by the information display system comes from multiple sources. Important real-time parameters displayed on fixed indicators receive data directly from the associated instrumentation. Other real-time displays receive data that has been aggregated by the plant control system. Displays that show past data receive this data from a database that is part of the plant control system.

2.8 Power conversion system

2.8.1 Introduction to the power conversion system

The power conversion system functions to remove heat from the heat exchanger system (see Section 2.6) and convert the heat energy to electricity. The power conversion system is synonymous with the secondary system for the Aurora. It can also operate in turbine bypass mode, in which heat is rejected to the environment. The power conversion system consists of a turbine, generator, controls, and auxiliary subsystems. This is an off-the-shelf system that will be installed with the Aurora reactor.

2.8.2 Bases of the power conversion system

2.8.2.1 Design bases of the power conversion system

The power conversion system has no functions that are relied on for safe operation of the Aurora during the maximum credible accident, and therefore has no design bases.

2.8.2.2 Performance bases of the power conversion system

The power conversion system is designed to meet all of the following performance bases:

- The power conversion system utilizes heat from the heat exchanger system to create electricity during power operation.
- The power conversion system provides the capability for complete turbine bypass flow in the event of a turbine trip.
- The power conversion system provides the capability for partial turbine bypass flow in the event of reduced electrical demand.
- The power conversion system provides the capability for storing the full volume of secondary system working fluid during reactor maintenance.
- The power conversion system is monitored continuously during operation to detect failures.
- The power conversion system turbine trips automatically under abnormal conditions.

2.8.3 Description of the power conversion system

The Aurora reactor is designed to be compatible with different off-the-shelf power conversion systems or customized secondary systems. Nominally, the power conversion system for the Aurora reactor will utilize a sCO_2 Rankine cycle.

Because the power conversion system does not have any design bases, a detailed description of the power conversion system is not provided.

2.9 Electric power system

2.9.1 Introduction to the electric power system

The electric power system functions to serve power from the power conversion system (see Section 2.8) and the onsite energy storage system to both onsite and offsite loads. Because the Aurora is intended to serve communities in off-grid locations, it is completely grid independent. The Aurora provides its own power for onsite systems and treats the offsite grid strictly as a load rather than as a potential source of power. An energy storage system is used between the power conversion system and the offsite grid to reduce short-term grid demand fluctuations on the power conversion system. These functions are accomplished via three subsystems:

- Facility power system
- Energy storage system
- Offsite power system

2.9.2 Bases of the electric power system

2.9.2.1 Design bases of the electric power system

The electric power system has no functions that are relied on for safe operation of the Aurora, and therefore no design bases.

2.9.2.2 Performance bases of the electric power system

The electric power system is designed to meet all of the following performance bases:

- The electric power system supplies all onsite and offsite electrical loads.
- The electric power system prioritizes supplying onsite electrical loads.
- The electric power system can both transfer power to and receive power from the energy storage system.
- The energy storage system can be charged by the power conversion unit.
- The energy storage system can supply power to the facility power system and offsite power system.
- The offsite power system supplies power to offsite loads.

2.9.3 Description of the electric power system

A simplified block diagram of the electric power system is illustrated in Figure 2-27. The power conversion system generator is connected to the energy storage system. The energy storage system transmits power to the facility power system and the offsite power system. The energy storage system includes battery storage and an inverter, and it can supply alternating current power to the facility power system when the power conversion system is offline. The offsite power system includes a disconnect switch to isolate the grid from the electric power system to prioritize supplying onsite electrical loads. The offsite power system also includes components, such as transformers, to interface with the grid. The offsite power system connects to an offsite transmission grid to distribute the power generated by the Aurora to electricity consumers. The transmission grid acts strictly as an electrical load, and not as a source of electrical power.

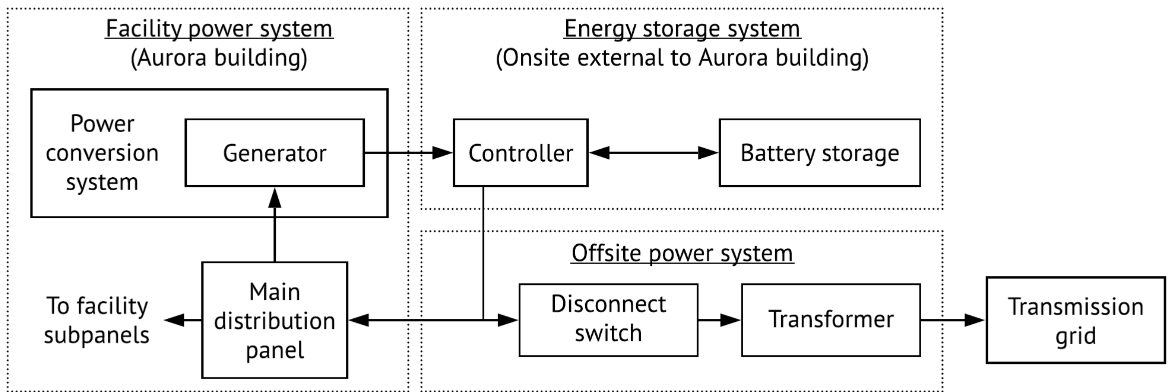

Figure 2-27: Overview of electric systems

Systems that are used to shut down the reactor and achieve a safe state are passive and do not require electricity. This is a key characteristic to the inherent safety of the Aurora design, because these systems can be maintained indefinitely in a safe shutdown through natural forces and simplicity of design. Therefore, the safe operation of the reactor is independent of onsite and offsite power, and the electric power subsystems are only briefly described in this section since they have no safety-significant function.

The electric power system functions to supply power from the power conversion system to both facility and offsite loads, as well as to charge the energy storage system. The electric power system prioritizes powering onsite loads over offsite loads. During normal operation, power that is generated in excess of onsite loads is transmitted via the offsite power system to serve offsite loads. Power generated in excess of both onsite and offsite loads is used to maintain the battery storage in the energy storage system. The power conversion system can also operate in bypass mode to reduce electrical power output if the energy storage system is at capacity and power generation is in excess of onsite and offsite loads.

The energy storage system functions to decouple the power conversion system from transmission grid load fluctuations. Batteries in the energy storage system store excess power produced by the power conversion system and provide that power to both onsite and offsite loads as required. The energy storage system provides power for startup and normal controlled shutdown capabilities. The system can also be used to provide uninterrupted power in the event the power conversion system is temporarily unavailable. The capacity of the backup storage is scaled based on expected need for such backup power. Because the safe operation of the reactor is independent of power, backup battery capacity is determined by the power requirements to restart the Aurora, offsite power demand, and the length of time the offsite power demand is to be maintained. The system can also be used to temporarily support power demand up to the rated output of the energy storage system, which may be greater than the rated output of the power conversion system generator.

The offsite power system can be configured to meet the specific needs of the customer, but it only functions to supply power to offsite loads and to disconnect the offsite loads from the electric power system. The electric power system is designed such that all onsite loads are met by onsite power sources.

2.10 Building and auxiliary systems

2.10.1 Summary description

The Aurora reactor and most of the associated systems described in this chapter are housed in a single A-frame structure (shown schematically in Figure 2-28) with two floors that occupies less than 5,000 square feet of land area. The simplicity and the small size of the Aurora and its associated systems enable this compact site footprint. Some components, such as the air-cooled coolers (or radiators) for the power conversion system (see Section 2.8), the energy storage and offsite power systems (see Section 2.9) are located outside of the building. Other auxiliary systems, described in this section, are located in and around the building.

The building system (see Section 2.10.2) is designed to house the Aurora reactor and associated systems.

The fire protection system (see Section 2.10.3) is designed to prevent fires and protect from damage by fire. It is configured as described in the Fire Protection Program, described in Chapter 21, "Fire Protection Program description," and submitted under Part VII.

Other auxiliary systems are described in Section 2.10.4.

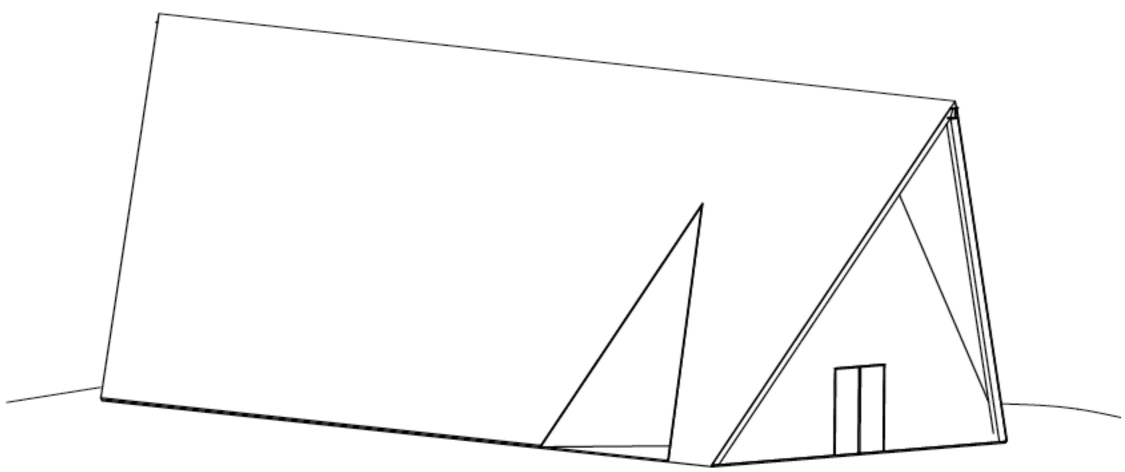

Figure 2-28: Building system

2.10.2 Building system

2.10.2.1 Introduction to the building system

The building system consists of a single building that functions to house the Aurora reactor and most of the associated systems. It has a very small footprint of less than 5,000 square feet and is designed to support the flexible siting approach described in Chapter 1, "Site envelope and boundary." The primary function of the building is to locate the reactor module in a configuration that ensures proper cooling can be maintained. It also serves to protect the reactor and associated systems, and to provide a habitable environment for personnel.

2.10.2.2 Bases of the building system

2.10.2.2.1 Design basis of the building system

The design basis for the building system, a subsystem of the building and auxiliary systems (BAS), is as follows:

DB.BAS.01 The building system provides for the emplacement of the reactor module in a configuration that supports passive cooling of the module shell.

2.10.2.2.2 Performance bases of the building system

The building system is designed to meet all of the following performance bases:

- The building system could provide two passive physical barriers to fission product release.
- The building system houses and protects the reactor system and most of the associated systems.
- The building system provides a habitable environment for personnel.
- The building supports solar panels for supplemental power generation.

2.10.2.3 Description of the building system

The building is a single A-frame structure. As shown in cutaway views in Figure 2-29 and Figure 2-30, it consists of three main areas:

- Atrium
- Power conversion system area
- Reactor area

These areas encompass two elevations. The atrium contains multiple rooms on both the ground floor and the basement, and is normally occupied. The power conversion system area is on the ground floor, and contains both the power conversion system, and the instrumentation and control cabinets, or enclosures (see Section 2.7). It is not normally occupied but entered regularly for routine maintenance. Further information on access authorization, authentication, and control is included in the Physical Security Plan, described in Chapter 18, "Security plans," and submitted under Part VII. The reactor area in the basement and is typically not occupied during normal operation.

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Figure 2-29: Side view cutaway of building

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Figure 2-30: Top view cutaway of building, showing ground floor and basement

The power conversion system area has a secure door that opens to the rear of the building, and is used for moving large components into and out of the building. It also contains a crane, mounted from the beams of the A-frame, that is used for lifting heavier components. A removable floor above the reactor module emplacement allows for the raising and lowering of reactor components by crane from the power conversion system area to the reactor area in the basement. The door, the crane, and the removable floor are only used during construction and maintenance, and are otherwise locked in place to prevent their use during normal operation.

The reactor module is emplaced in the basement of the building, as shown in Figure 2-31. The reactor module emplacement is designed to rigidly support the reactor module at the top of the module shell, such that the module equipment housing is located above the basement floor-level, and the module shell extends below floor-level. The module shell is not in direct contact with the emplacement except for at the rigid mounting, it has an air gap between the shell and the concrete cylinder that surrounds it. This gap, known as the reactor cavity, supports passive cooling of the reactor module through natural convection of air, as described in Chapter 5.1.

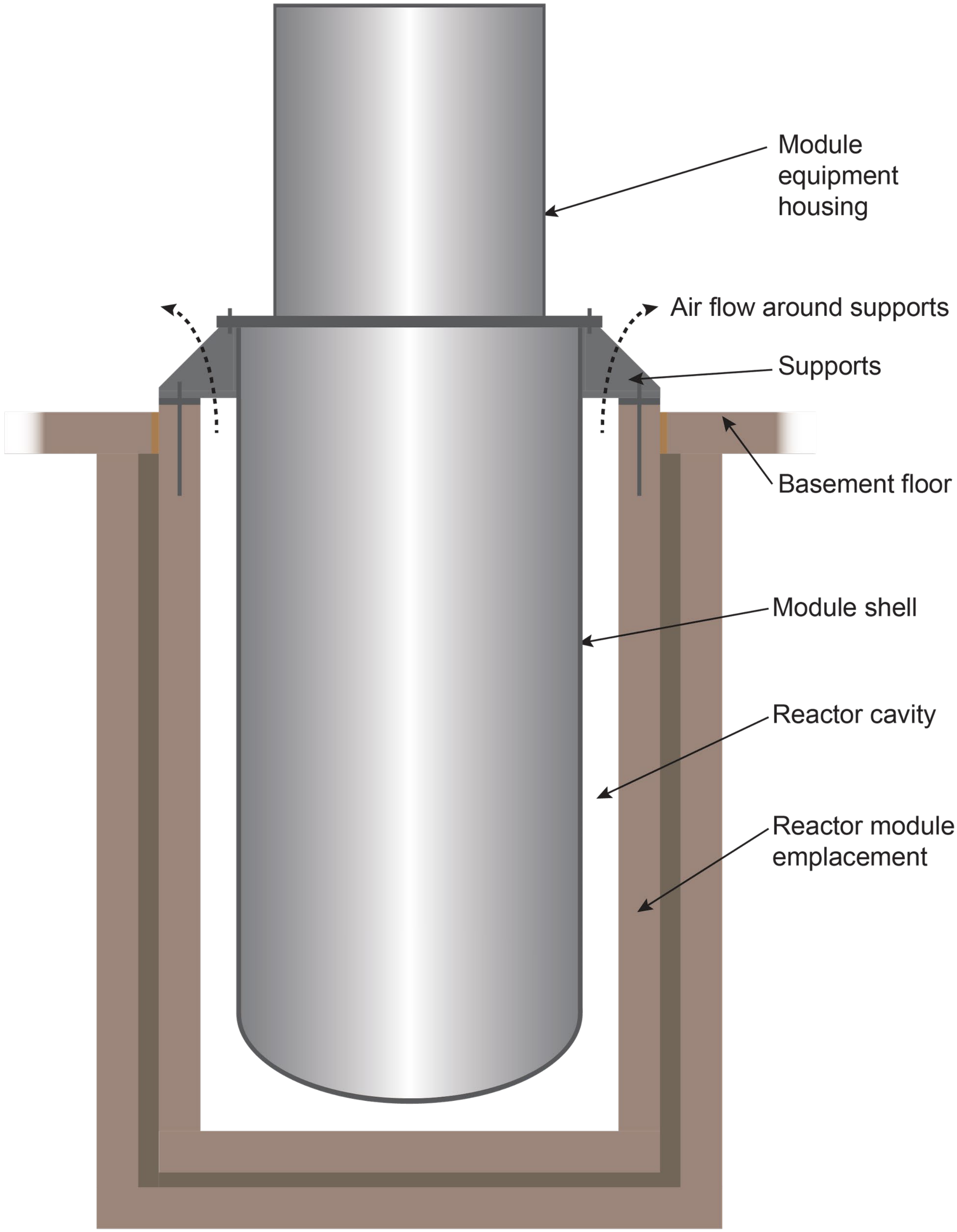

Figure 2-31: Reactor module emplacement

2.10.2.4 Materials of the building system

The Aurora building is mostly of standard steel and concrete construction. The materials for the construction of the Aurora plant are non-combustible materials in accordance with the Fire Protection Program.

The exterior walls of the building and certain interior walls serve as rated fire barriers as part of the fire protection system (see Section 2.10.3). The boundaries that separate fire areas within the building are constructed of rated concrete or glass barriers, with the appropriate fire rated doors between areas. External walls are steel and concrete.

2.10.2.5 Design evaluation of the building system

2.10.2.5.1 Passive convective cooling from the module shell

As described in the safety analysis in Chapter 5.1, the Aurora reactor does not rely on active cooling during the decay heat phase following shutdown during the maximum credible accident. The reactor passively conducts heat between the systems within the reactor module (the reactor core system, reflector system, shielding system, enclosure system, and heat exchanger system), distributing heat throughout the substantial thermal mass presented by the module. The heat is then removed from the module via natural convection at the surface of the module shell. The safety analysis shows that this passive heat removal is sufficient to maintain acceptable fuel temperatures and meet the top-level safety goal of the Aurora in the event of the maximum credible accident.

The building system provides the proper emplacement of the reactor module to ensure that natural convection can remove sufficient heat from the surface of the module shell. Design commitments are made for the building system to ensure that the reactor emplacement supports passive cooling of the module shell (DC.BAS.01.A).

2.10.2.6 Summary of the building system

Design basis:

DB.BAS.01 The building system provides for the emplacement of the reactor module in a configuration that supports passive cooling of the module shell.

Design evaluation summary:

This section describes the building system, which primarily functions to support the reactor module in a configuration that supports passive cooling of the module shell. The transient analysis in Chapter 5 shows that, when configured as designed, the reactor module has sufficient passive cooling due to natural convection on the surface of the module shell to maintain fuel temperatures below their required limits during the decay heat phase of the maximum credible accident (without active cooling). A design commitment is made to ensure proper as-built configuration of the building system prior to operation and the appropriate programmatic controls are in place to verify it.

Design commitments and programmatic controls:

DC.BAS.01.A The critical components of the reactor module, as identified in the appropriate procedure, are installed in the reactor module emplacement as described in the design documents referenced by the procedure.

POT.BAS.01.A (see Chapter 14)

2.10.3 Fire protection system

2.10.3.1 Introduction to fire protection system

The fire protection system consists of the barriers, systems, and equipment that function to prevent fires, and protect from damage by fire, as described in the Fire Protection Program. The fire protection system utilizes commercially-available equipment and does not have unique design characteristics.

2.10.3.2 Bases of the fire protection system

2.10.3.2.1 Design bases of the fire protection system

The design bases for the fire protection system, a subsystem of the building and auxiliary systems (BAS), are as follows:

DB.BAS.02 The fire protection system ensures that a single credible fire will not prevent achieving a safe state.

DB.BAS.03 The fire protection system provides the equipment to detects fires and to control and extinguish them promptly.

2.10.3.2.2 Performance bases of the fire protection system

The fire protection system is designed to meet the following performance bases:

- The fire protection system provides the means to manually extinguish fires in each fire area.

2.10.3.3 Description of the fire protection system

As described in Section 2.10.2, the exterior walls of the building system, and certain interior walls, are rated fire barriers. These rated fire barriers function to enclose separate fire areas and ensure that a single credible fire cannot spread from one fire area to another. By preventing the spread of fires from one fire area to another, the fire barriers ensure than a single fire cannot prevent the reactor from reaching a safe state, as described in the Fire Hazards Analysis.

In addition to the rated fire barriers, the Fire Protection Program, through the Fire Hazards Analysis requires for the installation of equipment to detect, control, and extinguish fires. Specifically, the fire protection system components include detectors, manual pull stations, fire extinguishers, and standpipes for fire department connection.

Administrative controls required by the Fire Protection Program are handled under a proposed license condition, in Part VI, “Proposed license conditions.”

2.10.3.4 Materials of the fire protection system

As described in Section 2.10.2.4, the building is constructed using non-combustible materials in accordance with the Fire Hazards Analysis. The fire barriers are rated concrete or glass barriers.

2.10.3.5 Design evaluation of the fire protection system

2.10.3.5.1 Rated fire barriers

As described in Chapter 6, “Fire protection,” and Chapter 21, the Aurora is designed to minimize both the probability of occurrence and the consequences of a fire. One of the key design aspects that minimizes the consequences of a fire is the use of rated fire barriers to create distinct fire areas that prevent the spread of fire within the building. The Fire Hazards Analysis describes the required barriers. Design commitments (DC.BAS.02A and B) are taken to ensure that the fire barriers properly separate components and cabling as described in the Fire Hazards Analysis, and that openings and penetrations in the fire barriers are properly protected to ensure the functionality of the barrier.

A further design commitment (DC.BAS.02.C) is taken to ensure the adequacy of the Fire Hazards Analysis, and the fire protection system via a safe state analysis. A final safe state analysis depends on the detailed design of electrical systems, and therefore a commitment is made to conduct the final safe state analysis on the Aurora plant once final design drawings, including detailed drawings of the electrical systems, are available.

2.10.3.5.2 Other fire protection system components

The Fire Protection Program, contained in the Fire Hazards Analysis, and the required fire protection system components, are informed by a detailed analysis of regulations, guidance documents, and standards. The appropriate commitments (DC.BAS.03.A-B) are taken to install and test these components.

2.10.3.6 Summary of the fire protection system

Design basis:

DB.BAS.02 The fire protection system ensures that a single credible fire will not prevent achieving a safe state.

Design evaluation summary:

The Fire Hazards Analysis describes how a single credible fire cannot prevent achieving a safe state. The Fire Hazards Analysis refers to a Safe State Analysis report that shows the preliminary results of the analysis, which must be updated for the final Aurora design. As described in this section, rated fire barriers are a critical component of the fire protection system for maintaining the ability to achieve a safe state. Design commitments are taken to ensure these barriers are in place and configured properly, and a design commitment is made to conduct the final Safe State Analysis after completion of the final design.

Design commitments and programmatic controls:

DC.BAS.02.A Components and cabling that could adversely impact an automatic reactor trip and initiate a loss of heat sink will be separated from each other by fire barriers.

POT.BAS.02.A (see Chapter 14)

DC.BAS.02.B Openings and penetrations through fire barriers are protected by components (e.g. fire doors, fire dampers, or penetration seals) having fire resistance equivalent to those of the barrier.

POT.BAS.02.B

DC.BAS.02.C A safe state analysis will be completed on the final Aurora design, and will show the final design to meet the acceptance criteria as defined in the safe state analysis report.

ITAAC.SD.02 (Part VI)

Design basis:

DB.BAS.03 The fire protection system provides the equipment to detects fires and to control and extinguish them promptly.

Design evaluation summary:

This section describes the fire protection system, which in addition to rated fire barriers includes equipment for the detection and extinguishing of fires. Design commitments are taken to ensure that this equipment is in place and tested and the appropriate programmatic controls are in place to verify it.

Design commitments and programmatic controls:

DC.BAS.03.A Manual pull stations or individual fire detectors provide fire detection capability and can be used to initiate fire alarms.

POT.BAS.03.A (see Chapter 14)

DC.BAS.03.B The fire protection system provides for manual fire fighting capabilities in each fire area.

POT.BAS.03.B

2.10.4 Other auxiliary systems

The Aurora contains additional systems that are required by relevant operating program but are not directly tied to the safety analysis in Chapter 5.1, and therefore do not correspond to specific design bases. These systems are described briefly in this section. The full descriptions of the required systems are contained in the respective program document; several operating programs are submitted under Part VII.

The Emergency Plan, as described in Chapter 9, "Emergency plans," and submitted under Part VII, require specific systems that provide the capability to identify an emergency, to monitor the emergency, and to appropriately to the emergency. An ITAAC (ITAAC.EP.01) is taken to ensure that the required preoperational testing is completed to verify that the installed systems meet the requirements described in the Emergency Plan. Administrative controls required by the Emergency Plan are handled under a proposed license condition, in Part VI.

Access to the building is controlled using the physical security system, as described by the Physical Security Plan (see Chapter 18 and Part VII). Security measures for each area of the building are commensurate with the size and other characteristics of the Aurora plant. These measures include an access control system, intrusion detection system, and the appropriate monitoring and communications equipment to evaluate and respond to potential security incidents. An ITAAC (ITAAC.PS.01) is taken to ensure that the required preoperational testing is completed to verify that the installed physical security system meets the requirements described in the Physical Security Plan. Administrative controls required by the Physical Security Plan are handled under a proposed license condition, in Part VI.

Radiation monitoring is accomplished via the radiation monitoring system, as described in the Radiation Protection Program (see Chapter 20 and Part VII). This equipment is used to ensure that occupational limits from 10 CFR Part 20, "Standards for protection against radiation," are met. An ITAAC (ITAAC.RP.01) is taken to ensure that the required preoperational testing is completed to verify that the installed systems meet the requirements described in the Radiation Protection Program. Administrative controls required by the Radiation Protection Program are handled under a proposed license condition, in Part VI.

II.03 Radioactive materials produced in operation

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3 RADIOACTIVE MATERIALS PRODUCED IN OPERATION

3.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(3) requires the following:

> The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in part 20 of this chapter

The purpose of this chapter is to provide an overview of the radioactive materials that are expected to be produced during the operation of the Aurora reactor and how radioactive effluents and exposures are controlled. This chapter includes a description of radioactive sources and a summary of the activation analyses of airborne effluents to show that limits set in 10 CFR Part 20, "Standards for protection against radiation," are met through plant design features and engineered protective systems.

This chapter is informed by several guidance documents, including:

- NUREG-0800, "Standard review plan for the review of safety analysis reports for nuclear power plants: LWR [light water reactor] edition," Revision 4, issued September 2013, Chapter 12.2, "Radiation Sources"
- Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants," Revision 0, issued June 2007, Part I, "Standard format and content of combined license applications," Chapter C.I.12 "Radiation Protection"
- NEI 07-03A, "Generic FSAR [final safety analysis report] template guidance for radiation protection program description," Revision 0, issued May 2009

3.1 Introduction

Activation of materials in the Aurora reactor is dominated by the radiation field generated by fission events in the core during full power operations. The types and quantities of radioactive materials produced during full power operation bound what is generated during other operating modes such as shutdown and maintenance. In these other modes, the only radioactive material produced comes from the secondary activation of materials by other already-activated materials, which have a much lower source strength than that of the full-power fission source.

Because full power operation bounds all other operating modes, the types and quantities of radioactive materials produced by the Aurora can be conservatively overestimated by assuming: (1) that the reactor operates at full power for its entire lifetime, and (2) that all materials have been irradiated for the duration of the reactor lifetime without any replacement or purging. All results presented in this chapter come from analyses that make these assumptions, and therefore conservatively bound activation levels. Radiation precautions and shielding design source terms are based on these conservative activation levels. The Radiation Protection Program and shielding system design descriptions are further discussed in Chapter 20, "Radiation protection program description" and Chapter 2, "Description and analysis of structures, systems, and components," respectively.

The Serpent Monte Carlo code is used to determine activated quantities of materials, material source strengths, and flux profiles. Steady-state calculations determine the neutron flux distributions, and depletion calculations provide the quantities of activated materials produced in these fluxes during the operating lifetime. Although activation levels vary depending on distance from the core, time of irradiation, and shielding, all materials that fall within that core radiation field are discussed to provide a comprehensive understanding of the management of those materials.

Where possible, materials are deliberately selected to help mitigate activation and secondary radiation effects, thereby reducing exposure and dose rates. For example, the absorber material in the shielding system is almost entirely composed of boron carbide. This reduces secondary radiation effects compared to other potential shielding materials because no secondary gamma radiation is produced upon neutron absorption in boron-10.

3.2 Design features

The Aurora reactor has been designed to optimize reactivity and minimize the escape of radiation from the core. This is accomplished by the reflector system that immediately surrounds the active core, and the shielding system that surrounds the reflector system. As a result, the neutron and gamma flux are significantly reduced as the distance from the core increases.

3.3 Controlling sources of radiation

The Aurora reactor has been deliberately designed to minimize, to the extent possible, the amount of radioactive materials produced during operation. Radioactive materials that are produced during the operation of the Aurora plant include radioactive fluids and structural material.

3.3.1 Fluids

Potential radioactive fluids and radioactive airborne effluents produced inside the reactor module include fission products, backfill gas, the power conversion system working fluid, heat pipe working fluid, and the sodium thermal bond inside the reactor cells.

3.3.1.1 Fission products

The Aurora fuel, which is metal uranium-zirconium alloy (U-10Zr), is operated at very low burnups, and therefore generates few fission products. At this low burnup there is not significant fission product release from the fuel matrix, as described in Chapter 2. The minimal amounts of fission products that are released from the fuel matrix are contained and isolated in the gas plenum volume of each reactor cell.

The Aurora is not expected to operate with any damaged reactor cells, which prevents fission product release to the interior of the capsule or the module shell, keeping both the capsule and the module shell radiologically clean. This minimizes the possibility of releasing even the small quantity of fission products that leave the fuel matrix.

3.3.1.2 Reactor enclosures backfill gas

Argon is used to backfill the reactor enclosures and each reactor cell gas plenum. The total volume of the reactor enclosure backfill is small, as most of the space within the enclosures is occupied. The total volume of the reactor cell backfill is also small as the plenum volume in each cell is small.

The Aurora operates at near atmospheric pressure, such that the driving force for the enclosure backfill to escape the enclosures is minimal. In addition, the reactor cells are sealed, such that the reactor cell backfill would only escape into the capsule if a reactor cell fails. Nevertheless, in this evaluation, it is conservatively assumed that the entire activated inventory of argon is released simultaneously into the powerhouse basement, the smallest room that can be occupied by personnel. This provides a conservative bounding analysis of all possible leak rates of backfill gas, and reflects conditions that could only occur in a major accident, rather than during normal operations. Additionally, it is assumed that the entire inventory of backfill gas is irradiated for the duration of the 20 year reactor lifetime without any leaks or replacement, in order to estimate the maximum activation of the fluid.

This unlikely release of radionuclides produced from the activation of argon is then conservatively compared to the occupational limits set in 10 CFR Part 20, Appendix B, “Annual limits on intake (ALIs) and derived air concentrations (DACs) of radionuclides for occupational exposure; effluent concentrations; concentrations for release to sewerage.” This comparison is conservative because the occupational limits assume exposure for 2000 hours per year, and the major release analyzed would result in only transitory exposure.

Despite these major conservatisms, all radionuclide quantities for the backfill gas are below the occupational limits set in Appendix B to 10 CFR Part 20, except Ar-41 and H-3. Meeting 10 CFR Part 20 for what would be considered a major event at the Aurora is not required, since this type of release is not expected during normal operations. Table 3-1 compares the concentrations of all nuclides present in the activated argon to the regulatory limit concentrations for occupational inhalation. Section 3.3.1.6 explains further precautions taken for airborne effluent control.

Table 3-1: Argon backfill gas activated radionuclide concentrations

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3.3.1.3 Power conversion system working fluid

The power conversion system uses supercritical carbon dioxide (sCO_2) as the working fluid. The sCO_2 flows through the heat exchanger system, removes heat from the heat pipes of the reactor cells, and transports it to the power conversion system. It is the only material that enters and exits the reactor module during normal operation, and is the only potential source of secondary activation to components outside the reactor module. Therefore, in addition to evaluating leaks, the sCO_2 must be evaluated for secondary activation of other components. Both of these factors can be minimized by limiting the activation of the sCO_2.

3.3.1.3.1 Activation of sCO_2

The activation of sCO_2 is limited by both the shielding system, and the spatial distance from the reactor core. As described in Chapter 2, the heat exchanger shield portion of the shielding system shields the heat exchangers from direct exposure to the radiation field generated by the reactor core system. Additionally, since the heat exchanger system is located at a distance of several meters from the active core, the spatial separation further reduces the radiation field experienced by the heat exchanger system. Due to these features, the activation of sCO_2 fluid while it is in the heat exchanger system is minimal.

The total activation of the power conversion system working fluid is determined using the total power conversion system inventory of 5,000 kg of sCO_2. The main contributing activation product in the fluid is nitrogen-16. However, to ensure that the source strengths are maintained to levels as low as reasonably achievable, all activation products in the fluid are considered.

Fission and corrosion product activities are not a concern for the power conversion system working fluid. Due to the low burnup of the fuel, fission products are expected to stay almost entirely within the fuel matrix, inside the reactor cell can. As a result, there is no direct path for these fission products to interact with the sCO_2. As described in Chapter 2, sCO_2 is compatible with stainless steel at temperatures up to 550°C, which is a conservative upper bound on the temperatures inside the heat exchanger system. Therefore, corrosion, and the resulting circulation of activated corrosion products in the working fluid, is not a concern. Because fission and corrosion products are not expected in the power conversion system working fluid, the activated nuclides in the sCO_2 are limited to activation products.

3.3.1.3.2 Evaluation of sCO_2 leaks

As with the argon backfill gas, the evaluation of an sCO_2 release assumes that the entire inventory is released simultaneously into the volume of the powerhouse basement, the smallest room that can be occupied by personnel. This simultaneous release of the entire inventory is used to bound all potential leak rates and is not expected during the entire lifetime of the plant. Additionally, it is assumed that the entire inventory of sCO_2 is irradiated for the duration of the reactor lifetime without any leaks or replacement, in order to estimate the maximum activation of the fluid. Activation values used were those calculated at the end of the 20-year operating lifetime.

This unlikely release of radionuclides produced from the activation of the sCO_2 is then conservatively compared to the occupational limits set in Appendix B to 10 CFR Part 20. This comparison is conservative because the occupational limits assume exposure for 2000 hours per year, and the major release analyzed would result in only transitory exposure.

Despite these major conservatisms, all radionuclides created during the irradiation of the sCO_2 are below the occupational values established in Appendix B to 10 CFR Part 20.

Table 3-2 compares the concentrations of all nuclides present in the activated sCO_2 to the regulatory limit concentrations for occupational inhalation. Every radionuclide generated is orders of magnitude below the applicable occupational limits.

Table 3-2: sCO_2 radionuclide concentrations

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3.3.1.3.3 Evaluation of secondary activation by sCO_2

Analysis of the secondary activation from the sCO2 is not performed as part of this chapter. Because the activation of the sCO2 is small, further activation by the sCO2 is expected to be negligible.

3.3.1.4 Heat pipe working fluid

Potassium is used for the working fluid in the heat pipes and is solid until operation has begun and operating temperatures melt the potassium; eventually a working vapor is formed that is used to carry the heat inside the heat pipe via evaporation and condensation in the sealed heat pipe volume. For details on the how the heat pipe and its materials operate see Chapter 2.

The potassium is sealed inside the heat pipes to facilitate heat pipe operation, and the heat pipes are designed to remain sealed for their entire operating lifetime. Heat pipes remain installed in the core except in rare cases of unexpected maintenance. If a heat pipe has failed and must be removed from its installed location, that single heat pipe will be removed and replaced. The specifics for the entire inventory of the heat pipe working fluid, including types, quantity, and specific activity, can be found in Table 3-3.

No further activation analysis is presented in this chapter because the heat pipe working fluid cannot exit the reactor module and be a hazard to onsite personnel.

3.3.1.5 Thermal bond

A sodium bond is used between the fuel ingot and the heat pipe in the reactor cell, as described in Chapter 2. The sodium bond is solid during cold shutdown and maintenance modes and is therefore easy to control. During full power operation, the sodium melts to become a liquid, but remains in the reactor module. The expected quantities and activities for the bond sodium are shown in Table 3-3.

No further activation analysis is presented in this chapter because the sodium bond cannot exit the reactor module or become a hazard to onsite personnel.

3.3.1.6 Aurora powerhouse atmospheres

The source of airborne activity in the Aurora powerhouse is primarily due to the minimal possible leakage of the reactor module backfill gas or power conversion system working fluid into the powerhouse volumes. The Aurora facility is equipped with two independent ventilation systems. Each system functions for a single floor and is outfitted with radiological filters for removing activated byproducts in the air. Despite the actual presence of these systems, material activation concentrations conservatively assume full releases of the entire inventories of backfill gas and power conversion system working fluid into the stagnant air of these volumes, with no ventilation or filtration operating. In practice, ventilation systems will constantly be circulating facility air and removing radionuclides from the air before concentrations reach the levels analyzed.

Table 3-3: Possible effluent types, quantities, and activities

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3.3.2 Fuel

The neutron and gamma source strengths from the fuel are determined for fuel with a uranium enrichment of 19.75% operating at 4 MWth for 20 effective full power years. The solid fuel is enclosed by the reactor cell can walls and is adequately shielded during the lifetime of the plant.

3.3.3 Structural and other materials

Other materials that become activated when exposed to the core radiation field include the following: reflectors, shielding, control drums, stainless steel cell cans, heat pipe walls, enclosures, and the reactor core base plate. All reactor core components are permanently located in the capsule which is located inside the module shell, and both the capsule and the module shell are sealed during normal operations, preventing access. Therefore, structural materials and other materials are easily shielded and isolated to protect personnel and the public from any exposure that would originate from these materials during normal operations.

The module equipment housing is located on top of the module shell lid, and acts as an extension of the module shell. It contains the control drum drive shafts and drive motors, and the shutdown rod drive lines and withdrawal motors, among other components. This equipment can only be accessed during a mode when the reactor is shut down. The module equipment housing is protected from the core radiation field with adequate shielding, described in Chapter 2, to keep all limits within 10 CFR Part 20.

3.4 Off-normal events

No postulated off-normal events introduce additional material into the reactor module for potential activation. The maximum credible accident analyzed in Chapter 5.1, "Transient analysis," results in no release of fission products. Compared to the 20 years of full power operation assumed for the activation analyses, the duration of any potential transient will have a negligible effect on the end-of-life activation of materials. Accordingly, no special considerations for off-nominal events must be taken when accounting for the activated materials produced during operation.

3.5 Maintenance and decommissioning

All activation analyses in this chapter assume 20 years of full power operation and assume that all materials have been irradiated for the full duration without replacement or purging, to conservatively bound the activation analyses. Therefore, the source strengths in this chapter are bounding for any maintenance or decommissioning activities.

II.04 Principal design criteria

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4 PRINCIPAL DESIGN CRITERIA

4.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(4) has requirements for the design of the facility including the following:

> (i) The principal design criteria for the facility. Appendix A to part 50 of this chapter, "General Design Criteria for Nuclear Power Plants," establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units;
>
> (ii) The design bases and the relation of the design bases to the principal design criteria;
>
> (iii) Information relative to materials of construction, arrangement, and dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with adequate margin for safety.

The purpose of this chapter is to provide the principal design criteria (PDC) for the facility, the design bases and the relation of the design bases to the PDC, and information sufficient to provide reasonable assurance that the design will conform to the design bases with adequate margin for safety.

4.1 Methodology

4.1.1 Background

As described in 10 CFR 52.79(a)(4)(i), Appendix A, “General design criteria for nuclear power plants,” to 10 CFR Part 50, establishes minimum requirements for PDC for light water reactors (LWRs). These general design criteria (GDC) are prescriptive and technology-specific, explicitly specifying the minimum requirements for LWRs. Section 52.79(a)(4) to 10 CFR states that the GDC provide, “guidance to applicants in establishing principal design criteria for other types of nuclear power units,” but does not supply specifics.

Due to the technology-specific nature of the GDC and unclear means of applying them to non-LWR designs, Regulatory Guide (RG) 1.232, “Guidance for developing principal design criteria for non-light-water-reactors,” Revision 0, was issued in April 2018 in an effort to provide guidance in “modifying and supplementing the GDC to develop PDC for any non-LWR designs.” RG 1.232 provides a new set of design criteria, termed advanced reactor design criteria (ARDC), that could serve the same purpose for non-LWRs as the GDC serve for LWRs. Essentially, each GDC was taken and the technology-specific terminology was modified to make it potentially more broadly applicable. However, while some of the language changed, many of the assumptions around the basic system design remained. The concepts outlined in the ARDC, like the GDC, are specific and based on large reactors requiring active safety systems.

4.1.2 Aurora approach

The approach presented here to meet the intent of 10 CFR 52.79(a)(4) is to utilize the safety case of the Aurora to develop the PDC. Rather than starting with design criteria that were developed for a different technology (e.g., GDC), and then defining the design bases that satisfy each of those design criteria for the Aurora, the methodology used to develop the Aurora's PDC is the converse. The principal design criteria development process provides a way to double-check, from a fundamental safety function perspective, that the design bases are adequate and encompassing.

The design bases for the Aurora were developed through an iterative process between design of systems and subsequent safety analysis of those systems. As the design of the Aurora evolved, the safety analysis advanced to continue confirming the design assumptions. As a result, design bases were developed to describe the various Aurora systems. Therefore, design bases are the characteristics of a system that ensure the safe operation of the Aurora reactor.

To provide reasonable assurance that the design will conform to the design bases with adequate margin for safety, design commitments and associated programmatic controls are taken for each design basis. Programmatic controls are used to verify that design commitments are met, and therefore that design bases are satisfied. These controls include preoperational tests (POTs), inspections, tests, and analysis acceptance criteria (ITAAC), startup tests (SUTs), and the Technical Specifications (TS). These programmatic controls are contained in the license application in the following ways:

- The POTs are conducted as the first phase of the initial testing program (ITP), which can be found in Chapter 14, “Preoperational testing and initial operations”. These tests must be completed, and a summary report created, to satisfy an ITAAC. The ITAAC are found in Part VI, “Proposed license conditions.”

- The SUTs are conducted during and after initial fuel loading. These are conducted as the second phase of the ITP and are found in Chapter 14. Collectively, the POTs and the SUTs are used to verify that the system is built and functions as described.

- The TS provide the operating limits for the reactor. These are found in Part IV, “Technical Specifications.” The TS are used to ensure that the reactor never reaches a more challenging condition than that analyzed in the safety analysis.

Next, the design bases are grouped together under a PDC to encapsulate the goal for those design bases and custom PDC are proposed. This methodology is shown in Figure 4-1. Section 4.2 describes the PDC for the Aurora and the design bases that contribute to each.

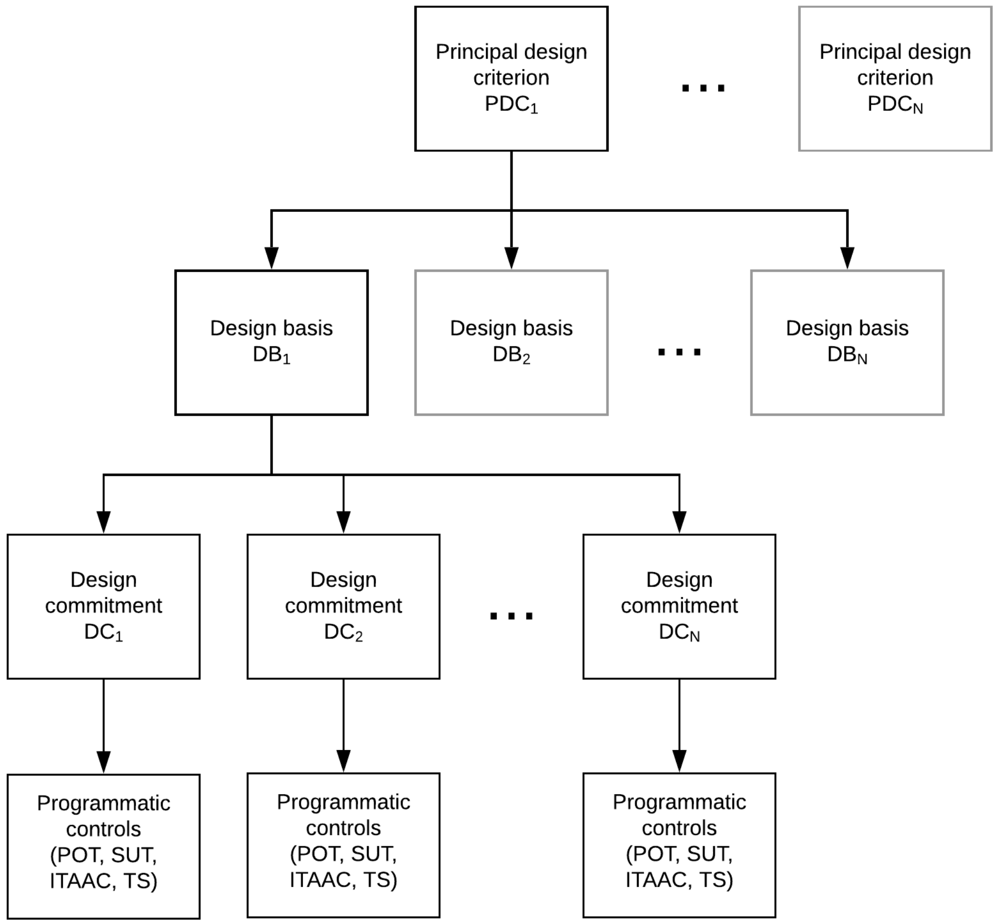

Figure 4-1: Relationship of principal design criteria to the design bases

4.2 Principal design criteria

The PDC for the Aurora closely parallel the fundamental safety functions, as discussed by IAEA Safety Standard, Specific Safety Requirements, No. SSR-2/1, Revision 1, "Safety of Nuclear Power Plants: Design." The IAEA Safety Standard, Specific Safety Requirements, No. SSR-2/1, Revision 1, "Safety of Nuclear Power Plants: Design," in addition to outlining the safety principles that were adapted for the Aurora, contains a discussion of the fundamental safety functions applicable to any nuclear reactor that must be met to satisfy these safety principles. The fundamental safety functions are very high-level, which explains their wide applicability. The three fundamental safety functions are adapted into the following:

1. Control of reactivity,
2. Removal of heat, and
3. Confinement of radioactive material.

Additionally, the Aurora PDC are consistent with the U.S. Nuclear Regulatory Commission (NRC) safety goals,[9] which are the following:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and not be a significant addition to other societal risks.

The Aurora PDC are consistent with the Aurora safety goal, which is to control the release of radionuclides to minimize the risk to the public and the environment. The safety analysis in Chapter 5 demonstrates that despite challenges to normal operation, the structures, systems, and components that are designed to respond to these challenges (either actively, passively, or inherently) are able to uphold the safety goal.

The PDC defined for the Aurora are listed in Table 4-1, along with the source of the language for the PDC.

Table 4-1: Principal design criteria for the Aurora

PDC	Title	Source
1	Confinement	Custom
2	Reactivity control	Custom
3	Heat removal	Custom
4	Fire protection	ARDC

4.2.1 PDC 1: Confinement

PDC 1 is unique to the Aurora, and is presented below:

> Structures, systems, and components responsible for maintaining confinement of radionuclides for the Aurora will perform their required functions during off-nominal events up to and including the maximum credible accident, or will minimize the severity of the challenges to those functions.

PDC 1 relates to the control of radionuclides. Specifically, since the fuel matrix is the primary confinement feature in the Aurora, the application of Appendix B quality assurance regarding fuel will ensure that the fuel is manufactured, shipped, stored, etc. in a quality assured manner in all stages.

[9] 51 FR 284044, August 4, 1986

Table 4-2: Design basis and design commitment for PDC 1

Design basis reference number	Design basis description	Design commitments and programmatic controls
DB.RXS.01	The reactor core system uses metal fuel with well characterized properties.	DC.RXS.01.A

Confinement of radionuclides in the Aurora occurs through inherent properties of the fuel material.[10] The two contributing factors to the confinement function that are related to the fuel are (1) the quantity of the material and (2) the metal form. These are further discussed as follows:

1. Because of the small size and power output of the Aurora, the small amount of fuel mass limits the amount of radionuclides present throughout the life of the reactor, which limits the amount of radionuclides present. Further, after a 20-year lifetime, the Aurora fuel has a burnup of less than 1 atom per cent (at.%). This small burnup means that very few radionuclides are generated in the fuel matrix during normal operation, which serves at all times to minimize the risk posed by challenges to the safety goal of the reactor.

2. The Aurora fuel is metal, in the form of a binary uranium-zirconium alloy, which has shown excellent performance to significantly higher burnups than the Aurora. Metal fuel, like other metals, is a relatively nonporous solid with a regular crystal lattice. As a result, and as shown in extensive data from decades of operation, the vast majority of fission products are retained within the fuel matrix at burnups less than 1%.

4.2.2 PDC 2: Reactivity control

PDC 2 is unique to the Aurora, and is presented below:

> Structures, systems, and components responsible for maintaining reactivity control of the Aurora will perform their required functions during off-nominal events up to and including the maximum credible accident, or will minimize the severity of the challenges to those functions.

PDC 2 relates to maintaining control of reactivity for the Aurora during off-nominal events. The design bases relating to PDC 2 are described in Table 4-3, together with a list of the design commitments and programmatic controls that verify these design bases are met.

[10] The Aurora also includes several structural barriers, including the reactor cell cans, the capsule, the module shell, the building basement, and the building first floor. However, the integrity of the fuel and the reactor cell cans is not challenged during even the maximum credible accident, and as such, discussion of the other structural barriers is not presented.

Table 4-3: Design bases, commitments, and programmatic controls associated with PDC 2

Design basis reference number	Design basis description	Design commitments and programmatic controls
	Reactor system	
DB.RXS.03	The reactor core system has inherently negative reactivity feedback.	DC.RXS.03.A SUT.RXS.03
	Control drum system	
DB.CDS.01	The control drum system is designed to limit both the rate and magnitude of reactivity insertion that the system can achieve so as to minimize the effect of an unintended reactivity insertion.	DC.CDS.01.A POT.CDS.01.A DC.CDS.01.B SUT.CDS.01.B DC.CDS.01.C POT.CDS.01.C
	Shutdown rod system	
DB.SRS.01	The shutdown rod system provides sufficient negative reactivity to achieve cold shutdown with insertion of one rod.	DC.SRS.01.A SUT.SRS.01.A1 SUT.SRS.01.A2
DB.SRS.02	The shutdown rod system fully inserts the shutdown rods within a sufficient time after receiving a trip signal to prevent damage to the reactor.	DC.SRS.02.A POT.SRS.02.A SUT.SRS.02.A TS.LCO.01
	Instrumentation and control system	
DB.ICS.01	The reactor trip system monitors reactor process variables and sends a reactor trip signal when a process variable exceeds a limit setpoint.	DC.ICS.01.A POT.ICS.01.A1 POT.ICS.01.A2 SUT.ICS.01.A1 DC.ICS.01.B POT.ICS.01.B1 POT.ICS.01.B2 DC.ICS.01.C POT.ICS.01.C1 POT.ICS.01.C2 SUT.ICS.01.C DC.ICS.01.D POT.ICS.01.D TS.LCO.02
DB.ICS.02	The reactor trip system sends a reactor trip signal to the shutdown rod system within a sufficient time of exceeding a limit to prevent damage to the reactor.	DC.ICS.02.A POT.ICS.02.A TS.LCO.02
DB.ICS.03	The reactor trip system provides the means for a reactor trip signal to be sent manually.	DC.ICS.03.A POT.ICS.03.A TS.LCO.02
DB.ICS.04	The reactor trip system requires deliberate action to reverse a reactor trip signal and return the system to normal operation.	DC.ICS.04.A POT.ICS.04.A TS.LCO.02

DB.ICS.05	The reactor trip system is protected against unauthorized configuration changes.	DC.ICS.05.A POT.ICS.05.A DC.ICS.05.B POT.ICS.05.B DC.ICS.05.C POT.ICS.05.C DC.ICS.05.D POT.ICS.05.D
DB.ICS.06	The reactor trip system is fail-safe.	DC.ICS.06.A POT.ICS.06.A DC.ICS.06.B POT.ICS.06.B DC.ICS.06.C POT.ICS.06.C DC.ICS.06.D POT.ICS.06.D

The importance of reactivity control is that it is the means to control the generation of heat in the reactor. Imbalances between the heat generation and the heat removal in the reactor core lead to changes in core temperatures. As such, one means of limiting fuel temperature during off-nominal events is by the control the reactivity of the reactor.

Reactivity is controlled in the Aurora through three distinct means: (1) the shutdown rods, (2) the control drums, and (3) the inherent characteristics of the reactor. These are further discussed as follows:

- Only a single rod must insert fully in order for the neutron chain reaction to be shut down and the core made subcritical, which is the required function of the shutdown rod system. As the core of the Aurora operates at near atmospheric pressure, there is no significant driving force that opposes rod insertion. Since the Aurora operates with a very low power density, it is relatively insensitive to any potential delay time that might elapse between the start of a transient until full rod insertion is achieved. As a result of this design, the shutdown rod system robustly provides its required function by inserting at least one out of three redundant shutdown rods in the safety analysis.

- The control drums are responsible only for compensating for slow reactivity changes due to fuel depletion during normal operations, and are not credited in the safety analysis. The control drums appear in the safety analysis (i.e., in the transient overpower) solely to describe the challenge they present in the case of their malfunction. As such, they provide no required functions.

- Inherent characteristics of reactivity control are a backstop to mitigate undesired off-nominal behavior. They are not considered passive means because “failure” of these characteristics is nonphysical. A degradation of inherent characteristics is possible, but complete failure cannot occur. In the Aurora, changes in reactor power are inherently controlled and limited through two means: (1) the physical core configuration, and (2) the large negative temperature coefficient of reactivity. The physical core configuration is the most reactive configuration during normal operations; any disruptions to the physical configuration of the core would lead the fuel to be in a less reactive state. The Aurora has an inherently large negative temperature coefficient of reactivity. This is primarily due to the large thermal expansion of the metal core materials during a heat up. The negative temperature coefficient of reactivity is conservatively modeled in the safety analysis by assuming the reactor power stays constant when heat removal by the secondary system is lost.

4.2.3 PDC 3: Heat removal

PDC 3 is unique to the Aurora, and is presented below:

> Structures, systems, and components responsible for removing heat from the fuel of the Aurora will perform their required functions during off-nominal events up to and including the maximum credible accident, or will minimize the severity of the challenges to those functions.

PDC 3 relates to maintaining heat removal capability from the fuel in the Aurora's reactor core system during off-nominal events. The design bases relating to PDC 3 are described in Table 4-4, together with a list of the design commitments and programmatic controls that verify these design bases are met.

Table 4-4: Design bases, commitments, and programmatic controls associated with PDC 3

Design basis reference number	Design basis description	Design commitments and programmatic controls
	Reactor system	
DB.RXS.02	The reactor core system is operated at steady state thermal power limits that prevent damage to the system during transients.	DC.RXS.02.A
DB.RXS.04	The reactor core system provides a pathway to conduct heat from the fuel to the surrounding systems and ultimately to reject it to the environment.	DC.RXS.04.A SUT.RXS.04.A DC.RXS.04.B SUT.RXS.04.B
DB.RXS.05	The reflector system provides a pathway to conduct heat from the reactor core system to the surrounding systems and ultimately to reject it to the environment.	DC.RXS.05.A POT.RXS.05.A
DB.RXS.06	The shielding system provides a pathway to conduct heat from the reactor core system and reflector system to the surrounding systems and ultimately to reject it to the environment.	DC.RXS.06.A POT.RXS.06.A SUT.RXS.06.A1 SUT.RXS.06.A2
	Reactor enclosure system	
DB.RES.01	The reactor enclosure system provides a pathway to conduct heat away from the systems inside it and to reject it to the environment.	DC.RES.01.A POT.RES.01.A1 POT.RES.01.A2 SUT.RES.01.A
	Heat exchanger system	
DB.HXS.01	The heat exchanger system provides a pathway to conduct heat from the heat pipes of the reactor core system to the surrounding systems and ultimately to reject it to the environment.	DC.HXS.01.A SUT.HXS.01.A
	Building system	

DB.BAS.01	The building system provides for the emplacement of the reactor module in a configuration that supports passive cooling of the module shell.	DC.BAS.01.A POT.BAS.01.A

Fuel temperatures can be limited if sufficient heat is removed from the reactor. In the Aurora, heat removal is controlled by three distinct means: (1) normal operation of the secondary system (i.e., the power conversion system), (2) conduction throughout the reactor materials, and (3) passive heat rejection to the environment. These are further discussed as follows:

- Since the maximum credible accident from the safety analysis involves a failure in the power conversion system, heat removal via the power conversion system is accordingly not included in this discussion on required functions.

- Because the fuel is the heat generation source, the first order heat removal function during a heatup is to cool the fuel. It is important to note that the Aurora operates at a very low power density,[11] which serves to limit the amount of heat generated in the fuel, both at power and in decay heat, that must be dissipated to surrounding materials when normal cooling via the secondary system is decreased or lost. The physical effect of interest is conducting the heat from the metal fuel to other reactor components, most of which are also metal. Specifically, the sensible thermal mass of the reactor is what initially drives the temperature response of the fuel following off-normal events. Conduction through the reactor occurs through the heat pipes and inherent heat transfer parameters. Thermal contact between adjacent bodies, as well as the heat pipes, ensure effective heat transfer throughout the entire reactor module.

- As the heat generated by the fuel is distributed throughout the reactor, some of this decay heat is rejected to the surrounding environment through natural convection to the air in the reactor cavity surrounding the reactor module. Because of the characteristics of the Aurora (e.g., low power density, high thermal conductivity, high heat capacity), this heat rejection rate need not be large to have a very beneficial effect on limiting temperatures during a heatup event.

4.2.4 PDC 4: Fire protection

PDC 4 for the Aurora is similar to GDC 3 and ARDC 3 and is quoted below:

> Structures, systems, and components shall be designed and located to minimize the probability and effect of fires and explosions. Noncombustible and fire-resistant materials shall be used wherever practical throughout the unit. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components. Firefighting systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

[11] The Aurora operates at a low power density that is one to two orders of magnitude smaller than a light water reactor, up to three orders of magnitude lower than a liquid sodium cooled fast reactors, and lower than other reactors that rely primarily on conduction for decay heat removal such as high temperature gas reactors.

PDC 4 relates to protection from fires and explosions. The design bases relating to PDC 4 are described in Table 4-5, together with a list of the design commitments and programmatic controls that verify these design bases are met.

Table 4-5: Design bases, commitments, and programmatic controls associated with PDC 4

Design basis reference number	Design basis description	Design commitments and programmatic controls
DB.BAS.02	The fire protection system ensures that a single credible fire will not prevent achieving a safe state.	DC.BAS.02.A POT.BAS.02.A DC.BAS.02.B POT.BAS.02.B DC.BAS.02.C ITAAC.SD.02
DB.BAS.03	The fire protection system detects, controls and extinguishes promptly those fires that do occur.	DC.BAS.03.A POT.BAS.03.A DC.BAS.03.B POT.BAS.03.B

Since fires can cause inadvertent action of equipment, the Aurora is designed with fire protection in mind. Specifically, the Fire Hazards Analysis takes into account two items, which are (1) that a single credible fire will not prevent the reactor from being shutdown (i.e., achieving a safe state), and (2) that programmatic controls are implemented onsite to limit the consequences of a fire. These are further discussed as follows:

- The Fire Hazards Analysis performed for the Aurora is largely deterministic. It takes advantage of the simple facility design and implements fire barriers to separate important equipment into independent fire areas. This physical separation ensures that even if a fire were to occur, it would not propagate to other components. Fire protection is taken account during the design phase of the Aurora, resulting in redundancy of equipment through the various fire areas.

The Fire Protection Program, which is included as part of the Fire Hazards Analysis, contains requirements for the facility for certain equipment and administrative controls. These requirements are such, that if a fire were to occur, it can be detected and mitigated as needed.

4.2.5 Conclusion

The development of the principal design criteria was able to be deeply iterative with the design of the Aurora as it evolved under the safety principles to meet the Aurora safety and operational goals, and as design bases were codified through the safety analysis. This ability to be deeply iterative and responsive in design for safety is largely enabled through the Oklo digital twin capabilities for the Aurora as a small, tightly coupled design. The final principal design criteria for the Aurora with the design bases which ensure those criteria are met still follow closely to the systems highlighted in the qualitative tree presented to the NRC in early 2019, shown in Figure 4-2.

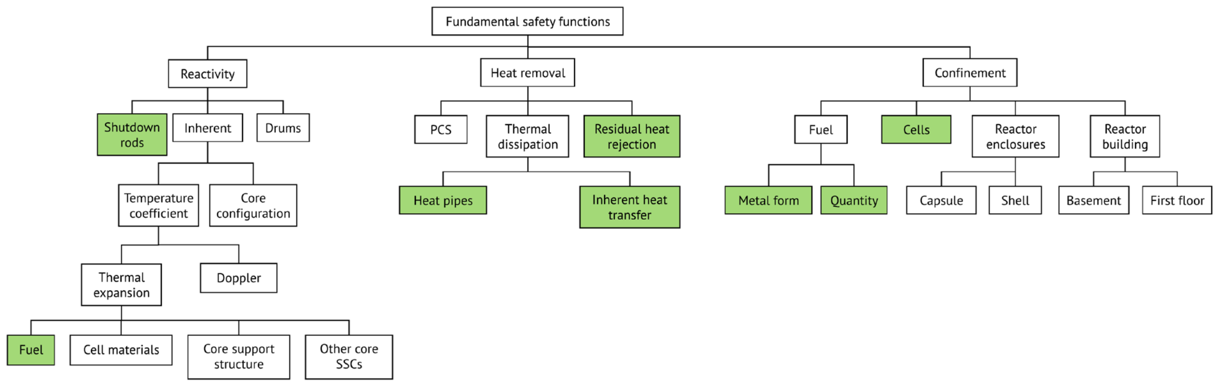

Figure 4-2: Top-level fundamental safety functions and the identified supporting safety functions for the Aurora

II.05 Transient analysis

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5 TRANSIENT ANALYSIS

5.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(5) requires the following:

> An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 and 50.46a of this chapter.

It is important to note that 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR 50.46a, "Acceptance criteria for reactor coolant system venting systems," do not apply to the Aurora because the Aurora is not a light water reactor (LWR) and does not have the relevant systems. Further information on the applicability of 10 CFR 50.46 and 10 CFR 50.46a is in Part V, "Non-applicabilities and requested exemptions."

The description and analysis of structures, systems, and components (SSCs) for the Aurora is included in Chapter 2, "Description and analysis of structures, systems, and components." Further, Chapter 2 provides the relevant Aurora SSC design bases, which are the characteristics of a system that ensure the safe operation of the reactor.

The purpose of this chapter is to document the methodology to transient analysis. These analyses consider a spectrum of events, ultimately showing how these events are bounded by a maximum credible accident (MCA). The results demonstrate an adequate plant response to challenging conditions, conformance with applicable regulations concerning SSC performance and postulated radiological consequences, and show that adequate protection of the public is expected during the plant lifecycle.

The transient analysis approach for the Aurora as outlined in this chapter is deterministic, drawing on analyses traditionally used in reactor licensing. It begins by considering the full range of potential challenges to safe reactor operation that might arise due to internal off-nominal events at various operating states, as informed by prior reactor experience and refined by considering the unique aspects of the Aurora design. These challenges are then grouped together based on their phenomenology, and the relative severity of their safety challenge is considered. Ultimately, the most challenging event deemed credible is identified and is considered the MCA.

Although deterministic, this approach to analyzing reactor safety was not performed by neglecting or minimizing probabilistic risk insights. To the contrary, physical and logical insights used in probabilistic risk assessment were used to focus deterministic analysis on those events which are physically possible and credible. The safety conclusions reached via the deterministic approach are consistent with those obtained from the probabilistic risk assessment, as described in Chapter 24, “Probabilistic risk assessment summary.”

5.1 Introduction

5.1.1 Background on maximum credible accident

The maximum credible accident has been a central feature of regulations from the earliest days and has influence in many of the methodologies in safety analysis today. It is useful to examine the fundamental concept instead of rote application of the LWR-specific outcomes of this methodology over decades, in order to consider accident selection for the Aurora. U.S. Nuclear Regulatory Commission (NRC) historians have described the origin of the MCA as follows [33]:

> Using their collective experience with evaluating earlier reactors, both the committee and the staff agreed that determining the maximum credible accident was the logical starting point… After fixing the maximum credible accident, the regulators assumed they could establish and appraise appropriate engineered design safeguards in conjunction with the site evaluation.

The term "maximum credible accident" acquired popularity around 1960, although the original Reactor Safeguard Committee had instituted a "worst case" accident scenario in the late 1940s. The worst case scenario was challenging for LWRs, since it would mean that most sites would have to be located hundreds of miles away from populated areas. On the other hand, assuming the analysis was correct, and the reactor could maintain safety in the worst credible case, any site could be acceptable.

The regulator decided that an accident was in the MCA category if it was caused by the one single equipment failure or operational error that would result in the most hazardous release of fission products; no other postulated credible accident could exceed those consequences. For LWRs, , the regulator postulated the MCA as the complete rupture of a major or large pipe resulting in complete loss of coolant, with consequent expansion of the coolant as flashing steam, meltdown of the fuel, and partial release of the fission product inventory to the containment atmosphere. This accident assumed a breach of the fuel cladding and reactor coolant boundary. This MCA for a light water resulted in core melt.

However, NUREG-0800, "Standard review plan for the review of safety analysis reports for nuclear power plants: LWR edition," Section 15.0.3, "Design basis accident radiological consequences of analyses for advanced light water reactors," issued March 2007, states the following:

> Although the loss-of-coolant (LOCA) is typically the maximum credible accident associated with the light-water reactor design, the applicant should consider other accident sequences of greater radiological consequence for the specific reactor designs selected by the applicants or for reasonably foreseeable future reactor designs…

It is clear that the intent is to provide reasonable assurance that the greatest potential radiological consequences of any credible event have been identified. The regulation does not require consideration of a core meltdown, stating only that meltdowns have "generally been assumed." When it incorporated the same text into design-oriented regulations (e.g., 10 CFR 50.34(a) and 10 CFR 52.79(a)), NRC stated that, "accident source terms and dose calculations currently primarily influence plant design requirements rather than siting," implicitly acknowledging the designer's ability to provide features that prevent or mitigate accidents[12]. The NRC Commission's "Policy Statement on the Regulation of Advanced Reactors"[13] also states that, "…the Commission expects that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions." Thus, it is reasonable to infer that the NRC would acknowledge that advanced reactors may be designed such that the probability of accidents yielding significant release of radioactivity is so remote that such accidents are not credible.

Although the LWR guidance is not applicable to the Aurora, it is useful to note that this approach is expanded in NUREG-0800 and Regulatory Guide (RG) 1.206, "Combined license applications for nuclear power plants", Revision 0, issued June 2007, and Revision 1, issued October 2018. The applicant is expected to determine bounding events for each category and type of event, resulting in a comprehensive analysis that down-selects from numerous possible events to a conservative number of limiting ones. As a result, a consistent through-line of regulatory expectations for transient analysis can be seen: a wide-range of safety challenges to the reactor should be considered to ensure that the safety response space of the system is well understood, and those credible events that most challenge the safety response should be characterized and presented to ensure acceptable safety performance is achieved by the design.

This is the general approach taken for the transient analysis of the Aurora design as presented in this chapter.

5.1.2 Design overview

Three important overarching characteristics of the Aurora should be kept in mind when evaluating its safety performance. The Aurora is a small reactor with all of the following characteristics:

- Small power density
- Low decay heat generation
- Small inventory of radionuclides

The specific system design features that provide these overarching characteristics will be briefly reviewed in this chapter; more detailed on the Aurora systems is in Chapter 2.

The Aurora is a small fast reactor that produces 4 megawatts thermal (MWth), using binary U-Zr metal fuel alloy (UZr). It uses alkali metal heat pipes, rather than a flowing coolant, to transport the heat generated by the fuel to the heat exchanger. The heat exchanger then transfers the heat to the supercritical carbon dioxide secondary system, which is the power conversion system. The fuel is shaped as an annular cylinder, with a heat pipe located in the annulus, and enclosed by a hexagonal steel can to form a reactor cell. Each reactor cell is somewhat analogous to a fuel assembly typically encountered for LWRs or sodium fast reactors, except that only a single fuel element is present in a single cell, whereas for an assembly, many fuel pins are bundled together into a single unit. The reactor cell also contains upper and lower

[12] 61 FR 65157, December 11, 1996

[13] 73 FR 60612, October 14, 2008

axial reflectors immediately above and below the fuel, and the heat pipe used to transport heat from the fuel to the secondary system. Figure 5-1 shows the design of a reactor cell.

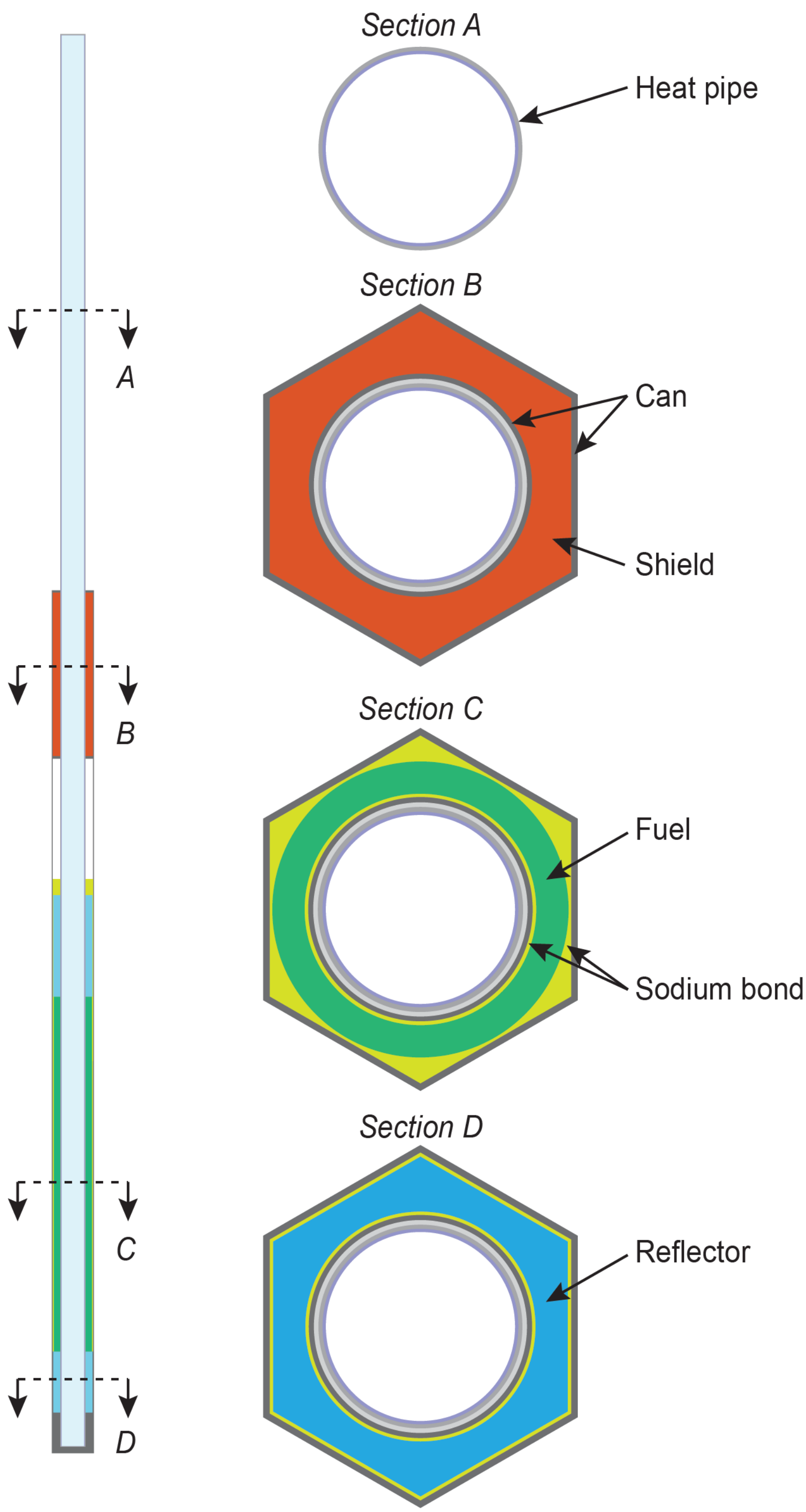

Figure 5-1: Cross-sectional views of a reactor cell

The reactor cells are arranged in a hexagonal lattice and surrounded by radial reflector material. A steel enclosure, referred to as the capsule, surrounds the radial reflector. The capsule is placed into another steel enclosure referred to as the module shell, and shielding material is placed in between the capsule and the module shell. The reactor core is dry, meaning no liquids are present outside the reactor cells. Only an argon backfill is present in the free space inside both the capsule and the module. Figure 5-2 shows a radial view of this arrangement.

{

}

Figure 5-2: Radial view of the reactor core layout and surrounding structures

The fast spectrum system has strong neutron leakage characteristics. Thermal expansion of the fuel leads to enhanced leakage in overpower and overtemperature transients, providing strong resistance to undesired power and temperature increases. Additionally, being a fast reactor, the reactivity swing over life from fuel depletion is relatively small. Subsequently, the necessary excess reactivity that must be provided by control drums is also small.

The design uses rotating control drums located outside the core in the reflector region to compensate for reactivity loss over the core lifetime due to fuel depletion. These rotate at a very slow nominal average speed of approximately 0.03 deg/day. The design also includes three shutdown rods located above the core that are used solely to shut the reactor down. The rods are fully withdrawn during normal operations. The three shutdown rods are redundant; just one rod is needed to shut the reactor down at any temperature condition.

The heat pipes operate passively and are designed to move heat from the fuel to the heat exchanger system inside the capsule, which then transfers the heat to the secondary system. The reactor operates at a very low power density; there is a very small amount of heat produced per unit mass or volume of the core. This, combined with the thermal mass of the core and surrounding materials, as well as the relatively high thermal conductivity of the steel and metallic fuel, allows for effective radial and axial conduction of heat away from the fuel.

The core produces 4 MWth, which is far smaller than any commercial reactor in the U.S., and smaller even than some research reactors. The lower power of the core also leads to low decay heat production. For reference, 30 seconds after shutdown, the reactor is generating 153 kW of decay heat, 30 minutes after shutdown, the reactor is producing 65 kW of decay heat, one day after shutdown the reactor is producing 21 kW of decay heat, and 1 month after shutdown, the reactor is producing 7 kW of decay heat. This is generated in a core that contains nearly 29 tons of fuel, steel, and zirconium, and more than 60 tons of additional shielding, insulation, and structural steel in the module.

For comparison, one fuel assembly at Diablo Canyon produces approximately 4 times as much power as the entire Aurora core, and contains approximately 0.5 tons of fuel, cladding, and structural material (or about 0.6 tons when the assembly is immersed in water and the water mass is included).

The low total power of the reactor also leads to low burnup of fuel at the end of its design lifetime, which is 1 atom per cent (at.%) or less depending on core location.[14] This corresponds to a relatively small inventory of fission products that are contained in a relatively large mass of fuel. The fuel has demonstrated minimal release of fission products into the space surrounding the fuel below 1 at.% burnup, so the fuel matrix itself contains the vast majority of fission products generated. For context, the fission product inventory in the Aurora core after 20 years is less than 0.5% of the inventory generated in a 3,000 MWth pressurized water reactor (PWR) core halfway through one cycle.

Additionally, the reactor does not have any systems that are significantly pressurized. The heat pipes operate at a slight vacuum { }, the core region has a slight backfill of argon gas and operates at ambient pressure, and the fuel plenum is at only a slightly elevated pressure at the end of the core lifetime.

Altogether, these characteristics lead to a system with inherent reactivity controls and inherent and passive heat transport, all within multiple barriers at non-pressurized conditions in the core.

[14] This burnup is equivalent to a peak burnup of approximately 10 MWd/kg.

5.2 Principles

5.2.1 Safety principles

The safety principles of the Aurora are derived from IAEA Safety Standard, Specific Safety Requirements (SSR) 2/1, "Safety of nuclear power plants: design," Revision 1, issued February 2016. In order to ensure that nuclear power plants are operated, and activities are conducted so as to attain the highest standards of safety that can reasonably be achieved, the IAEA standard recommends that all of the following measures be taken into account:

- Control the radiation exposure of people and radioactive releases to the environment in operational states
- Restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source, spent nuclear fuel, radioactive waste or any other source of radiation at a nuclear power plant
- Mitigate the consequences of such events if they were to occur

Therefore, the safety principles of the Aurora are the following:

- Provide power with minimal risk to the public health and safety and the environment
- Restrict the likelihood and consequence of abnormal events by inherent, physical characteristics

5.2.2 Defense-in-depth principles

An important consideration during the design of a nuclear power plant is defense-in-depth. This concept is applied to ensure that independent layers of provisions are available so that if a failure were to occur, it would be detected and compensated for or corrected appropriately. Defense-in-depth is considered throughout the Aurora design, including all of the following:

- Small thermal power and low burnup of fuel results in limited available source term.
- Inherent reactivity feedback ensures reactor power is controlled during overpower or overtemperature events.
- Multiple barriers result in complicated success paths for fission product release.
- Robust passive design ensures adequate heat removal during off-nominal events.
- Arrangement of high-conductivity components ensures high thermal capacity.
- Atmospheric operation limits driving forces for release.

These are design implementations of defense-in-depth principles. Defense-in-depth was also applied to the analytical basis of the maximum credible accident, by incorporating risk insight from the PRA (Chapter 24) to add further conservatism to an already conservative event analysis methodology described further in this chapter.

5.3 Safety and operational goals

The safety and defense-in-depth principles describe the underlying safe design philosophy that drove the design of the Aurora. The safety goal provides a high-level target for evaluating the safe design and operation of the reactor, as guided by these principles.

The safety goal of the Aurora is to control the release of radionuclides to minimize the risk to the public and the environment. This goal is achieved by maintaining fuel integrity, which is the primary contributor to radionuclide release. Fuel integrity is maintained by ensuring the fuel stays below a safety limit of 1200 C, the temperature at which fuel melting occurs.

For investment protection and defense-in-depth purposes, the Aurora is operated with significant margin below the safety limit. This margin ensures that the system is not damaged during normal operations or off-normal events. The primary operational challenge is damage to the reactor cell cans; therefore, an operational goal is specified to maintain reactor cell can integrity. Cell can integrity is maintained by limiting the total time at temperatures greater than 720 C, a conservatively defined temperature where fuel-steel eutectic formation may begin, as described in Section 5.3.2.

Because the operational limit is substantially lower than the safety limit, the safety analysis modeling can be evaluated against the operational limit, and, if the operational limit is not exceeded, both the operational and safety goals are satisfied. The safety and operational goals, along with associated limits, are summarized in Table 5-1 and described in more detail in the following sections.

Table 5-1: Safety and operational goals and metrics for the Aurora

Type	Goal	Limit
Safety	Control release of radionuclides by maintaining fuel integrity	T_{fuel} < 1200 C
Operational	Maintain reactor cell can integrity by keeping fuel-steel temperatures within time-temperature limits	T_{fuel} < 720* C

*Onset of eutectic formation is conservatively defined to begin at 720, but is very slow if it occurs at all at these temperatures, and no cliff-edge effects occur

5.3.1 Safety goal: control release of radionuclides

The risk to the public and the environment is minimized by controlling dose. Dose is determined by the total amount of radionuclides released and the atmospheric dispersion parameters. Because the latter is influenced by conditions external to the Aurora, the safety goal of the Aurora is to control the release of radionuclides by maintaining fuel integrity.

The fuel matrix is the first barrier to the release of fission products. At low burnups (< 1 at.%), the vast majority of fission gases are retained within the fuel matrix. At higher burnups, pores created by fission gases begin to interconnect and provide release pathways into the plenum. This is discussed further in Chapter 2. The safety limit is set such that the fuel temperature remains below the melting (solidus) temperature of 1200 C. Fuel melt is to be avoided in the Aurora because the fuel significantly loses its capability to retain fission products upon melting. Because the melting (solidus) temperature is more than 400 C higher than the onset of eutectic formation, this safety limit is always bounded by the operational limit.

5.3.2 Operational goal: maintain reactor cell can integrity

As described in the previous section, preventing fuel melt ensures the retention of the vast majority of fission products within the fuel matrix. The small amount of fission products that are not retained within the fuel matrix, primarily fission gases and volatile fission products that escape into the plenum, are retained by the reactor cell cans. Because the quantity of radionuclides is so much smaller, preventing the release of these fission products is substantially less consequential than maintaining fuel integrity.

The integrity of reactor cell cans may be challenged by mechanical, thermal, and irradiation effects. The reliability of a barrier to the release of radioactive materials "depends on the inherent strength of the materials relative to the potentially applied loading" [34]. It is worth noting that the thickness of the cell cans is more than four times thicker than the cladding used in Mark-I, II, and III EBR-II fuel pins [35].

The similarity of the fuel and cell can thermal expansion and compliance of the fuel lead to negligible fuel-can mechanical interaction [27]. The main mechanisms for cell can rupture are related to plastic strain, which in a high burnup metal fuel system is due to an increase in gas pressure, not to fuel/metal mechanical interaction [34]. Contrary to fast reactors designed to achieve high burnup, plenum gas pressurization in the Aurora does not significantly stress the cell can, as pressure is expected to be less than 100 kPa.

Fluences in the cell cans are on the order of 10^{22} n/cm^2 ($E > 0.1$ MeV), and displacement damage is estimated at less than 15 dpa [36]. For comparison, in EBR-II, several hundred stainless-steel 316 (SS316)-clad U-10Zr fuel pins were irradiated beyond 10^{23} n/cm^2 ($E > 0.1$ MeV), and many pins were exposed to fast fluences greater than 1.5×10^{23} n/cm^2 without failure [35]. At the low fluences seen in the Aurora reactor, typical irradiation-induced effects, including neutron-induced swelling, are not expected to significantly degrade the performance of the cell cans [37].

The combination of thick cell can, low plenum gas pressurization, and limited irradiation effects ensure that mechanical failure does not occur. Therefore, the operational limit of interest for reactor cell can integrity is eutectic formation at the fuel-steel interface.

Eutectic effects between steel and fuel have been analyzed at length, primarily during the Integral Fast Reactor program. Effects caused by fuel-steel chemical interaction occur at elevated temperatures where interdiffusion occurs between the uranium component of the fuel and the stainless steel and begins to form a lower melting point eutectic. Depending on the local burnup and fuel positioning, the fuel may or may not swell enough to contact the reactor cell can. The present analysis conservatively assumes contact between the fuel and the reactor cell cans.

The defined operational limit accounts for the temperatures reached at the fuel-steel interface, as well as the time at those temperatures. The operational goal is satisfied if the cumulative eutectic formation does not breach the cell can.

A correlation developed by Argonne National Laboratory (ANL) conservatively shows that this process may begin at 720 C but progresses slowly at low temperatures. Operation at 720 C does not result in a cliff-edge effect, but for practical purposes, if the fuel-steel temperature remains below 720 C at all points during the event, no fuel-steel chemical interaction occurs, and the operational goal is satisfied. More detail on fuel-steel chemical interaction is provided in

Chapter 2. As described, eutectic may not form at 720 C in the Aurora design, and eutectic formation of interest may only exist if temperatures exceed 830 C for extended periods of time, but 720 C was defined as a conservative operating limit.

5.4 Analysis approach

The approach to safety analysis for the Aurora seeks to show how the safety and defense-in-depth principles are upheld by confirming the safety goal is satisfied during off-nominal events. The approach follows an event evaluation process that ensures a wide variety of possible challenges are considered, while ultimately focusing the analysis on the events of highest importance. It accomplishes this by applying a methodology to consider the range of potential challenges posed by possible events, grouping these events together into event categories based on similar phenomenology of challenge, identifying which events in a category are bounding, and focusing analysis on these bounding events to ultimately designate a single MCA. More formally, the event evaluation process applies the following steps to achieve both a wide-ranging yet ultimately focused analysis of the safety of the Aurora:

1. Perform a literature review to understand the historical context and past challenges considered for fission reactor systems, both those that have operated and those proposed. In the context of these past events considered, determine which events are applicable and credible for the Aurora, and what, if any, new events specific to the Aurora could exist.

2. Group these applicable and credible events together into event categories based on similar phenomenology of challenge to safety and identify the bounding events in each category that challenge safety. Review this set of bounding events to determine whether the bounding event in one category is also bounded by the bounding event in another category, to develop a final set of overarching bounding events.

3. Focus the safety analysis on this final set of bounding events, ultimately identifying the event that most challenges safety based on the single worst failure of an active component or worst single cause of common cause failures, which is then designated the MCA.

4. Show that the safety goal is satisfied for the MCA.

In essence, the event evaluation process funnels a large number of events and progressively screens, bounds, and analyzes events until reaching a single bounding event, which is designated as the MCA. Figure 5-3 presents a visual representation of this funnel.

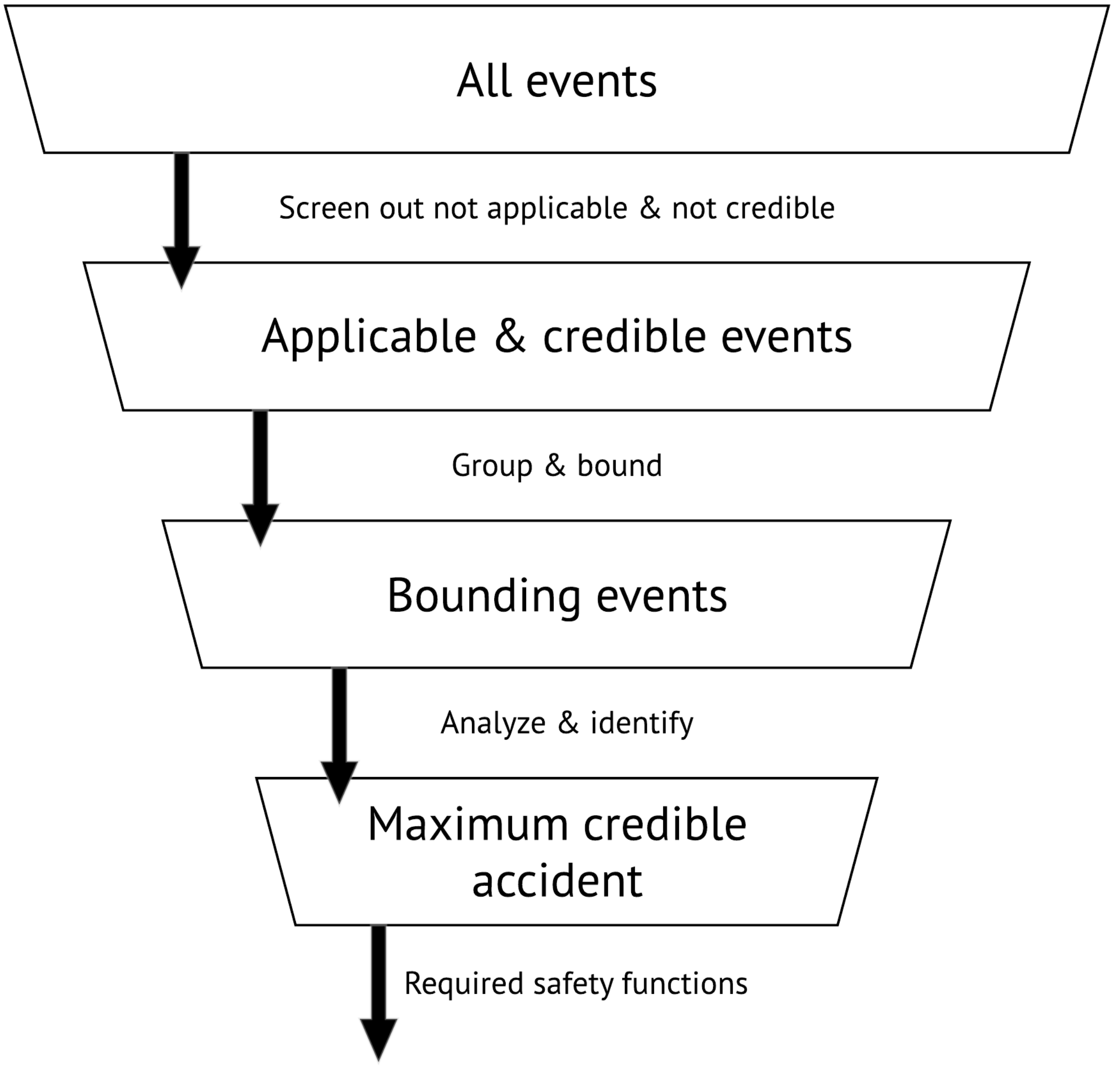

Figure 5-3: Visual representation of the event evaluation process

Step 1 in the event evaluation process is discussed in Section 5.5.1 of this chapter, where the historically-considered events from key references are presented and evaluated for applicability and credibility, and events unique to the Aurora are also discussed. The possible and credible events for the Aurora are then grouped together in step 2 into event categories based on similar challenge phenomenology, and bounding events are identified for each category and across categories, in Section 5.5.2. A detailed discussion focusing on the computational modeling and simulation of the identified bounding events, step 3, is then presented in Section 5.6. This discussion includes a description of the simulation model used and an evaluation of the results of the safety simulations to show satisfaction of the safety goal for the MCA (step 4). The Aurora design bases, which are system characteristics that ensure safe operation, are described as part of this analysis (these descriptions are repeated from their introduction in Chapter 2). Finally, the analysis is summarized in the context of the fundamental safety functions of a nuclear reactor.

5.5 Initiating event selection

5.5.1 Initial event identification and applicability and credibility review

The event evaluation process begins with an evaluation of operating experience and literature as well as utilizing phenomenological event categorization, as in NUREG 0800, in order to determine applicability and credibility, and ultimately to identify the maximum credible accident (MCA). Credibility in this sense is deterministic although risk insights are useful for adding defense-in-depth. This approach avoids excessive reliance on a PRA without significant operational data. Credibility is based on whether something is physically, fundamentally, or mechanistically possible. Credibility is also determined through the lens of the MCA. The historical basis for the MCA as described in 5.1.1 is to analyze any plausible single failure as well as any single initiating event to cause a common set of failures, even if extreme. Oklo has utilized PRA insight to add further defense-in-depth to this very conservative MCA analysis, as described in these sections.

While the Aurora is the first commercial reactor of its type, much insight can be gained by considering the safety challenges that have been analyzed and encountered by previous reactors. Accordingly, the development of the possible initiating events for the Aurora began with a systematic review of relevant operating experience and literature describing the history of reactor accident evaluations, including all of the following types of resources:

- Generic events to all nuclear reactors
- Metal-fueled fast reactor events and operating experience
- Light water reactor events
- Compact reactor operating experience and analytical methods
- Review expert opinion on similar conceptual designs

Of course, not all resources had the same relevance: the discussion presented here will focus on those sources that were deemed most useful in ultimately developing the list of possible initiating events for the Aurora. Two reactor categories stood out in their usefulness, whether in terms of technological similarity to the Aurora or their depth of analysis history and regulatory experience: metal-fueled fast spectrum reactors, and LWRs.

5.5.1.1 Metal-fueled fast spectrum reactors: the General Electric PRISM

The first category of reactor that has notable correspondence to the Aurora is that of metal fueled fast spectrum reactors. In this reactor category, focus is presently placed on the PRISM sodium-cooled fast reactor from General Electric (GE). Its development began in the 1980s as part of the Advanced Liquid Metal Reactor program funded by the U.S. Department of Energy. In 1987, GE developed what was known as a Preliminary Safety Information Document (PSID), which contained both a detailed system description as well as an accident analysis [38]. This document, submitted to the NRC in the late 1980s, is publicly available and is a useful resource for evaluating the transient analysis of a reactor that possesses similarities in terms of core (including fuel) materials and operating spectrum to the Aurora.

The PRISM PSID accident analysis grouped initiating events into the following categories:

- Reactivity insertion
- Undercooling
- Local faults
- Sodium spills
- Fuel handling and storage accidents
- Other

The events considered by the PRISM PSID in each of these categories are subsequently reviewed, and their applicability and credibility for the Aurora design are evaluated.

5.5.1.1.1 Reactivity insertion

Only one event is considered in the reactivity insertion category in the PRISM PSID: an uncontrolled single rod withdrawal at power. The PRISM reactor uses control rods to manage reactivity letdown compensation with fuel depletion, and thus the control rods are present in the core during power operations. The analyzed event has a single control rod (of the highest reactivity worth) withdrawing at the nominal rod movement speed at the beginning-of-cycle core configuration, starting with the reactor at 100% nominal power. The reactor response is to trip once 115% power is reached, at which point the other control rods (besides the malfunctioning rod) are inserted into the core and normal shutdown cooling commences.

The Aurora does not use rods for fuel depletion reactivity compensation, instead incorporating slow-moving control drums for this purpose. The Aurora uses shutdown rods, which are positioned fully withdrawn from the core during normal operation, to achieve reactor shutdown. Thus, because a rod removal event cannot cause a reactivity insertion at full power, the corresponding event for the Aurora is a control drum rotation malfunction. Conservatively, the control drum rotation malfunction event has been analyzed with the rotation speed of the drum at the maximum speed that the motors are capable of rotating, higher than necessary for properly compensating for fuel depletion. Chapter 2 discusses these design features in more detail.

Given that the control drums, even at maximum rotation speed, add far less reactivity per second to the Aurora core than the control rods of the PRISM reactor, the reactor power of the response of the Aurora is much more benign. The very low nominal reactivity worth that must be added per second on average is due to the very low power density, and thus very slow fuel depletion rate, of the Aurora.

5.5.1.1.2 Undercooling

In the undercooling event category, again only one event is analyzed for PRISM: loss of normal shutdown cooling. In this event, while the reactor is shut down and only decay heat is generated, the nominal heat removal pathway through the intermediate heat transport system is lost, and decay heat removal is subsequently accomplished only by the reactor vessel auxiliary cooling system (RVACS). The RVACS is a natural-circulation system where air from the environment is routed down through flow channels directly next to the reactor vessel, removing heat directly from the vessel wall and exhausting back to the environment.

This event proceeds similarly in the Aurora. If the nominal heat removal pathway through the power conversion system is lost, decay heat removal occurs off the surface of the module shell, which is the outermost surface of the structures that surround the Aurora core. The key difference for the Aurora compared with PRISM is that a specially designed RVACS cooling system is not necessary to remove the small amount of decay heat being generated; instead, the mere presence of some residual heat rejection due to natural convection off the module shell surface is sufficient.

5.5.1.1.3 Local faults

The local faults considered in the PRISM PSID are sub-categorized into: (1) Increased heat generation local faults, and (2) Reduced heat removal local faults. These local fault sub-categories mirror those presented in the PSID for the global faults discussed in 5.5.1.1.1 and 5.5.1.1.2. The specific local faults that are described in the increased heat generation sub-category are enrichment error (placing an assembly with a higher enrichment than desired into a wrong loading location, leading to greater heat generation than expected) and oversized fuel. The reduced heat removal local faults include flow blockages, as well as fuel element bond defects.

The Aurora core utilizes fuel with the same enrichment in every position: as such, an enrichment error due to fuel misloading is not possible. Oversized fuel in the Aurora is minimized by the quality assurance program applied to the fuel fabrication; nonetheless, the response to oversized fuel would be more benign due to the lower power densities in the Aurora relative to PRISM. Flow blockages due to foreign object intrusion are not possible in a heat pipe cooled reactor, since each heat pipe is a self-enclosed unit of working fluid evaporation and condensation that is significantly phenomenologically distinct from a large primary coolant system, and each heat pipe is tested before installation. Fuel element bond defects are also limited for the Aurora, since the fuel is large and the gaps that are occupied by the bond sodium are also large, helping to ensure effective bond distribution all around the fuel.

5.5.1.1.4 Sodium spills

The single event considered in PRISM's sodium spills category is a leak from the PRISM's primary sodium cold trap. The sodium cold trap contains 1,000 gallons (approximately 4,000 liters) of irradiated liquid sodium, which during this event is entirely released to the floor of the enclosing vault that holds the trap. Catch pans serve to help limit the initiation of sodium fires. The resulting dose at the site boundary is then calculated due to both sodium activation as well as a minor amount of circulating fission products and transuranics present.

The Aurora does not contain a sodium cold trap as it does not incorporate large volumes of flowing sodium requiring cleanup. The only sodium present in the Aurora is a relatively small volume of sodium used as a thermal bond in each reactor cell, on the order of a few liters per cell. This results in a much smaller total sodium inventory, as well as an inventory that is segmented into volumes enclosed by individual barriers. In addition, the reactor cells are located inside an inert environment of the capsule and module shell enclosures, providing additional barriers that provide protection from exposing the bond sodium to reactive materials. All of these design features of the Aurora serve to significantly differentiate the response of the Aurora to a reactor cell can sodium leak from that of the PRISM reactor's sodium cold trap spill event and render sodium spills not applicable and credible.

5.5.1.1.5 Fuel handling and storage accidents

For the PRISM fuel handling and storage accidents, the only event analysis presented is for a cover gas release from a fuel transfer cask. The PRISM reactor uses a sealed, shielded, passively-cooled cask to serve as the transport container for spent fuel following its removal from the reactor vessel. During a fuel removal operation, the transfer cask is sealed first to the reactor vessel fuel transfer port (to receive used fuel from the reactor vessel), and then subsequently to the adapter port at the fuel cycle facility (the PRISM plant design includes an onsite fuel processing facility). The analyzed event includes a failure of a total of five fuel pins located in one of the three fuel assemblies present in the cask, with these five failed pins releasing their fission gas and volatile fission product inventory to the transfer gas volume, as a result of the increased temperatures these pins experience due to the limited passive heat removal provided by the cask. This initial pin failure is then coupled with a failure of the transfer cask's gate valves to seal, resulting in a slight leakage of radioisotopes from the cask's inner volume to the room where the transfer cask is located.

The Aurora does not refuel since the initial fuel load is designed to last the entire operating life of the core. Fuel handling during initial loading does not pose any safety challenge associated with radionuclide release since no radionuclides are present before operation commences. Defueling at end of life will be accomplished via a transfer machine similar in concept to that of PRISM.

5.5.1.1.6 Other

In the "Other" category, the only event presented is a postulated cover gas release accident, which consists of the non-mechanistic failure of a pipe or valve that leads into the reactor vessel. The equilibrium concentration of noble gas fission products from two failed fuel pins that are assumed present in the reactor vessel cover gas is then released, as well as the noble gas fission products released from a single additional pin which is assumed to fail non-mechanistically at the time of the cover gas release.

Since the Aurora is not a sodium-cooled fast reactor, instead using heat pipes as the mechanism for removing heat from the fuel and transferring the heat to the secondary system, it does not possess a cover gas volume where fission gases from possible fuel pin failures would collect. The Aurora has several barriers to fission gas release:

- The fuel itself, which operates at a low enough burnup that most fission gases are retained in the fuel matrix.
- The reactor cell can, which is much thicker and sturdier than the very thin cladding that surrounds the PRISM fuel.
- The capsule, which is a stainless-steel container that encloses the reactor and reflector cells, shielding materials, and the heat exchanger system.
- The module shell, which is a stainless-steel container that encloses the capsule and additional shielding materials.

As a result, no single barrier failure would result in a release of radioactive material for the Aurora.

5.5.1.2 Light water reactors

The second category of reactors with a useful and relevant depth of safety analysis experience is that of large LWRs. While many of the characteristics of LWRs are significantly different from those present in the Aurora, the extensive operating and regulatory experience gained with LWRs, together with the fact that LWRs and the Aurora are both nuclear fission systems, means that much can still be learned about reactor safety challenges to the Aurora by reviewing some of the more relevant events considered for large LWRs.

An excellent reference for the transient events considered in large LWR safety analysis is Chapter 15, "Transient and accident analysis," of NUREG-0800. NUREG-0800 presents a selection of events that LWRs are expected to analyze as part of their FSAR. NUREG-0800 groups events that challenge the safety of LWRs into the following seven categories based on the type of challenge they present to the plant:

1. Increase in heat removal by the secondary system
2. Decrease in heat removal by the secondary system
3. Decrease in reactor coolant system (RCS) flow rate
4. Reactivity and power distribution anomalies
5. Increase in reactor coolant inventory
6. Decrease in reactor coolant inventory
7. Radioactive release from a subsystem or component

Some of these categories are more relevant to the Aurora than others: for example, since both the Aurora and LWRs are fission chain reacting systems, the safety challenges associated with reactivity and power distribution anomalies in LWRs can be considered to determine whether the Aurora will be challenged by similar phenomena. Some categories, however, are of less direct relevance to the Aurora, namely those that deal with an increase or decrease in the reactor coolant inventory: since the Aurora is cooled by an array of sealed, independent heat pipes, it is not susceptible to changes in the primary system reactor coolant inventory; indeed, the very concept of a singular 'primary system' is not applicable. In the following sections, the events in the more relevant categories are reviewed to illustrate the types of challenges typically considered for LWR safety analysis that have relevance to the Aurora. Events for PWRs are discussed since these reactors, like the Aurora, have a secondary system with a turbine for generating useful work from heat energy; in contrast, in boiling water reactors (BWRs), the primary coolant system directly turns the turbine to generate electricity.

5.5.1.2.1 Increase in heat removal by the secondary system

Events that fall into the "increase in heat removal by the secondary system" for PWRs (as presented in NUREG-0800) include any of the following:

- Decrease in feedwater temperature
- Increase in feedwater flow

- Increase in steam flow
- Inadvertent opening of a steam generator relief or safety valve
- Steam system piping failure

5.5.1.2.1.1 PWR event description

The decrease in feedwater temperature event for a PWR typically occurs when a feedwater heater is bypassed or inoperable, such that the feedwater returned to the steam generator is at a lower temperature than expected. The increase in feedwater flow event can be caused by the full opening of a feedwater control valve due to a control system malfunction or operator error. The decrease in feedwater temperature event typically bounds the increase in feedwater flow event in terms of the amount of increased heat removal by the secondary system from the primary system.

An increase in steam flow event can be caused by excessive turbine loading from an operator, or a malfunction in the steam dump control or turbine speed control. The effect of these secondary-side events on the primary system is to cause fluid of lower temperature to enter the reactor core than expected, causing a reactivity insertion and associated power increase due to the negative net temperature reactivity coefficient these designs operate with.

The inadvertent opening of a steam generator relief or safety valve causes a depressurization in the secondary system, which causes a short-lived increase in heat removal from the primary coolant system. A steam system piping failure also similarly causes a short-term increase in heat removal from the primary coolant system, but to a greater degree than in the inadvertent opening of a steam generator relief or safety valve. In contrast to the first three “increase in heat removal by the secondary system” events, these events have a short period of highly increased heat removal, followed by a reduced amount of heat removal over a longer timeframe. However, the initial high heat removal time period dominates the event response of the PWR system, hence why these events are placed in the “increase in heat removal by the secondary system” category.

5.5.1.2.1.2 Aurora analysis

The Aurora does possess a secondary system for heat removal, though it is very different from those present for PWRs. However, while the Aurora may thus experience events in the “increase in heat removal by the secondary system” category, the impact of these events on the safety of the Aurora is markedly different that the impact for PWRs. This is due to the difference in operating neutron spectrum between the two designs, as well as the difference in how heat is removed from the fuel.

In a PWR, the primary coolant (high pressure water) serves two roles: (1) to remove heat from the fuel (via convective heat removal off the fuel cladding surface), and (2) to moderate (slow down) the neutrons created by the fission chain reaction in the nuclear fuel. As the moderator density increases, its effectiveness at slowing down neutrons also increases, resulting in more thermal neutrons and increasing the number of fissions that result from these thermal neutrons reentering the fuel and getting absorbed. The increase in fissions means more heat is generated by the fuel; the reactor power increases as a result of the moderator temperature decreasing; the reactor thus possesses a negative moderator temperature coefficient of reactivity. The increase in power with reduced coolant temperature drives the transient response of the system to the event; while the lower temperature of the primary system coolant does very briefly increase the amount of heat removal from the fuel, this increased heat removal is outweighed by the increased power produced by the fuel due to the increased neutron moderation. As such, the temperature of the fuel increases during this event, and a challenge to safety exists for the PWR.

However, the Aurora operates in a fast neutron spectrum: neutrons born in fission slow down only slightly before causing fission in the fuel. This is due to the absence of moderating material in the Aurora core: unlike an LWR, which has an abundance of light (low atomic mass) elements present in its core, mainly in the form of hydrogen in water, the Aurora core is chiefly composed of metals, including the U-10Zr fuel, the stainless-steel structural materials, and the sodium bond. Additionally, the Aurora has no primary coolant system: instead, a series of heat pipes (one per reactor cell) passively transfers heat from the fuel to the heat exchanger system via evaporation and condensation of a working fluid sealed inside each heat pipe. The working fluid in the heat pipe is almost entirely vapor, save for a very small amount of liquid in the porous metal wick located in the inside of the heat pipe. As the very low density heat pipe working fluid does not significantly affect neutrons present in the system, the increase in evaporation and condensation rates that are caused by an increase in heat removal by the secondary system essentially have no effect on the reactor power.

As a result, the primary effect of an increase in heat removal by the secondary system is to remove more heat from the fuel than is being generated by fission, which causes a reduction in fuel temperature. A reduction in fuel temperature is an improved state for the safety of the system; it is not safety-challenging, in contrast to the response for PWR systems[15].

In summary, for the Aurora, events involving an increase in heat removal by the secondary system are not challenging for the safety of the system, which stands in contrast to the events considered for PWRs.

5.5.1.2.2 Decrease in heat removal by the secondary system

Events that fall into the "decrease in heat removal by the secondary system" for PWRs in NUREG-0800 include any of the following:

- Loss of external load with and without loss of nonemergency alternating current (AC) power
- Turbine trip
- Loss of condenser vacuum
- Main steam isolation valve closure
- Steam pressure regulator failure
- Loss of normal feedwater flow

[15] While a reduction in fuel temperature in the Aurora will subsequently introduce positive reactivity, the reactor must ultimately stabilize at the same average temperature as before the transient, since it was at this initial average temperature that the system was operating at steady-state with $k_{eff} = 1$.

- Feedwater system pipe break inside and outside containment

5.5.1.2.2.1 PWR event description

In the loss of external load event, a postulated electrical disturbance significantly reduces the load placed on the generator. Immediate fast closure of turbine control values results, which causes a reduction in steam flow, in turn resulting in higher secondary-side temperatures and thus higher primary coolant temperatures. Two versions of this event are considered: one with nonemergency offsite AC power available, and one with a loss of offsite nonemergency AC power. If AC power is available, the station auxiliary systems (in particular, the reactor coolant pumps) can continue to operate; if nonemergency AC power is lost, the reactor coolant pumps will trip, resulting in a primary coolant system flow coastdown in parallel with the turbine trip.

A PWR turbine trip leads to a similar situation as a loss of external load with continued AC power, except that turbine stop valves are closed instead of turbine control valves; the turbine stop valves close more quickly than turbine control valves, which causes a more bounding transient. The loss of condenser vacuum is an event that can cause a turbine trip. If steam dump to the condenser is not included in the pure turbine trip analysis, then the loss of condenser vacuum event is equivalent to the turbine trip event analysis. The closure of main steam isolation valves also causes a turbine trip and is bounded by the turbine trip analysis, as is the failure of a steam pressure regulator.

A loss of normal feedwater flow may come from a number of possible sources, including pump failures, valve malfunctions, or loss of offsite power. Loss of feedwater flow decreases the amount of heat removed from the secondary system, which results in a reduction of heat removal from the primary coolant system.

A feedwater system pipe break results in a loss of secondary coolant inventory, and depending on the size of the break, may result in either a reactor coolant system cooldown (due to the energy discharged by the break) or a reactor coolant system heatup. For the case where the primary system experiences cooldown, this event is equivalent to the “steam system piping failure” considered in Section 5.5.1.2.1, so only the case where the break causes a reactor coolant system heatup is analyzed in the “decrease in heat removal by the secondary system” category. This event is challenging for PWRs, since in addition to the reduction in feedwater flow (which causes temperatures in both the secondary and primary systems to rise), the fluid inventory in the secondary system also drops, meaning that the system is not available for decay heat removal.

5.5.1.2.2.2 Aurora analysis

The initiating event of loss of external load is applicable to the Aurora, as the Aurora’s secondary system includes a turbine attached to a generator for the production of electricity. However, the phenomenology following the initial event would differ: the Aurora’s secondary system moves to a turbine bypass heat removal mode if external load decreases. In the turbine bypass mode, heat removed from the heat pipes by the heat exchanger system is rejected to the ultimate heat sink without passing through the turbine. Additionally, since the Aurora removes heat from the fuel via passive-acting heat pipes and not a primary coolant system, no primary pump coastdown would occur, and furthermore, the Aurora does not rely on offsite AC power at all. The turbine bypass valves employed by the Aurora are the same whether the turbine trip occurs due to a loss of external load or another reason, so there is no differentiation made in the Aurora’s safety analysis between a loss of external load event and a turbine trip event.

The Aurora's secondary system rejects its waste heat to the outside environment via an air-cooled radiator or cooler, also known as the ultimate heat sink (UHS). A malfunction in this radiator will result in a reduced amount of heat rejected from the secondary system to the outdoor air, and as temperatures rise in the secondary system in response to this reduction in heat rejection, a reduced amount of heat will be removed from the heat exchanger system that is coupled to the heat pipes, and as such a reduced amount of heat will be removed from the reactor system. A similar situation will occur if the flow rate of the secondary system fluid decreases due to a malfunction of the pump.

These events will be bounded by a pipe break event in the secondary system, both in terms of the speed at which heat removal decreases, and in terms of the thermal mass present in the system that is available to receive the heat generated in the core. In contrast to the event categorization employed for PWRs, in the Aurora, any secondary system pipe break (regardless of size) is considered a decrease in heat removal by the secondary system. This is because, as discussed in Section 5.5.1.2.1.2, the phenomenology of a large secondary system pipe break for a PWR leads to a short-time increase in heat removal that dominates the transient response due to the negative moderator temperature coefficient. For any pipe break in the Aurora, this short-time increase in heat removal only serves to reduce fuel temperature slightly prior to the decrease in heat removal dominating the transient response. Secondary system pipe break events are best described for the Aurora as a decrease in heat removal by the secondary system, as it is this decrease in heat removal that dominates the transient response.

5.5.1.2.3 Decrease in reactor coolant system flow rate

The events that cause a decrease in the reactor coolant system flow rate in PWRs are separated into the following categories in NUREG-0800:

- Partial loss of reactor coolant flow
- Complete loss of reactor coolant flow
- Reactor coolant pump rotor seizure
- Reactor coolant pump shaft break

5.5.1.2.3.1 PWR event description

A partial loss of reactor coolant flow may be caused by a mechanical or electrical failure in a reactor coolant pump motor, a disruption of electrical power to the motor, or a motor trip caused by electrical anomalies in this motor power supply. A complete loss of reactor coolant flow necessarily requires a disruption to all reactor coolant pumps and may be caused by a simultaneous loss of electrical power to all pump motors. For the partial loss of reactor coolant flow, analysis is performed with and without the effects of a loss of offsite power, which is considered a potential consequence of the event due to the disruption of the electrical grid that results from the event. The primary impact of this loss of offsite power is the coastdown of the other, operating reactor coolant pumps once the loss of offsite power occurs following the initial partial loss of reactor coolant flow. In the complete loss of reactor coolant flow event, the coastdown of all reactor coolant pumps occurs simultaneously. Thus, the primary difference between the partial loss of reactor coolant flow event (with subsequent loss of offsite power) and the complete loss of reactor coolant flow event is the timing of the pump coastdowns.

The reactor coolant pump rotor seizure and shaft break events are considered more severe than the complete loss of reactor coolant flow event, even though the seizure and shaft break only affect a single reactor coolant pump. This is because the coolant flow through the loop of the affected pump is significantly and immediately reduced, relative to the more gradual coastdown seen for pump motor disruption events. The rotor seizure event causes a greater initial reduction in coolant flow than the shaft break event, but the shaft break event permits a greater reverse flow through the affected loop later in the transient. In both cases, the immediate flow reduction results in significantly increased primary coolant temperatures relative to the partial and complete loss of flow events.

5.5.1.2.3.2 Aurora analysis

The Aurora does not use a forced circulation coolant loop to remove heat from its nuclear fuel. A series of self-contained, passively-acting, independent heat pipes serves as the heat transfer pathway for removing the heat generated in the fuel. These heat pipes move heat very efficiently and can be described as "thermal superconductors." As such, the phenomenology of heat removal is very different. There are no concerns with flow reversal, pump coastdowns, and the complex interactions that occur as a result in PWR systems, as in the Aurora each heat pipe automatically matches the applied heat flux in its evaporator region to the removed heat flux in its condenser region.

For the Aurora, the category of "decrease of reactor coolant system flow rate" cannot thus retain the same nomenclature. Instead, the idea must be generalized somewhat, as the effect of interest is that heat removal from the fuel by the most direct method is being reduced. The designation of "decrease of heat removal by the heat pipes" is used instead.

In practice, such an event might occur for the Aurora is if a heat pipe wall were to fail; if the heat pipe ceases to be a sealed volume inside which the working fluid can evaporate and condense, it is no longer able to efficiently move heat in this way. The failure of a heat pipe wall is expected to be very rare, since the heat pipe's interior volume operates at low pressure (sub-atmospheric). Additionally, since each heat pipe is independent, a global reduction in the ability of all heat pipes to move heat is not considered credible. Cascade failures have been shown in Oklo analysis not to be credible, as the heat pipes in the cells that surround the failed heat pipe can accommodate removing the additional heat without any issue. As such, the only reasonable event in the decrease in heat removal by the heat pipes event category is a local fault where a single heat pipe experiences an enclosure failure.

5.5.1.2.4 Reactivity and power distribution anomalies

Events considered as part of the reactivity and power distribution anomalies category in NUREG-0800 for PWRs include the following:

- Uncontrolled control rod assembly bank withdrawal from a subcritical or low power startup condition
- Uncontrolled control rod assembly bank withdrawal at power
- Control rod misoperation (system malfunction or operator error)
- Startup of an inactive loop or recirculation loop at an incorrect temperature
- Inadvertent decrease in boron concentration in the reactor coolant system

- Inadvertent loading and operation of a fuel assembly in an improper position
- Spectrum of rod ejection accidents

5.5.1.2.4.1 PWR event analysis

PWRs use a combination of dissolved boric acid in the reactor coolant (also known as chemical shim) and occasional control rod assembly bank movements to manage reactivity letdown with fuel depletion. Accordingly, most of the events in the "reactivity and power distribution anomalies" category feature occasions where these control methods remove more reactivity than is desired.

The uncontrolled control rod assembly bank withdrawal from a subcritical or low power startup condition occurs when two rod banks with the maximum combined worth are withdrawn at the same time at the maximum withdrawal rate of the rod drive system while the reactor is critical and operating at very low power (relative power fraction of 10^{-9}). The reactor coolant system is assumed operating, but no steam generation is occurring on the secondary side since essentially no sensible heat is being added to the reactor coolant at such a low power. No loss of offsite power associated with the event is assumed to occur, since the plant is not providing power to the grid under this condition.

The uncontrolled control rod assembly bank withdrawal at power event similarly involves a removal of the two control rod banks with the combined maximum reactivity worth resulting in a positive reactivity insertion, but with some key differences from the low-power case. The largest difference is that, at power, the peak fuel temperature is much higher than the low-power case. Accordingly, temperatures in the reactor coolant are also greater. Additionally, steam is generated in the secondary system by the heat transfer from the reactor coolant at power. These differences result in a very high sensitivity of the transient response to the rate of reactivity insertion, the assumed magnitude of the reactivity feedback coefficients, and the operating power at the time of the event, due to the interplay between the heat generation in the poorly-conducting oxide fuel, the heat removal from the fuel cladding via convection of the flowing primary system coolant, and heat removal from the reactor coolant by the secondary system. The safety metric for the evaluation of these events is typically the minimum departure from nucleate boiling ratio (minimum DNBR, or MDNBR), which describes how close the coolant on the surface of the fuel rods is to reaching the condition where heat transfer from this surface rapidly decreases, which in turn further causes significant increases in cladding and fuel temperature leading to fuel failure and radioactive material release. This makes reaching a departure from nuclear boiling a cliff edge effect where significant damage occurs when a DNBR of one is reached.

5.5.1.2.4.2 Aurora analysis

In contrast, the Aurora does not incorporate a flowing convective liquid as the primary heat removal mechanism from its fuel. As such, the significant cliff edge effect associated with the DNBR is not applicable. For the Aurora, fuel temperature is generally the safety metric of interest. The phenomena associated with fuel temperature changes is much simpler than for calculating MDNBR. No complicated turbulence or boiling models are required; instead, the safety response of the Aurora can be characterized by a simple heat balance driven by fission or decay heat generation in the fuel, and heat conduction away from the fuel. This simpler phenomenology also leads to a more linear response in terms of the sensitivity of the safety metric to the input conditions applied or assumed. The most limiting inputs can generally be determined by inspection: larger reactivity insertion rates at the highest operating power are the most limiting events in the reactivity anomalies category for the Aurora. This is in contrast to the need to search over all possible combinations of initial operating power, reactivity insertion rate, and reactivity coefficient magnitude to determine the most limiting response in a PWR.

The Aurora does not incorporate control rods or chemical shim for reactivity letdown with fuel depletion, instead using three slowly rotating control drums for this purpose. The Aurora incorporates three shutdown rods that are inserted to achieve reactor shutdown. The shutdown rods are fully withdrawn and suspended above the core during normal operation. Only a single rod is required to achieve shutdown at any temperature condition. The shutdown rods are withdrawn, and the control drums are rotated to manage the rise to full operating temperature.

As a result, applicable events for at-power uncontrolled reactivity insertions in the Aurora involve malfunctions in the rotation of the control drums. The fission heat generation rate is the key driver of the system response and strongly influences the fuel temperature, which is the primary safety metric of interest. The fuel temperature is highest at full operating power. Therefore, low power events are bounded by full power events.

Control rod mis-operation events in PWRs involve one or more rods moving or displaced from normal or allowed control bank positions; this can involve events such as dropped rods and rods left behind when inserting or withdrawing banks. The Aurora only has three shutdown rods, and rod withdrawal actions are only taken to perform reactor startup. Each rod is withdrawn individually during startup. If a rod gets left behind, the startup process cannot be completed. If a rod moves too far in a step during startup removal steps, the reactor temperature will increase more for that step than originally planned, but since this may only occur during reactor startup from zero power at cold temperatures, and due to the waiting period and the small maximum withdrawal step size, this reactor temperature increase is not challenging. Conversely, a single dropped rod simply results in a reactor shutdown for the Aurora.

The other events considered for PWRs have no analogs for the Aurora. The startup of a reactor coolant loop at an incorrect temperature cannot be analogized to the series of passively-acting heat pipes that operate as thermal superconductors to remove heat from the fuel. The Aurora does not use chemical shim for reactivity letdown management, so there is no analog to an inadvertent decrease in the boron concentration in the reactor coolant system of a PWR. The Aurora operates with all reactor cells at the same enrichment, so an inadvertent loading and operation of a fuel assembly in an improper position cannot occur. And finally, the Aurora does not use control rods for reactivity holddown and does not possess a high-pressure in-core environment, so there is no analog to the range of rod ejection accidents considered for PWRs.

5.5.1.2.5 Increase in reactor coolant inventory, decrease in reactor coolant inventory, and radioactive release from a subsystem or component

The last three event categories presented in NUREG-0800 are: (1) increase in reactor coolant inventory; (2) decrease in reactor coolant inventory; and (3) radioactive release from a subsystem or component. The first two event categories have no events that are relevant for the Aurora. The Aurora has no reactor coolant system, and as such is not susceptible to increases or decreases in reactor coolant inventory.

Events included in the “radioactive release from a subsystem or component” category include releases of effluents from either gaseous or liquid waste management systems, releases of effluents from storage tanks, and fuel handling accidents.

The Aurora has no gaseous or liquid waste management systems. Indeed, the only circulating fluids present for the Aurora are the power conversion system working fluid and the air in the building. The activation of these fluids is minimal: the entire activity of the secondary system, or the building air, is well below the limits of 10 CFR Part 20, "Standards for protection against radiation" (see Chapter 3, "Radioactive materials to be produced in operation"). As such, none of these events are relevant for the Aurora.

The Aurora does not perform refueling operations during its operating lifetime. The fuel loaded at the beginning of life is designed to last the life of the core. Accordingly, fuel handling accidents like those analyzed for LWRs, where a single assembly is assumed to drop such that the cladding of every fuel pin is breached and the radiological inventory present in the gap of each pin is subsequently released, are not directly applicable. For final defueling, reactor cells are removed from the core independently, and multiple concurrent failures would have to occur for significant radionuclide release to occur for the Aurora during defueling: the fuel itself (where the majority of radionuclides are retained), the reactor cell can, and the defueling machine would all need to be compromised for this to occur, which is not credible.

5.5.2 Identified Aurora applicable and credible events

As a result of the first step of the event evaluation process, which included both a broad examination of events that challenged reactor safety for past and proposed reactors as well as a consideration of the unique aspects of the Aurora, the following set of event categories was developed. Each Aurora event category groups applicable and credible events together based on a similar phenomenological challenge. These event categories are analogous to those presented in NUREG-0800, adjusted for the unique design features of the Aurora. The Aurora event categories are shown in Table 5-2, where they are compared to their analogs from NUREG-0800

Table 5-2: Comparison of NUREG-0800 event categories to those identified for the Aurora

NUREG-0800 group	Aurora group
1. Increase in heat removal by the secondary system	1. Increase in heat removal by the secondary system
2. Decrease in heat removal by the secondary system	2. Decrease in heat removal by the secondary system
3. Decrease in reactor coolant system flow rate	3. Decrease in heat removal by the heat pipes
4. Reactivity and power distribution anomalies	4. Reactivity anomalies
5. Increase in reactor coolant inventory	-
6. Decrease in reactor coolant inventory	-
7. Radioactive release from a subsystem or component	-

As step 2 in the Aurora event evaluation process, the applicable and credible events that were identified for the Aurora in each category are discussed in the subsequent sections, bounding events for each category are identified, and bounding events across categories are identified for further, detailed analysis.

5.5.2.1 Increase in heat removal by the secondary system

The applicable and credible events identified for the "increase in heat removal by the secondary system" category are shown in Table 5-3. Note that for the Aurora, the terms "secondary system" and "power conversion system" are synonymous.

Table 5-3: Events identified for the 'increase in heat removal by the secondary system' category

Initial event	System response	Result

UHS fan controller overspeed failure	UHS rejects more heat than desired Secondary system cold leg temperature decreases Fuel temperature decreases Power increases	Safety not challenged Bounded by fan controller underspeed failure
UHS controller fails to adjust for lower ambient air temperature	UHS rejects more heat than desired Secondary system cold leg temperature decreases Fuel temperature decreases Power increases	Safety not challenged, similar to UHS fan controller overspeed failure Bounded by controller failure to adjust for higher ambient air temperature

Since the Aurora's secondary system does not possess standalone heaters to reheat the cold leg return fluid to the heat exchanger, the only possible means of removing too much heat from the secondary system's working fluid is related to the UHS heat exchanger (the radiator) experiencing a malfunction. This malfunction can, broadly, take one of two forms: (1) a controller failure that causes an overspeed in the air cooling fan that drives the ambient air through the UHS heat exchanger, thereby removing more heat than necessary; or (2) a controller failure that fails to adjust the fan speed to accommodate a decrease in ambient air temperature. While the conditions that might cause the overcooling event are different, the location of the failure is the same, and so is the result: more heat is removed from the secondary system cold leg than desired.

As discussed in Section 5.5.1.2.1.2, overcooling events are not challenging for the safety of the Aurora, due to the operating neutron spectrum and the use of heat pipes to remove the heat generated by the fuel and transfer it to the heat exchanger system, which in turn transfers it to the secondary system working fluid. The primary challenge associated with overcooling events in LWRs is the positive reactivity insertion caused by the entrance of cooler primary system coolant into the core; this positive reactivity insertion occurs because the coolant also serves as the neutron moderator material in the core. This injection of lower temperature primary coolant causes a power increase in the fuel, which can lead to a reduced DNBR. This is because the coolant gets closer to reaching the point of departure of nuclear boiling, a cliff edge effect where significant fuel and cladding damage can result.

In the Aurora, no moderating material is present, and no large volume of primary system working fluid exists in the core. As additional heat is removed in an overcooling event, the fuel temperature initially decreases. Although there will be power oscillations due to the decrease in temperature, the temperature will not increase enough to be of concern. This is also true in the case of localized overcooling in the core. Since fuel temperature is itself the primary metric of interest for safety evaluations, the safety of the system is not challenged due to this temperature decrease. This remains true for all power operating conditions.

As such, overcooling events are less challenging than events in the 'decrease of heat removal by the secondary system' category. Focus is subsequently placed on the analysis of events in this bounding undercooling category for the Aurora.

Note that secondary system large pipe break events are not included in the 'increase of heat removal by the secondary system' category for the Aurora, since, for the spectral and heat removal reasons mentioned above, the short time period of overcooling is not challenging for the system, and instead the longer-term loss of heat removal drives the transient response. Accordingly, these events are placed into the 'decrease of heat removal by the secondary system' category.

5.5.2.2 Decrease in heat removal by the secondary system

The applicable and credible events identified for the "decrease in heat removal by the secondary system" category are shown in Table 5-4.

Table 5-4: Events identified for the 'decrease in heat removal by the secondary system' event category

Initial event	System response	Result
Turbine trip	Turbine bypass valve closes Heat rejected from secondary system through UHS at a reduced rate Fuel temperature initially increases, causing power to decrease. Fuel temperature returns to nominal while steady-state power is reduced to match new heat rejection rate.	Brief period of fuel temperature increase. Bounded by extended period of temperature increase associated with pipe break.
Pump trip	Flow coastdown slowly reduces heat removal Secondary system working fluid temperature increases Fuel temperature increases Power decreases Pump can be restarted in 30min or less, restoring cooling	Temperature heatup in the fuel bounded by heatup from secondary system pipe break, which is both more rapid and has a much longer recovery time
Ultimate heat sink malfunction	UHS rejects less heat than desired Secondary system cold leg temperature increases Fuel temperature increases Power decreases	Temperature heatup in the fuel bounded by heatup from secondary system pipe break, which is both more rapid and has a much longer recovery time
Safety valve actuation	Secondary system inventory decreases Heat removal by heat exchanger system from heat pipes is reduced Fuel temperature increases Power decreases	Similar to but bounded by small secondary system pipe break, which will cause a larger reduction in inventory and has a much longer recovery time
Pipe break	Secondary system inventory decreases Heat removal by secondary heat exchanger from heat pipes is briefly increased, but subsequently significantly reduced Fuel temperature briefly decreases, then increases Power briefly increases, then decreases	Can occur for various pipe and break sizes. Brief period of fuel temperature decrease followed by longer period of fuel temperature increase. Bounding fuel temperatures obtained during this event for this category.

Events that cause a decrease in heat removal from the heat pipes by the secondary system bound the fuel temperature increase from the events discussed in the "increase in heat removal by the secondary system" category. Reduced heat removal by the secondary system may occur due to the system taking the proper actions in response to external factors, such as loss of external load causing a turbine trip, or due to a malfunction or failure in the secondary system such as a pipe break.

While many different possible events may drive this reduction in heat removal by the secondary, they ultimately only differ in terms of the magnitude of the heat removal reduction that occurs, the time period that elapses to reach this full reduction in heat removal, and the duration of the reduction in heat removal. For example, for a representative configuration, following a turbine trip the heat removed by the secondary system drops quickly to about 75% of the full power value for the short time required to reset the turbine. The time required to reset the turbine is typically about 30 minutes given the small size of the turbomachinery. Minor pump malfunctions, such as an erroneous pump trip, allow for quick recovery back to the nominal operating condition, also approximately 30 minutes. For a pipe break, cooling will decrease to approximately zero of the full power value in a very short period of time, and the repair of the system could take an extended period of time, on the order of hours or days depending on the size of the break.

As such, the bounding event for this category is one where all cooling capability is lost nearly instantaneously, and quick recovery is not possible. The assumption of instantaneous and total loss of cooling is a very conservative assumption which does not allow for the coastdown of loss of cooling which is more realistic. These outcomes are characteristic of significant failures, such as a large break in the cold leg or hot leg piping of the secondary system. These are not analogous to small or large breaks in the primary system, which are of concern in light water reactors. Thus, the large pipe break is the bounding event for the "decrease in heat removal by the secondary system" category. This event is also referred to as a loss of heat sink. For conservatism, the small period of initial increase in heat removal following a pipe break is neglected in the analysis of the event.

5.5.2.3 Decrease in heat removal by the heat pipes

Only one event is identified as applicable and credible in the "decrease in heat removal by the heat pipes" category, as shown in Table 5-5.

Table 5-5: Events identified for the 'decrease in heat removal by the heat pipes' category

Initial event	System response	Result
Single heat pipe fails due to manufacturing defect	Fuel temperature in failed location increases	Fuel temperature increase bounded by the bounding event in the 'decrease in heat removal by the secondary system' event category

The core of the Aurora is not convectively cooled by a large volume of coolant circulating in a forced flow loop: instead, it is cooled by an array of heat pipes, with one heat pipe cooling each reactor cell. As such, no credible failure would challenge the heat removal capability of all heat pipes simultaneously; this multifold redundancy is one of the principal reasons for the selection of heat pipes as the mechanism for removing heat from the fuel. A realistic failure is then limited to the failure of a single heat pipe. The most credible failure mechanism is due to a weld failure that was not detected during initial manufacturing and testing as part of the quality assurance program applied to the heat pipes. This weld failure could cause the sub-atmospheric sealed interior volume of the heat pipe to be significantly exposed to the external atmosphere, resulting in a degradation in heat pipe performance, potentially including total loss of function.

Since this is a local failure limited to a single reactor cell, which provides about 1% of the heat transport from the reactor core, the consequences of this event are bounded by that of the limiting, global event in the "decrease in heat removal by the secondary system category", the large secondary system pipe break.

5.5.2.4 Reactivity anomalies

The events identified in the "reactivity anomalies" category are shown in Table 5-6.

Table 5-6: Events identified for the 'reactivity anomalies' event category

Initial event	System response	Result
Drum control failure resulting in a positive reactivity insertion	Reactor power increases Fuel temperature increases Negative fuel temperature feedback coefficient reduces reactor power back to steady-state level	Maximum physically possible rotation speed at full power results in bounding fuel temperatures obtained during this event for this category.
Drum control failure resulting in a negative reactivity insertion	Reactor power decreases Fuel temperature decreases Negative fuel temperature reactivity coefficient increases reactor power back to steady-state level	Fuel temperatures observed in this event are bounded by those in the positive reactivity insertion drum control failure event.
Rod withdrawal malfunction during low-temperature startup	Reactor power increases Fuel temperature increases Negative fuel temperature feedback coefficient reduces reactor back to zero power	Fuel temperatures observed in this event are bounded by those in the positive reactivity insertion drum control failure event, since this rod withdrawal event may only occur during startup from low temperatures.

The Aurora uses slow-rotating control drums to manage the reactivity letdown associated with fuel depletion. The control drums are the only method by which, during power operations, reactivity may be inserted into the core. Three shutdown rods are fully withdrawn during normal operation and positioned above the core and are inserted to shut down the reactor; insertion of only one out of three rods is needed to accomplish reactor shutdown under any operating condition.

The cylindrical control drums span the axial length of the core, providing a uniform axial profile for neutron absorption and precluding the creation of axial flux shape distortions associated with using rods for reactivity letdown. The interior volume of each drum cylinder is divided in two via an axial dividing plate, such that half of the drum consists of boron carbide absorber material and half consists of zirconium reflector material. Rotating the absorber half into the core introduces additional negative reactivity into the system; rotating the absorber half out of the core introduces positive reactivity. The critical configuration of the reactor has the drum absorbers rotated partially inward towards the core at the beginning of life. All three drums are rotated simultaneously over life to remove their absorbers from the core and introduce positive reactivity that compensates for the reactivity decrease associated with fuel depletion.

Accordingly, the drums are the system by which positive or negative reactivity may be added during power operations, and as such, are the mechanism by which this positive or negative reactivity may be added in an undesirable way in the case of a failure of the control system. Undesirable reactivity additions at power may be of either positive reactivity or negative reactivity: if the control system erroneously sends a signal to rotate the drum absorber inward, then negative reactivity is added to the system. Conversely, if the control system erroneously sends a signal to rotate the drum absorber outward faster than necessary to compensate for fuel depletion, positive reactivity is added to the system.

The nominal outward rotation of the drum absorbers is designed to accommodate the reactivity letdown due to fuel depletion, and since the Aurora experiences only a few hundred pcm decrease in reactivity over its 20 year operating life with a very linear profile. This corresponds to a very slow nominal average drum rotation rate of { }. The drum rotation motors are attached to transmissions that are geared to provide this slow rate on average (by applying slightly larger steps at discrete intervals), but are also designed to allow faster rates that are used for initial startup. The transmission is geared such that the maximum rate for drum insertion is 0.01 deg/sec, which is still very slow. At this rate, rotating the entire drum absorber 180 degrees out of the core starting from a position of full absorber insertion takes 5 hours.

The spectrum of reactivity anomalies that might be encountered in the Aurora while at power is therefore differentiated only by the rotation speed and direction of an undesired drum rotation. Faster drum rotation speeds, up to the mechanical maximum of 0.01 deg/sec in either direction, result in faster rates of either positive or negative reactivity insertion, depending on the direction of rotation. Adding negative reactivity via drum absorber insertion results in reduced fuel temperatures, and consequently a less challenging state. Thus, negative reactivity insertion events are bounded by positive reactivity insertion events. The most bounding positive reactivity insertion event occurs when all three drums rotate absorbers-outward at their mechanically limited maximum rate of 0.01 deg/sec with the reactor at full power and at the nominal operating temperature. This event results in the highest fuel temperatures encountered in the reactivity anomaly event category, and as such it is the bounding event in this category. Since the positive reactivity insertion event results in a reactor power increase, which then causes the fuel temperature increase, it is also referred to as a transient overpower.

The only unique reactivity anomaly event for the Aurora that cannot occur during full power operation is a shutdown rod withdrawal malfunction, which may occur during startup from low temperatures and zero power. The shutdown rods are positioned outside the core while operating at full power, and the control drums are used to startup the reactor at hot temperatures. However, since this event can only occur at low temperatures and zero power, and the maximum possible worth a single rod may add during withdrawal is limited by the maximum withdrawal step size, the potential fuel temperature increase associated with a rod withdrawal malfunction is not challenging when compared to the transient overpower event, which occurs at full operating temperature and full power. Thus, the transient overpower remains the bounding event in the 'reactivity anomalies' category.

Note that, unlike the similar category presented in NUREG-0800, events where power distribution anomalies occur are not included in the reactivity anomalies category for the Aurora. Significant power distribution anomalies are precluded by the Aurora's design and operating characteristics, including the small size of the Aurora core, its operation in the fast spectrum, and the very long mean free path of neutrons in the core. The Aurora shuts down if a single shutdown rod is inserted into the core, meaning that no power distribution anomaly results. As a result of these characteristics and the low reactivity worth of its drums, significant radial power asymmetries cannot be caused by the spurious rotation of a single control drum.

5.5.3 Summary of bounding events

All events in the increase in heat removal by the secondary system category are bounded by the events in the decrease in heat removal by the secondary system category. Additionally, the single event in the decrease in heat removal by the heat pipes category is bounded by the bounding event in the decrease in heat removal by the secondary system category. The bounding event in the decrease in heat removal by the secondary system category was identified as a large break in the secondary system piping. This event, also referred to as a loss of heat sink event, is the bounding event that covers all the event categories involving events initiated by heat balance irregularities.

The remaining event category, reactivity anomalies, includes events that are initiated by reactivity insertions (either positive or negative). The bounding event in this event category was identified as a drum rotation malfunction involving the outward rotation of the absorber of all three drums at the very slow but maximum possible speed allowable by the control drum motor and gearing at full power, resulting in a positive reactivity insertion. This bounding event is referred to as a transient overpower event.

Accordingly, the two overarching bounding events identified for the Aurora in step 2 of the event evaluation process are the transient overpower event (from the reactivity anomalies event category), and the loss of heat sink event (from the decrease in heat removal by the secondary system event category). Detailed analysis of these events in presented in Section 5.6, as part of step 3 in the Aurora event evaluation process.

5.6 Safety analysis

5.6.1 Bounding events summary

The prior sections identified two transients as the most extreme, or bounding, and credible internal events: the transient overpower event, and the loss of heat sink. These two events then may represent the MCA as the maximum credible accident. These events would be considered beyond design basis in traditional analysis. But to be holistic, defense-in-depth was added by considering insights from risk analysis. Insights from the PRA, described in Chapter 24, illuminated that the failure of one shutdown rod insertion has probability on the order of 10^{-6}, and the probability of failure of two shutdown rods has a probability on the order of 10^{-12}. The order of magnitude of failure of one shutdown rod merited analysis in the MCA. Although the reactor reaches a safe state even with the failure of two shutdown rods, for defense-in-depth, analysis of the two transients also included the failure to insert one shutdown rod.

Therefore, the two internal transients analyzed in the internal event safety analysis are:

1. The transient overpower (TOP), where all three control drums spuriously rotate their absorbers outward at the maximum speed and insert positive reactivity, in conjunction with a failure to insert one of three shutdown rods.

2. The loss of heat sink (LOHS), where a failure of the power conversion system (PCS) occurs in conjunction with a failure of flow bypass to the radiator, and a failure to insert one of three shutdown rods.

A brief overview of how these events proceed is presented in Section 5.6.1.1 and Section 5.6.1.2. These summary descriptions focus on the expected physical response of the system (by active, passive, or inherent means) to the initiating events, and as such, conservative assumptions or modeling approaches are not discussed in these short presentations. The conservative models developed to analyze these initiating events are presented in Section 5.6.2, and the design bases that describe the characteristics of the system that ensure the safe operation of the reactor during these events are also presented. The detailed results generated using these conservative models are presented in Section 5.6.3 and Section 5.6.4, the overall MCA is identified in Section 5.6.6, and the results are compared to fundamental safety functions discussed in Sections 5.6.11, 5.6.12, and 5.6.13, fulfilling steps 3 and 4 in the Aurora event evaluation process.

It is important to note that, as described in the results presented in Section 5.6.3 and Section 5.6.4, there are no credible bounding events that result in a release of radioactive material. Therefore, no analysis of a radiological release is described.

5.6.1.1 Transient overpower

The phenomenology of events that occur during the TOP follow the below order:

1. All three control drums begin to spuriously rotate their absorbers outward, inserting positive reactivity too quickly relative to reactivity letdown with fuel depletion, introducing an undesired positive reactivity insertion. The drums are assumed to rotate at the maximum speed.

2. Power increases in response to the reactivity insertion, which leads to higher temperatures in the core.

3. The fuel expands in response to heating up, introducing negative reactivity into the system via increased neutron leakage from the core.

4. An overtemperature condition is detected by thermocouples in the reactor trip system, sending a reactor trip signal to the shutdown rod system,

5. Two of three rods insert into the core, adding significant negative reactivity to shut the reactor down. Only one rod is needed to achieve this, but one failure is assumed in line with the MCA methodology with defense-in-depth.

6. Heat generation drops rapidly and becomes dominated by decay heat generation within seconds.

7. Heat is conducted through the fuel, reflectors, and steel, both radially and axially out of the core. As both the heat pipes and PCS continue to operate, decay heat generation can easily be managed, and heat is removed from the fuel effectively. Peak fuel temperature drops below steady-state operating conditions.

8. Significant margin is maintained below safety and operational limits, and no material or structural damage occurs.

5.6.1.2 Loss of heat sink

The phenomenology of events that occur during the LOHS follow the below order:

1. Heat is no longer removed through the PCS by any means, causing the system to heat up.

2. As the reactor heats up, the fuel expands which introduces negative reactivity, slowing the reaction down and thereby reducing the rate of increase of core temperature.

3. An overtemperature condition is detected by thermocouples in the reactor trip system, sending a reactor trip signal to the shutdown rod system.

4. Two of three rods insert into the core, adding significant negative reactivity to shut the reactor down. Only one rod is needed to achieve this, but one failure is assumed.

5. Heat generation drops rapidly and becomes dominated by decay heat generation within seconds.

6. Heat is conducted through the fuel, reflectors, and steel, both radially and axially out of the core. Heat pipes distribute heat throughout the system, including the shielding and structural materials surrounding the core. Fuel temperatures drop below steady-state operating conditions.

7. Decay heat generation in the fuel continues to exceed passive heat removal via conduction from the fuel to the outer surfaces of the shell and ultimately to the air in the reactor cavity. Fuel temperatures increase during this period.

8. This period ends when passive heat removal exceeds decay heat generation. Afterwards, fuel temperatures decrease.

9. Significant margin is maintained below safety and operational limits, and no material or structural damage occurs.

5.6.2 Model description

The models, parameters, and assumptions used in the transient analysis are described in this section. Throughout the safety analysis process, an emphasis has been placed on selecting conservative and bounding parameters. Each of the analysis assumptions and conservatisms are addressed in detail in this section, and a summary is shown in Table 5-7.

The computational analysis is conducted using ANSYS Mechanical. Each simulation is initialized from nominal steady-state conditions and starts with the initiating event. Some of the input parameters for the ANSYS calculation are generated using other computer codes; the particular code used to generate an input parameter is noted when discussing that input parameter in the sections that follow. Additional information on the usage of codes in this analysis process can be found in Section 5.6.9.

Table 5-7: Summary of assumptions in safety analysis and reason for conservatism

Topic	Assumption	Conservatism
Power	Highest power reactor cell in ring	Maximizes internal heat generation
Heat pipe temperature	All heat pipes set to highest power cell	Overpredicts initial temperatures
Heat transfer from shell	h = 7 W/m^2-K, T = 225 C	Underpredicts passive heat transfer to ultimate heat sink
	No radiative heat transfer	Underpredicts passive heat transfer to ultimate heat sink
Decay heat	Decay heat during timestep assumes initial value	Overpredicts decay heat generation throughout analysis
Reactivity feedback	No negative reactivity feedback effects	Overpredicts rate of power increase during TOP
Overpower heat generation	Power during timestep assumes end value	Overpredicts fission heat generation throughout active phase
Power conversion system	Instantaneous stop of rotating components	Underpredicts heat removal by PCS associated with flow coastdown
Shutdown rod insertion delay	10-second delay from trip setpoint to reactor trip	Overpredicts fission heat generation prior to trip
Fuel thermal conductivity	30% decrement due to burnup	Overpredicts thermal gradients in fuel
Cell-to-cell contact conductance	100 W/m^2-K assumes only radiative heat transfer	Overpredicts thermal gradients in module

5.6.2.1 Model overview and design bases

The simulation model seeks to describe the structures of the reactor module. Time-dependent heat generation, conduction, and removal are the physical phenomena included in the finite-element heat transfer analysis of these structures. The transient response of the Aurora to the two bounding initiating events can be accurately described by these phenomena with this geometry. Heat generation occurs in the fuel due to either fission reactions (while at power) or radioactive decay (after reactor trip). Conduction is the means by which heat generated in the fuel is transferred to other core structures: note that the heat pipes may be effectively modeled as thermal superconductors, that is, as solid conduction volumes with a very high equivalent thermal conductivity. More description of equivalent thermal conductivity is provided in 5.6.2.12.2. Heat removal occurs either via the heat exchanger system or by passive heat rejection via convection off the surface of the module. The reactor system and its components are described in Chapter 2.

Certain model parameters may be important to ensuring safety, and those parameters are codified as design bases. Design bases are used to ensure that the as-built Aurora design reflects the modeled parameters. The nomenclature surrounding the design bases are introduced in the gray box below, and the specific design bases related to each assumption and input are referenced in similar boxes in the relevant sub-sections that follow.

Introduction to design bases, design commitments, and programmatic controls

Design bases are the characteristics of a system or sub-system that ensure the safe operation of the Aurora reactor. Each design basis has one or more design commitments, which are the specific commitments made to ensure the design basis is met. Each design commitment has one or more programmatic controls that are used to verify that the commitment is met. See Chapter 2 for more details.

The central importance of the design bases is reflected in this chapter, as key modeling assumptions and inputs are directly connected to the design bases (and the resulting design commitments). This is done to ensure that the assumptions and inputs modeled accurately describe the as-built Aurora reactor. In each of the following sections about inputs and assumptions, the related design bases, design commitments, and programmatic controls are referenced. The description and analysis of structures, systems, and components section (Chapter 2) contains the full description of design bases and design commitments. The programmatic controls are described in more detail in the appropriate sections noted below.

The following abbreviations are used in the summaries:

- Design basis (DB)
- Design commitment (DC)
- Preoperational test (POT) (see Chapter 14)
- Startup test (SUT) (see Chapter 14)
- Inspections, tests, and analysis acceptance criteria (ITAAC) (see Part VI)
- Technical specification (TS) (see Part IV)

For example: a design basis (DB) for the shutdown rod system (SRS), the resulting design commitment (DC), and the required programmatic controls, would be listed as follows in the summary box:

DB.SRS.01 The shutdown rod system provides sufficient negative reactivity to achieve cold shutdown with insertion of one rod.

DC.SRS.01.A The worth of each shutdown rod will be greater than 1400 pcm, where 1400 pcm is greater than the total of: the reactivity worth associated with the temperature decrease from hot full power conditions to cold zero power conditions, and an additional margin of 500 pcm.

SUT.SRS.01.A1 and A2 (see Chapter 14)

5.6.2.2 Geometrical simplification

A simplified geometry is used to conduct the safety analysis. The module is reduced based on symmetry in order to decrease computational requirements. One-sixth of the reactor module is included in the model, with a zero heat flux boundary condition set on the boundaries that lie on the lines of symmetry. The location of the shutdown rod that would normally appear in this geometry was replaced with a reactor cell to conservatively overpredict the heat generation present in this reduced geometry. The geometry of the model is shown in Figure 5-4, Figure 5-5, and Figure 5-6.

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Figure 5-4: Isometric view (left) and zoomed isometric view (right) of the geometry modeled in ANSYS

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Figure 5-5: Side view of geometry modeled in ANSYS

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Figure 5-6: Top-down radial slice at core midplane of geometry modeled in ANSYS

Design bases are taken for each of the major systems in the reactor module to ensure that the as-built geometry corresponds to the as-designed geometry. These systems include the reactor system (referred to as RXS in the design basis summary boxes), heat exchanger system (HXS) and reactor enclosure system (RES). The appropriate design commitments and related programmatic controls are taken to ensure that this is properly verified. As a result, the simplified geometry modeled here can be considered representative of the as-built design. As described in the uncertainties (Section 5.6.7), the contact conductance between bodies is of low significance to the analysis. As a result, the primary design commitment taken for each design basis is to ensure that each component of the system is installed in the appropriate location. In addition, a design commitment is taken to ensure via a startup test that conduction and subsequent convection from the module shell can sufficiently cool the reactor core system as described in Section 5.6.2.5.

DB.RXS.04 The reactor core system provides a pathway to conduct heat from the fuel to the surrounding systems and ultimately to reject it to the environment.

DC.RXS.04.A The critical components of the reactor core system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.

SUT.RXS.04.A (see Chapter 14)

DC.RXS.04.B The reactor core system can be cooled by conduction through the surrounding systems (reflector system, shielding system, heat exchanger system, and reactor enclosure system) and subsequent convection from the module shell after shutdown.

SUT.RXS.04.B

DB.RXS.05 The reflector system provides a pathway to conduct heat from the reactor core system to the surrounding systems and ultimately to reject it to the environment.

DC.RXS.05.A The critical components of the reflector system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.

POT.RXS.05.A (see Chapter 14)

DB.RXS.06 The shielding system provides a conduction pathway to conduct heat from the reactor core system and reflector system to the surrounding systems and ultimately to reject it to the environment.

DC.RXS.06.A The critical components of the shielding system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.

POT.RXS.06.A (see Chapter 14)

SUT.RXS.06.A1 and A2

DB.RES.01 The reactor enclosure system provides a pathway to conduct heat away from the systems inside it and to reject it to the environment.

DC.RES.01.A The critical components of the reactor enclosure system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.

POT.RES.01.A1 and A2 (see Chapter 14)

SUT.RES.01.A

DB.HXS.01 The heat exchanger system provides a pathway to conduct heat from the heat pipes of the reactor core system to the surrounding systems and ultimately to reject it to the environment.

DC.HXS.01.A The critical components of the heat exchanger system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.

SUT.HXS.01.A (see Chapter 14)

5.6.2.3 Power assumptions

The radial power distribution, calculated by Serpent, is used to define the peaking factors applied to each reactor cell in the ANSYS model. Power is input as a volumetric heat generation rate within the fuel component of each reactor cell and accounts for the axial power shape. Steady-state heat generation rates are axially discretized in ANSYS using a cosine-shaped axial power profile, which is conservative (overpredicting the actual per-reactor cell power by approximately 2%) compared to the expected axial power peaking distribution as calculated by Serpent, as shown in Figure 5-7. The heat generation rate within each reactor cell is assumed to equal the highest-power reactor cell in its ring, which overpredicts the total heat generation in the model. This is shown in Figure 5-8. For the loss of heat sink event, the reactor power is taken as 102% of the rated power, which is a common assumption in NUREG-0800.

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Figure 5-7: Axial power profile used in ANSYS compared to axial power profile calculated by Serpent

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Figure 5-8: Radial power profile used in ANSYS (right) compared to the power distribution generated by Serpent (left)

A design basis is taken to operate the reactor at thermal powers below 4 MWth. As previously discussed, the limiting operating state is the full power operating state. Accordingly, the model assumes the reactor is operating at this maximum steady-state power prior to the initiating event.

DB.RXS.02 The reactor core system is operated at steady state thermal power levels that prevent damage to the system during transients.

DC.RXS.02.A The power level of the reactor system is limited to 4 MWth.

License condition (see Part VI)

See also DC.ICS.01.A through D

5.6.2.4 Steady-state temperature

Transient analysis is initialized from steady-state temperature distributions, which are calculated using ANSYS Mechanical as described in Section 5.6.9.1. Steady-state heat removal from the fuel is modeled by fixing the temperature of the heat pipe vapor cores and wicks in each reactor cell to 538 C. This temperature corresponds to the peak power described in Section 5.6.2.3. Setting the heat pipe vapor cores and wicks of each reactor cell to this temperature is conservative, as it slightly overpredicts the initial temperatures throughout the module.

5.6.2.5 Passive heat removal

The ultimate heat sink during the LOHS is passive heat removal to the air in the reactor cavity that surrounds the reactor module. Heat transfer is modeled on the sides of the module shell using convective boundary conditions. Convective heat removal is conservatively neglected on the bottom of the shell. The heat transfer coefficient was determined to be 7 W/m^2-K, and this value is used in the LOHS analysis. The ambient air temperature in the analysis is 225 C.

Radiative heat transfer is conservatively neglected in the model; if it were included, this would increase the amount of heat transfer appearing in the simulation off the surface of the shell, resulting in reduced peak fuel temperatures. Based on the temperatures of the reactor module, it is expected that radiative heat transfer would be responsible for as much as and likely more heat removal than convective heat transfer.

A design basis is taken for the reactor module emplacement in the site building system, a subsystem of the building and auxiliary systems (BAS). The design basis ensures that the air flow area is consistent with the calculations mentioned above, and therefore that the assumed heat transfer coefficient for the model is indeed conservative. As described in Section 5.6.2.2, a design commitment is also made to verify effectiveness of the passive heat removal during startup testing.

DB.BAS.01 The building system provides for the emplacement of the reactor module in a configuration that supports passive cooling of the module shell.

DC.BAS.01.A The critical components of the reactor module, as identified in the appropriate procedure, are installed in the reactor module emplacement as described in the design documents referenced by the procedure.

POT.BAS.01.A (see Chapter 14)

5.6.2.6 Decay heat

The time-dependent fractional power curve from decay heat following reactor trip is generated by Serpent at discrete time points. The values used in ANSYS Mechanical at each timestep assume the decay heat generation from the beginning of the timestep is constant over the entire timestep. This assumption is conservative, as decay heat generation is overpredicted for any given timestep throughout the transient since the decay heat is not actually constant for the full timestep and is instead continuously decreasing with time. This decay heat curve applied in ANSYS bounds the curve generated by Serpent, as shown in Figure 5-9. Further, because the decay heat curve describes the fractional power, and the initial steady-state power (P_0) is set to the maximum allowable thermal power (see Section 5.6.2.3), the modeled decay heat generation represents the maximum decay heat generation expected in any transient.

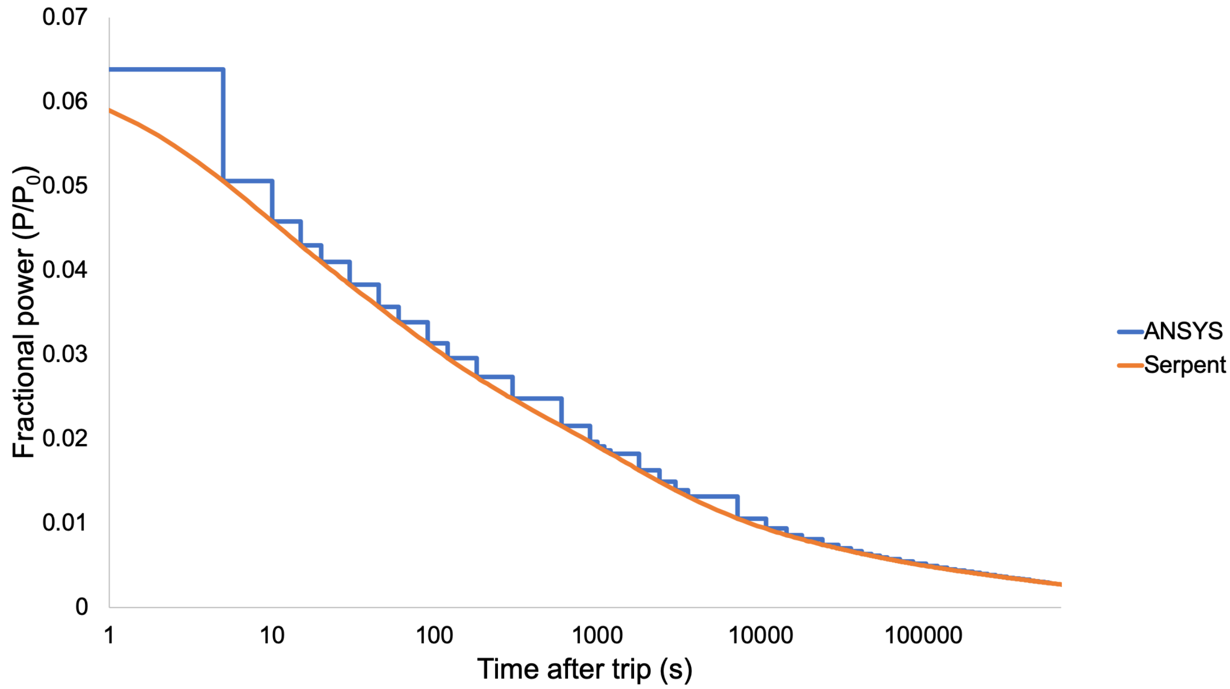

Figure 5-9: Decay heat curve used in safety analysis (ANSYS) vs. curve generated by Serpent

5.6.2.7 Reactivity effects

For the TOP, the reactivity insertion is determined based on the maximum rate and magnitude of rotation allowed by the control drum system. This assumption is conservative, as the maximum rotation rate is several orders of magnitude larger than the actual speed of insertion during reactivity letdown. The six-group point kinetics equations are used to calculate power changes during the active phase of the transient. An in-house tool is used to obtain this solution and is described in Section 5.6.9.3. The calculated power increase from positive reactivity insertion does not include reactivity feedback effects, which is conservative provided that the net temperature coefficient of reactivity is negative.

The reactivity insertion by the control drums is modeled as all three drums rotating their absorbers out of the core at their maximum mechanically-possible rate of 0.01 deg/sec. A design basis for the control drum system (CDS) is taken to ensure that the modeled insertion rate and magnitude reflect the largest the control drum system can achieve. In addition to limiting the reactivity insertion, a commitment is made to use stepper motors to prevent spurious drum rotation. Therefore, not only is the TOP event conservatively analyzed, it is also protected against in multiple ways.

DB.CDS.01 The control drum system is designed to limit both the rate and magnitude of reactivity insertion that the system can achieve so as to minimize the effect of an unintended reactivity insertion.

DC.CDS.01.A The maximum rotation speed of the drums is limited to 1×10^{-2} deg/sec.

POT.CDS.01.A (See Chapter 14)

DC.CDS.01.B The total reactivity worth of the drums is less than 700 pcm at all operating conditions.

SUT.CDS.01.B

DC.CDS.01.C The control drum actuators use stepper motors to eliminate the possibility of unintentional rotation.

POT.CDS.01.C

Inherent reactivity feedback effects are not included in the TOP analysis presented in this chapter. However, it is important to note that the Aurora has strongly negative reactivity coefficients, dominated by the strongly negative fuel temperature coefficient of reactivity. Neglecting negative reactivity feedback is conservative, as inherent negative reactivity feedback effects would put the Aurora into a less challenged state than that modeled. A design basis is taken for the reactor core system to ensure that the power coefficient of reactivity is negative, ensuring that neglecting the feedback is conservative.

DB.RXS.03 The reactor core system has inherently negative reactivity feedback.

DC.RXS.03.A The net power coefficient of reactivity of the reactor core system is negative.

SUT.RXS.03.A (see Chapter 14)

The time-dependent, normalized power used during the active phase of the TOP assumes the normalized power from the end of each timestep for the duration of the timestep. This conservatively overpredicts the amount of heat generated throughout each timestep and is shown in Figure 5-10.

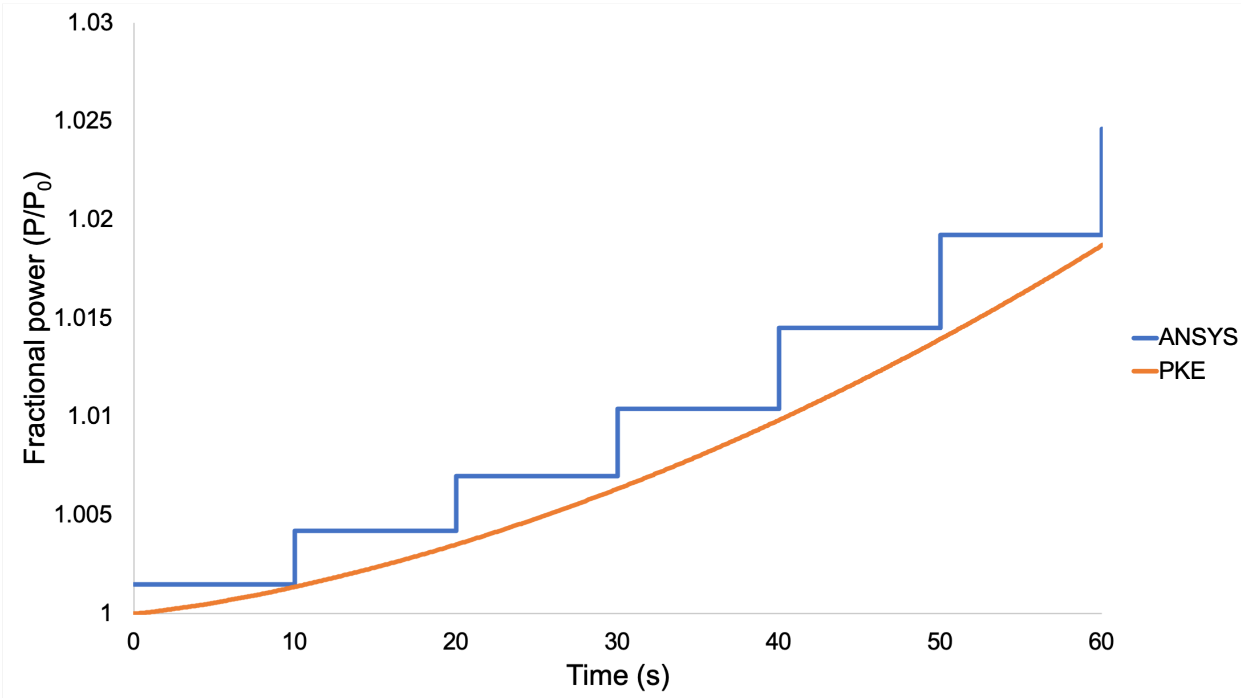

Figure 5-10: Fractional power curve used during active phase of TOP (ANSYS) vs. curve generated by point kinetics equations

5.6.2.8 Shutdown rod insertion

During both the TOP and LOHS, one of three shutdown rods is assumed to fail to insert. This is an unlikely event, as the shutdown rods are located directly above the active core in a protected sheath, and a substantial gap exists surrounding the rod to ensure a clear path to insertion. It is important to note that only one of three shutdown rods is needed to shut down the reactor at any temperature condition. Though it is unlikely for one of three not to insert (probability on order of 10^{-6}), and more unlikely for two of three (probability on order of 10^{-12}), either of which still resulting in a safe state, this consideration is in line with the MCA methodology to consider any one single failure or group of failures caused by a single event, while considering risk insight and defense-in-depth by considering one rod drop failure as of probability worthwhile to account for in the analysis. Although no event including TOP and LOHS is expected to simultaneously cause any shutdown rod failure, this extra failure was taken as a risk-informed, defense-in-depth interpretation of the MCA methodology.

A design basis is taken to ensure that even in the case of two stuck rods, the remaining rod provides sufficient negative reactivity to shut down the reactor.

DB.SRS.01 The shutdown rod system provides sufficient negative reactivity to achieve cold shutdown with insertion of one rod.

DC.SRS.01.A The worth of each shutdown rod will be greater than 1400 pcm, where 1400 pcm is greater than the total of: the reactivity worth associated with the temperature decrease from hot full power conditions to cold zero power conditions, and an additional margin of 500 pcm.

SUT.SRS.01.A1 and A2 (see Chapter 14)

5.6.2.9 Power conversion system

During the LOHS, the PCS is assumed to fail instantaneously. This is a conservative assumption, as the instantaneous stop of rotating components and heat removal by the PCS presents the most limiting heat transfer conditions; as discussed in Section 5.5.2.2, fuel temperatures reach their bounding values when no PCS flow coastdown is applied and the entirety of heat removal comes from the passive heat removal off the outer surface of the module shell.

During the TOP, the PCS continues to operate at the nominal system parameters; this is modeled with a fixed heat removal boundary condition in ANSYS while the reactor is at full power. This is conservative, as the capability of the PCS to reject additional heat above the nominal full-power rate is neglected. The PCS continues to remove decay heat post-shutdown in the normal shutdown cooling mode.

5.6.2.10 Reactor trip system

During both the TOP and LOHS, a reactor trip is assumed to be initiated when specific limits are exceeded. The reactor trip system sends a reactor trip signal to the shutdown rod system when a limit is exceeded. The only trip setpoint explicitly modeled in the safety analysis presented in this chapter is the fuel over-temperature setpoint of 655 C. Once the fuel temperature exceeds this value in the model, the simulation switches from full-power heat generation to decay heat generation (after the applicable rod insertion delay has been applied, as described in Section 5.6.2.11). More detail surrounding the reactor trip system, including the full list of reactor trip setpoints and information on how these parameters are measured, can be found in Chapter 2.

A design basis is taken to ensure that all setpoints are configured correctly. This ensures that the assumption of a reactor trip is representative of actual system behavior.

DB.ICS.01 The reactor trip system monitors reactor process variables and sends a reactor trip signal when a process variable exceeds a limit setpoint.

DC.ICS.01.A The reactor trip system sensors are installed in the correct locations.

POT.ICS.01.A1 and A2 (see Chapter 14)

SUT.ICS.01.A1

DC.ICS.01.B The reactor trip system process limit monitors are connected to the correct locations, and are configured with the correct sensor scaling information and limit setpoints.

POT.ICS.01.B1 and B2

TS.LCO.02 (see Part IV)

DC.ICS.01.C The reactor trip system sensors are connected to the correct process limit monitors.

POT.ICS.01.C1 and C2

SUT.ICS.01.C

DC.ICS.01.D The reactor trip system process limit monitors send a fault signal when a process variable exceeds a limit.

POT.ICS.01.D

TS.LCO.02

5.6.2.11 Shutdown rod insertion delay

During both the LOHS and TOP, a 10 second delay is assumed to occur from exceeding a trip setpoint (described in Section 5.6.2.10) to reactor shutdown. This is a conservative assumption, as the detection, signal processing, shutdown rod release, and shutdown rod insertion are cumulatively analyzed to occur in less than 10 seconds.

Design bases for both the reactor trip system and the shutdown rod system are taken to ensure that this assumption is conservative. Design commitments are made to limit the detection and signaling time to 6 seconds and the shutdown rod insertion time to 2 seconds, such that the cumulative time from exceeding a setpoint limit to rods being fully inserted into the core is less than 8 seconds, which includes a conservative margin to the 10 second assumed value.

DB.ICS.02 The reactor trip system sends a reactor trip signal to the shutdown rod system within a sufficient time of exceeding a limit to prevent damage to the reactor.

DC.ICS.02.A The reactor trip system detects the exceedance of a limit setpoint and sends a reactor trip signal within 6 s.

POT.ICS.02.A (see Chapter 14)

TS.LCO.02 (see Part IV)

DB.SRS.02 The shutdown rod system fully inserts the shutdown rods within a sufficient time after receiving a trip signal to prevent damage to the reactor.

DC.SRS.02.A The shutdown rod system fully inserts shutdown rods within 4 seconds of receiving a trip signal.

POT.SRS.02.A (see Chapter 14)

SUT.SRS.02.A

TS.LCO.01 (see Part IV)

5.6.2.12 Material properties

The majority of materials present in the reactor module are well-characterized metals, primarily SS316L, SS304, and zirconium. For these materials, standard temperature-dependent material properties are used. SS316L and SS304 temperature-dependent properties are taken from [39]. Temperature-dependent zirconium density, thermal conductivity, and specific heat capacity are taken from [40], [41], and [24], respectively. For other materials, including fuel and heat pipe vapor core, additional description is provided in this section.

5.6.2.12.1 Fuel

U-10Zr is a well characterized material up to significantly high burnups, with significant operational experience during the Integral Fast Reactor program. Material properties, namely density, thermal conductivity, and specific heat capacity, are modeled with temperature dependence from [42] and [43], [44], and [24], respectively; additional effects due to burnup are described below.

Thermal conductivity is known to decrease with burnup as fission gases create voids in the fuel matrix. A decrement of 30% is applied to the thermal conductivity throughout the analysis (i.e., the assumed thermal conductivity is 70% of its nominal value) [24]. This decrement is conservative as it accounts for greater burnup effects than the Aurora fuel is analyzed to experience.

The density of the fuel decreases with burnup as the fuel swells; however, the mass is largely unchanged and is the factor of interest during the transient. The heat capacity does not change significantly with burnup and is accordingly not decremented in the model.

A design basis is taken for the reactor core system to ensure that the fuel is of the material, with associated properties, assumed in this safety analysis.

DB.RXS.01	The reactor core system uses metal fuel with well characterized properties.
DC.RXS.01.A	The fuel in the reactor system is procured according to 10 CFR Part 50 Appendix B, with a critical characteristic of thermal conductivity. (see Oklo QAPD)

5.6.2.12.2 Heat pipe vapor core

The heat pipe vapor core occupies the region inside of the wick and contains low pressure vapor. The equivalent thermal conductivity of heat pipes was calculated using heat pipe performance correlations, which show it to be 34,000,000 W/m-K and generally between 10,000,000 to 100,000,000 W/m-K in the range of operating conditions expected. The vapor core is modeled in ANSYS as a solid body with a high thermal conductivity (conservatively set to 1,000,000 W/m-K), low specific heat (1 J/kg-K), and low density (0.016 kg/m^3). This simplification drastically reduces computational requirements (no complicated evaporation, condensation, or turbulence effects need be simulated) while accurately capturing the thermal behavior of heat pipes. Note that the results of the ANSYS analyses are not especially sensitive to the absolute value used for the heat pipes' thermal conductivity; since this thermal conductivity is exceptionally high, the temperature change in the vapor core from the evaporator region to the condenser region is still minimal.

5.6.2.13 Contact conductance

Contact conductance between bodies are previously analyzed and considered in the analysis. Perfect contact conductance is specified between the bodies inside each reactor cell, as the sodium bond ensures effective thermal contact throughout this volume. An exceptionally low contact conductance of 100 W/m^2-K is applied to adjacent cells within the capsule, which accounts for radiative heat transfer alone and neglects conduction effects of bodies in contact. Contact conductance between bodies in contact horizontally is assumed to be 1000 W/m^2-K, and bodies in contact vertically have a contact conductance of 4000 W/m^2-K. Each of these values overpredicts the expected thermal resistance between the active core and the ultimate heat sink in the reactor cavity.

As described in Section 5.6.2.2, the geometry of the system is verified through the appropriate design commitments and programmatic controls.

5.6.3 Transient overpower results

As described in 5.6.1, the transient overpower event assumes a maximum reactivity insertion of all three control drums at the maximum speed, 0.01 deg/s, coincident with a failure of one of three shutdown rods to insert. The active phase begins with the initiation of drum rotation, at t=0 s. The reactor trip system detects an overtemperature setpoint when the fuel reaches 655 C, and following a 10-second delay, two shutdown rods insert, shutting down the reactor, ending the active phase. As the PCS continues to operate, decay heat is removed from the fuel, called the cooldown phase.

The transient overpower event can thus be described by two phases:

1. Active phase, where power increases from steady-state until the reactor is tripped.

2. Cooldown phase, where decay heat is removed by the PCS.

5.6.3.1 Active phase

The active phase begins with the initiation of the drum rotation, at t=0 s. The overtemperature trip setpoint of 655 C is reached at t=36s. As described in 5.6.2.11, a 10-second delay is assumed between exceeding the overtemperature trip setpoint until the reactor is tripped at t=46s. Heat removal by the PCS continues throughout the active phase.

The response of the peak fuel temperature to the transient overpower is benign: the rate of temperature increase is slow and relatively linear, occurring at an average rate of +0.45 C/s. The final peak fuel temperature reached during the active phase (before shutdown rods insert and the reactor trips) is 660 C, a total increase of only 20 C above the nominal peak temperature of 640 C.

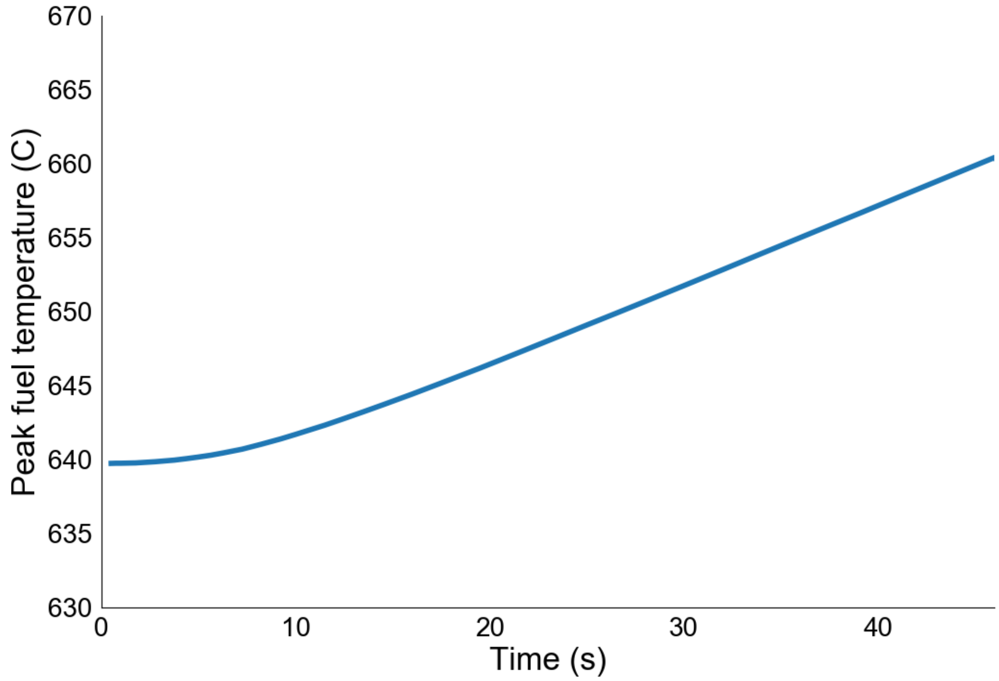

Figure 5-11: Peak fuel temperature during active phase of transient overpower

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Figure 5-12: Active phase of transient overpower. Temperature distributions at steady state (left) and end of active phase (right)

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Figure 5-13: Active phase of transient overpower. Temperature distribution at steady state (left) and end of active phase (right)

During the active phase, peak fuel temperatures increase as fission heat generation exceeds heat removal to the PCS. This can be seen in Figure 5-12 and Figure 5-13, as areas surrounding the fuel heat up. Note that in these two figures, a nonuniform temperature scale is used to highlight the temperature gradient in the range of interest. This nonuniform temperature scale is consistent across all the two-dimensional temperature gradient plots presented in this chapter. The overall temperature distribution maintains a similar relative shape throughout the active phase of the event.

5.6.3.2 Cooldown phase

After the reactor is tripped, the PCS continues to cool the fuel following the cessation of fission heat generation. Upon reactor trip occurring and fission heat generation ceasing, peak fuel temperature decreases rapidly as the heat generation in the fuel transitions to being driven by radioactive decay. The peak fuel temperature decreases from its maximum value of 660 C at the end of the active phase to approximately 596 C in only 60 s, for an average rate of decrease of 1.1 C/s. As decay heat generation continues to be removed by the PCS, the peak fuel temperature more slowly approaches the nominal heat exchanger steady-state temperature value: the peak fuel temperature ultimately reaches a value of 558 C one hour after shutdown.

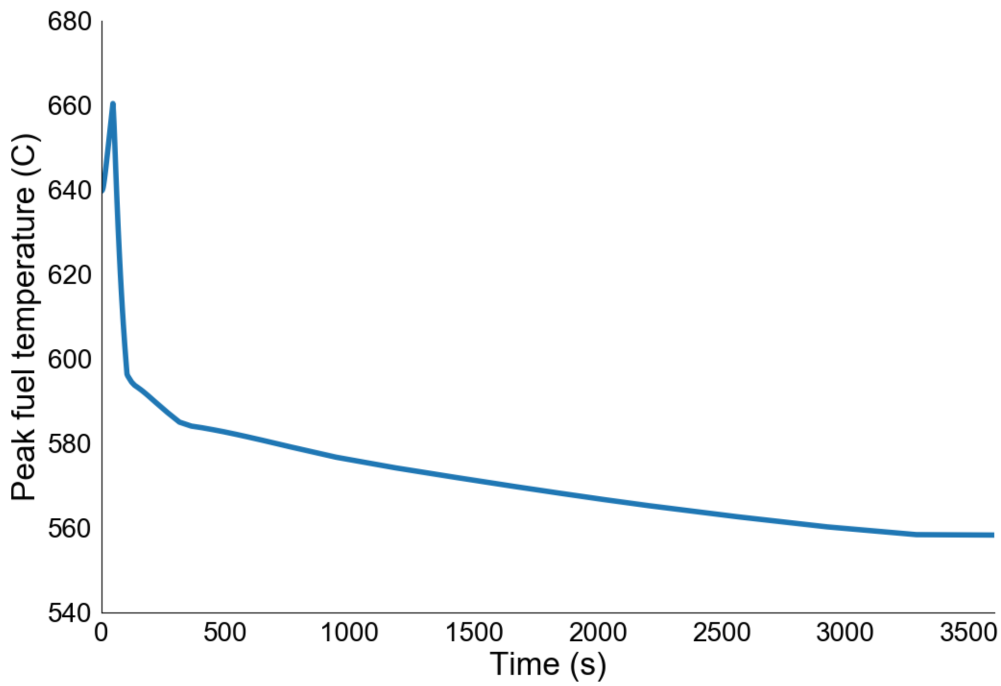

Figure 5-14: Peak fuel temperature during cooldown phase of transient overpower

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Figure 5-15: Cooldown phase of transient overpower. Temperature distributions at end of the active phase (left) and one hour after reactor trip (right)

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Figure 5-16: Cooldown phase of transient overpower. Temperature distributions at end of the active phase (left) and one hour after reactor trip (right)

During the cooldown phase, the peak fuel temperature decreases quickly as the PCS continues to remove heat from the fuel effectively. This can be seen in Figure 5-15 and Figure 5-16, as the fuel and surrounding structures cool down quickly. Temperatures throughout the system decrease to below steady-state values within minutes.

5.6.3.3 Conclusion

The peak temperature reached during the event is 660 C. Substantial margin is maintained below the operational limit, and both the safety and operational goals are satisfied.

5.6.4 Loss of heat sink results

As described in 5.6.1, the loss of heat sink event assumes a complete failure of the PCS coincident with a failure of one of three shutdown rods to insert. The reactor trip system detects an overtemperature setpoint when the fuel reaches 655 C, and, following a 10-second delay, two shutdown rods insert, shutting down the reactor. Following a trip, decay heat continues to produce heat. Heat in the module is removed only through passive, convective air flow in the reactor cavity surrounding the shell.

The LOHS event can be captured by four phases:

1. Full-power heatup, where temperature increases from steady state until the reactor is tripped.
2. Initial heat redistribution, from reactor trip until t=252s.
3. Decay-driven heatup, from t=252s until t=6.6h.
4. Residual heat rejection cooldown, after t=6.6h.

5.6.4.1 Full-power heatup

The full-power heatup begins with the complete failure of the PCS, and fission heat is generated at its nominal, steady-state values. In reality, temperature feedbacks would reduce core power; for these simulations, these feedback effects are conservatively neglected. The overtemperature trip setpoint of 655 C is reached at t=20s. As described in Section 5.6.2.11, a 10-second delay is assumed between exceeding the overtemperature trip setpoint until the reactor is tripped. Once the reactor is tripped, the initial heat redistribution phase begins.

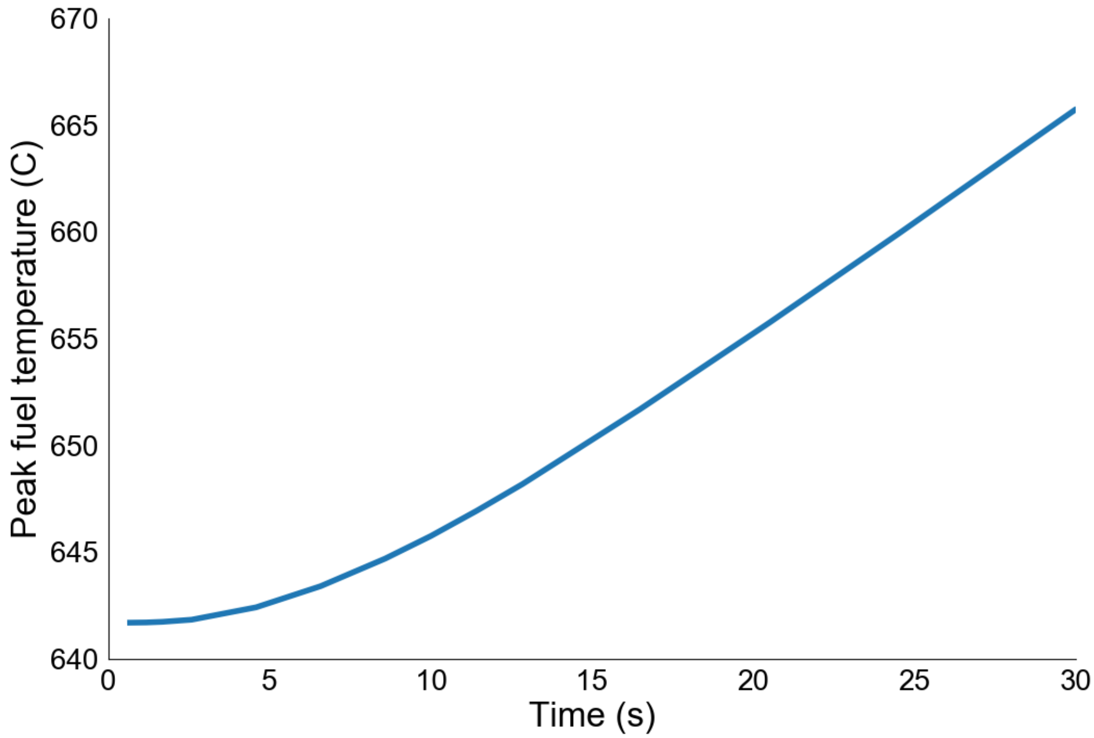

Figure 5-17: Peak fuel temperature during full-power heatup phase of loss of heat sink

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Figure 5-18: Full-power heatup phase of loss of heat sink. Temperature distributions at steady state (left) and reactor trip (right)

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Figure 5-19: Full-power heatup phase of loss of heat sink. Temperature distributions at steady state (left) and reactor trip (right)

5.6.4.2 *Initial heat redistribution*

During the initial heat redistribution phase, peak temperatures in the fuel decrease quickly as heat distributes to nearby bodies. Heat is conducted through the fuel, reflectors, and steel, both radially and axially out of the core. Heat is further distributed throughout the system, including the shielding and structural materials surrounding the core. Because of the tightly coupled and highly conductive materials surrounding the fuel, the majority of this redistribution occurs within the first 60 seconds and continues until t=252 seconds, when the peak fuel temperature reaches a local minimum of approximately 600 C.

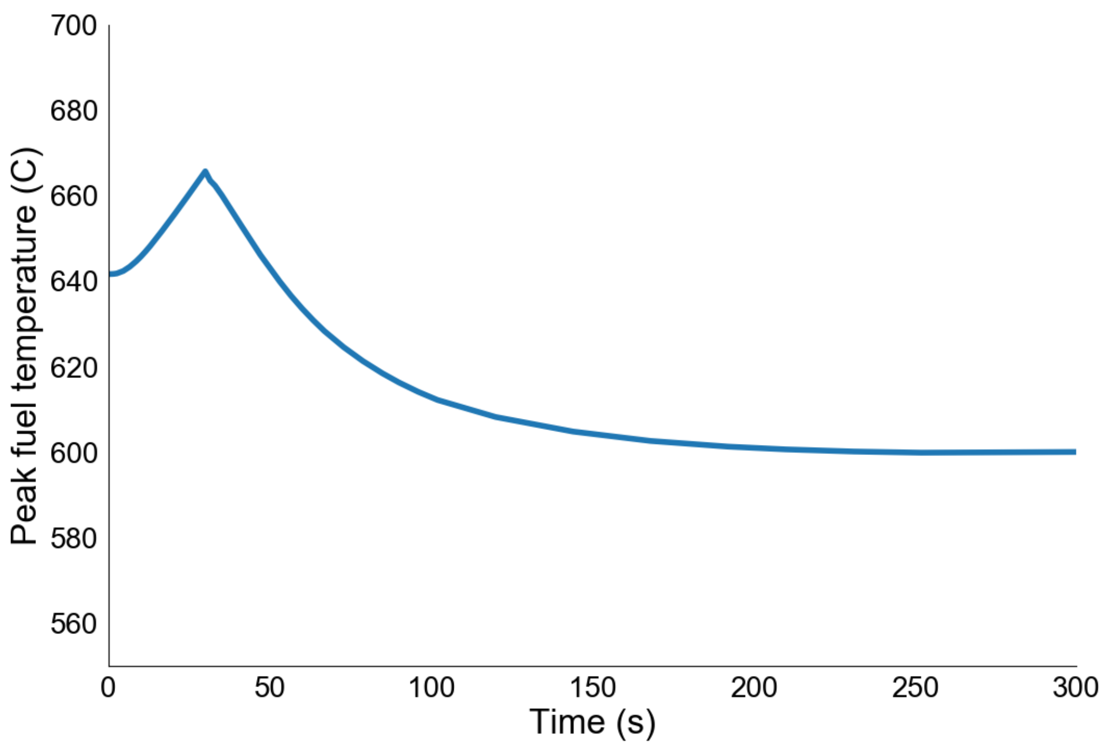

Figure 5-20: Peak fuel temperature during initial heat redistribution phase of loss of heat sink

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Figure 5-21: Initial heat distribution phase of loss of heat sink. Temperature distributions at reactor trip (left) and end of initial heat redistribution phase (right)

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Figure 5-22: Initial heat redistribution phase of loss of heat sink. Temperature distributions at reactor trip (left) and end of initial heat redistribution phase (right)

During the initial heat redistribution phase, localized steady-state peak fuel temperatures begin to decrease as heat is conducted to surrounding structures. This is visible in the temperature distribution plots in Figure 5-21 and Figure 5-22. As the fuel areas cool down, surrounding areas, including the heat pipes and heat exchanger, heat up. Outer structures remain largely unchanged during this phase because of the larger heat transfer distance.

5.6.4.3 Decay-driven heatup

Following the initial heat redistribution phase, the peak fuel temperature begins to increase as decay heat generation in the fuel exceeds heat removal. This phase continues until t=6.6 hours, when a peak fuel temperature of 662 C is reached. This increase in peak fuel temperature occurs somewhat asymptotically, increasing relatively quickly during the first 3 hours, then much more slowly until the peak fuel temperature is reached. This behavior is driven by the shape of the decay heat generation curve, which follows that of an exponential decrease (since radioactive decay is causing the heat generation).

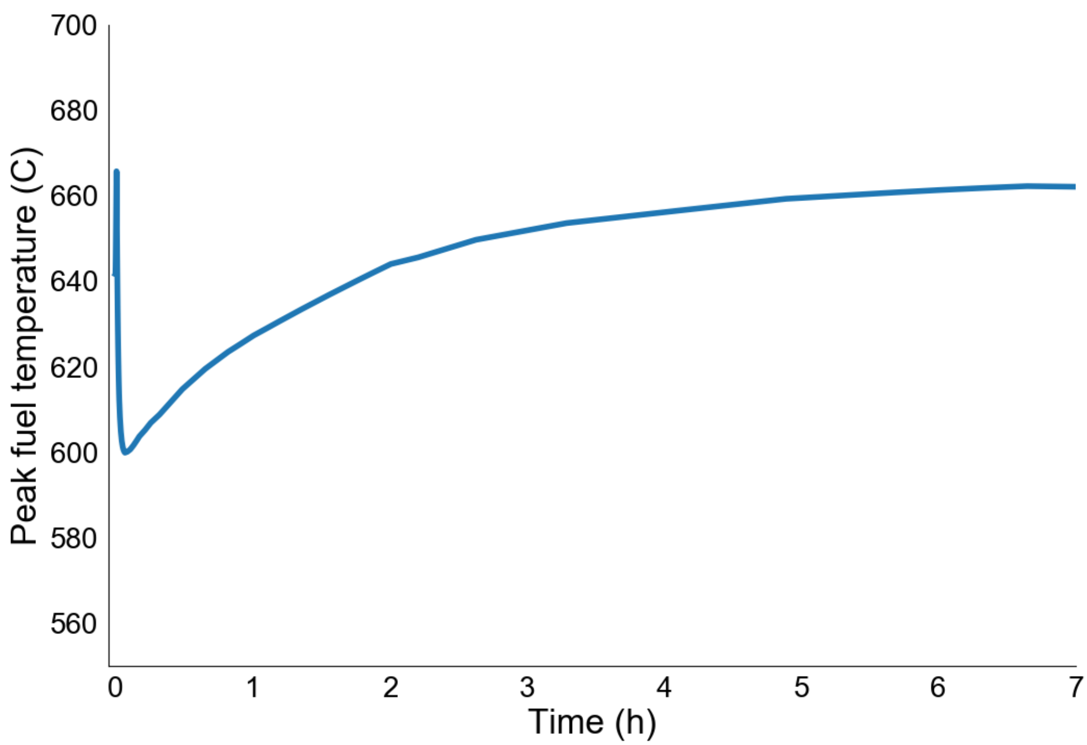

Figure 5-23: Peak fuel temperature during decay-driven heatup phase of loss of heat sink

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Figure 5-24: Decay-driven heatup phase of loss of heat sink. Temperature distributions at end of the initial heat redistribution phase (left) and end of decay-driven heatup phase (right)

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Figure 5-25: Decay-driven heatup phase of loss of heat sink. Temperature distributions at end of the initial heat redistribution phase (left) and end of decay-driven heatup phase (right)

During the decay-driven heatup phase, peak fuel temperature increases to its local maximum as decay heat generation exceeds passive heat removal via conduction from the fuel to the outer surfaces of the shell and ultimately via convection to the air in the reactor cavity. This is visible in the temperature distribution plots in Figure 5-24 and Figure 5-25. Structures in close thermal contact to the fuel increase in temperature.

5.6.4.4 Residual heat rejection cooldown

Following the decay-driven heatup phase, the residual heat rejection cooldown phase continues indefinitely. Passive, convective heat removal from the shell to the air surrounding the shell exceeds the heat generated by decay heat, and peak fuel temperatures decrease monotonically. After 24 hours, the peak fuel temperature is 620 C, decreasing from the peak of 662 C at 6.6 hours with an average cooldown rate of -2.4 C/hr during this period. After 72 hours, the peak fuel temperature has decreased to 521 C, giving an average cooldown rate of -2.1 C/hr for the time period from 24 to 72 hours. After 7 days, the peak fuel temperature has decreased to 418 C, which corresponds to an average cooldown rate of -1.1 C/hr during the time period from 72 hours (3 days) to 7 days after shutdown. This result reflects the continued decrease in decay heat generation coupled with the continued heat transfer from the fuel to the outer core structures such as the heat exchanger and radial reflectors, where that heat is progressively removed by convection to the surrounding air.

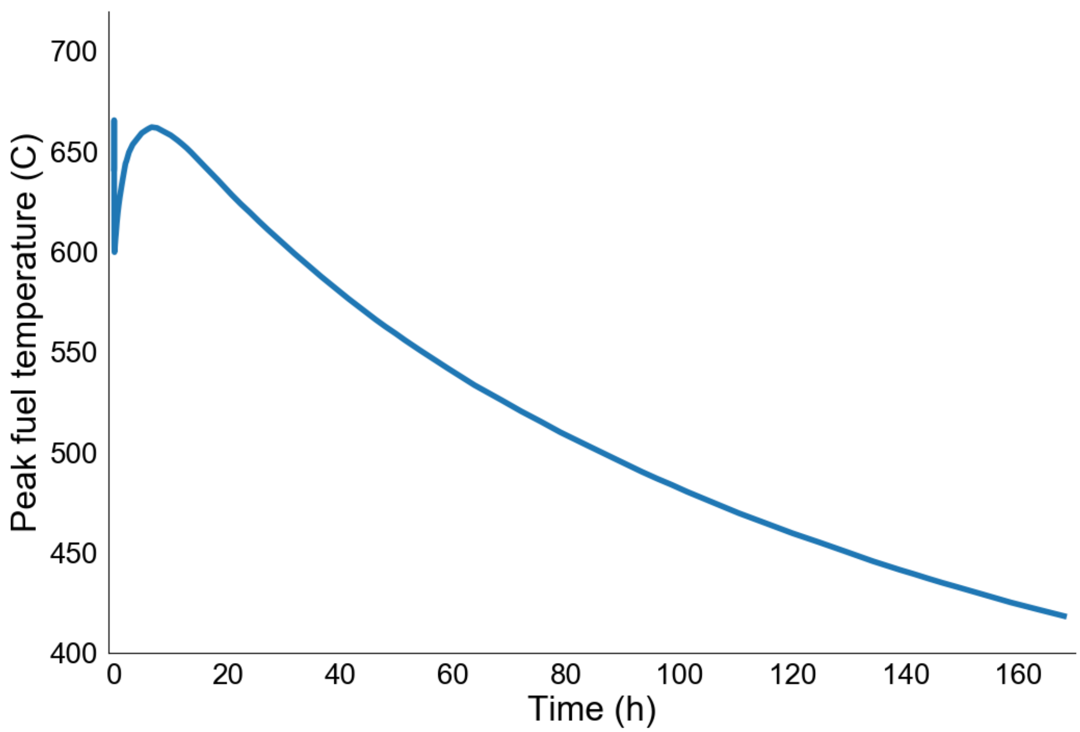

Figure 5-26: Peak fuel temperature during residual heat rejection cooldown phase of loss of heat sink

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Figure 5-27: Residual heat rejection cooldown phase of loss of heat sink. Temperature distributions at end of decay-driven heatup (left) and one day after trip (right)

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Figure 5-28: Residual heat rejection cooldown phase of loss of heat sink. Temperature distributions at end of decay-driven heatup (left) and one day after trip (right)

During the residual heat rejection cooldown phase, peak fuel temperature decreases monotonically from its maximum as passive heat removal to the building air exceeds decay heat generation. As shown in Figure 5-27 and Figure 5-28, temperatures throughout the module decrease, most notably in the fuel and nearby structures. Peak fuel temperature is less than its steady-state value at t=17 hours after reactor trip.

5.6.4.5 Conclusion

The peak temperature reached following reactor trip is 662 C. Substantial margin is maintained below the operational limit, and both the safety and operational goals are satisfied.

5.6.5 End state

The end state of the safety analysis is when the analysis reaches the following conditions:

- The core is in a subcritical, shutdown state.
- Peak temperature in the core is less than the peak steady-state temperature and is decreasing.

The end state represents a successful shutdown of the reactor. Therefore, the safety analysis is performed until the end state is reached. If the safety and operational limits are maintained throughout the transient and the end state is reached, then the safety and operational goals are satisfied. It is important to note that no human actions are necessary to achieve the end state. As shown in Section 5.6.3 and Section 5.6.4, the insertion of shutdown rods ensures that the end state is reached following the most challenging failures of two systems, coincident with a failure to insert one of three shutdown rods.

5.6.6 Maximum credible accident determination

The results of the TOP and LOHS events are shown in previous sections. Each of these events assumes a single equipment failure that would result in the most challenged state, coincident with a failure of one of three shutdown rods to insert. These postulated events are consistent with the MCA definition with defense-in-depth, as described in Section 5.1.1.

During the LOHS event, peak fuel temperatures are higher than during the transient overpower. This is due to the failure of the PCS, as well as bypass and decay heat removal via the radiators, and reliance on passive cooling to the air outside of the shell, in comparison to active cooling occurring via the PCS during the TOP. Further, temperatures that exceed steady-state temperatures are experienced for over 17 hours during the LOHS, while temperatures return to steady-state values within minutes during the TOP. Therefore, the loss of heat sink event is considered the more challenging event and is designated as the MCA.

5.6.7 Uncertainties

Uncertainties in the accident analysis are addressed through both of the following items:

1. Conservative assumptions and models, discussed in Section 5.6.2
2. Sensitivity analyses on parameters related to safety and operational limits

Sensitivity analyses have been conducted on several parameters of interest. Parameters that have significant impact on the results were analyzed further, and conservative values are ultimately assumed for the results presented here, as described in Section 5.6.2. Of the parameters of interest, the following are most impactful:

- Heat transfer coefficient on the outer surface of the shell

- Air temperature on the outer surface of the shell.

The following parameters are not found to significantly impact the results:

- Shutdown rod insertion delay, as the cumulative energy produced during the seconds of delay is insignificant in comparison to the decay heat generated during the hours following a trip

- Cell-to-cell contact conductance, as the heat pipes provide effective and independent heat transfer pathways to large heat sinks

- Contact conductance between other bodies, due to the redundant, independent heat transfer pathways throughout the module and the low power density

5.6.8 Required human actions

None.

5.6.9 Computer codes used

Several computer codes are used as part of the transient analysis presented in this chapter. The analyzed models presented here were built and simulated using ANSYS Mechanical. The power and decay heat parameters used in the analysis were generated using Serpent. The power increase curve due to drum rotation in the transient overpower was generated using an in-house point kinetics solver. Oklo performed commercial grade dedication of these codes for use in safety analysis.

5.6.9.1 ANSYS Mechanical

ANSYS offers a comprehensive software suite spanning a broad range of engineering simulation. ANSYS Mechanical is used to evaluate steady-state and transient temperature evolution in the Aurora.

ANSYS Mechanical is a finite element analysis (FEA) tool for structural analysis, thermal analysis, and coupled-physics capabilities, including thermal-structural analysis. ANSYS Mechanical provides a complete set of finite element behavior, material models, and equation solvers for a wide range of mechanical design problems.

For the safety analysis, steady-state heat generation rates are used in ANSYS Mechanical as axially discretized, volumetric heat generation rates in each reactor cell. Heat generation in the fuel, along with the boundary conditions described in Section 5.6.2, are used to initialize the steady-state temperature distribution throughout the module. The spatial power generation distribution is kept constant throughout the transient; the time-dependent, normalized power curves in the TOP and LOHS are applied as scaling factors on this steady-state heat generation distribution to model the time evolution of the total heat generation.

The time-dependent, normalized power following shutdown is calculated using Serpent. In ANSYS Mechanical, for each timestep, the normalized power assumes the power from the beginning of the timestep, which conservatively overpredicts decay heat throughout the event, shown in Figure 5-9.

The time-dependent, normalized power during the active phase of the TOP is calculated using the six-group point kinetics equations in the in-house Oklo point kinetics solver. In ANSYS Mechanical, for each timestep, the normalized power assumes the power from the end of the timestep, which conservatively overpredicts heat throughout the active phase of the event.

While the ANSYS suite offers a wide range of engineering simulation capabilities, the model used in the safety analysis is three-dimensional conduction with temperature-dependent material properties.

5.6.9.2 Serpent

Serpent is a three-dimensional, continuous-energy Monte Carlo reactor physics burnup calculation code specifically designed for reactor analysis applications. The standard output includes effective and infinite multiplication factors, point-kinetic parameters, effective delayed neutron fractions, and precursor group decay constants.

For safety analysis, the steady-state initial radial power distribution (the power generated in every reactor cell in the core) is calculated by Serpent. These radial heat generation rates are used to determine the heat generation rates in any particular cell at various axial levels by further applying a cosine-shaped axial power distribution to the total cell power. The axially discretized heat generation rates are then used to initialize steady-state temperatures in ANSYS Mechanical.

Serpent's fuel depletion calculation capability is used to estimate decay heat generation rates following shutdown, which are used as time-dependent heat generation rates in ANSYS Mechanical. The core-wide heat generation rate is calculated at hundreds of timesteps and normalized to nominal power. The values used in the safety analysis consistently overpredict the values generated by Serpent; a conservative assumption described in Section 5.6.2.6.

Serpent is also used to calculate control drum reactivity worths and point kinetics parameters, which are input into the in-house Oklo point kinetics solver (described in Section 5.6.9.3) to solve for the reactor power increase that results from the transient overpower.

5.6.9.3 Point kinetics equation solver

The point kinetics equation solver, developed in-house by Oklo, is used to solve the six-group point kinetics equations for the reactor power changes that result from the positive reactivity worth added by the drum rotation in the transient overpower bounding event. The drum worths input into the point kinetics solver are calculated by Serpent. Serpent is also used to calculate the input point kinetics parameters such as mean neutron generation time and the six-delayed-group decay constants and β fractions (which sum to β-effective).

5.6.10 Summary

This safety analysis approach and accompanying methods are based on NRC regulation and guidance, adapted where necessary for a non-LWR design. The design of the Aurora has several key characteristics that ensure a robust and safe response to a wide range of events:

- Small thermal power, several orders of magnitude smaller than a large LWR
- Low burnup fuel, which results in limited fission product inventory

- Metal fuel, which has high thermal conductivity and low heat capacity, meaning stored heat is manageable, and heat can be removed effectively
- Heat pipes, which provide redundant, independent heat transfer pathways and ensure fuel can be cooled
- Solid-state arrangement of high thermal conductivity components, which provides effective heat transfer throughout the module
- Low power density, which allows decay heat to be removed effectively by passive means

The results of the safety analysis demonstrate a significant margin to the safety and operational limits during two bounding events, thereby ensuring that the Aurora safety goal is met. During the transient overpower event, the three control drums are assumed to introduce positive reactivity at their maximum rate. During the loss of heat sink event, a complete failure of the power conversion system is assumed to occur coincident with the failure of a shutdown rod to insert. In both of these events, the reactor is shut down, no material or structural damage occurs, and no radioactive material is released. The loss of heat sink event results in higher peak fuel temperatures than the transient overpower event, and as such, is designated the MCA.

The safety analysis includes substantial conservatisms that, combined with sensitivity analyses on key parameters, provide confidence in the results and conclusions. Design commitments and programmatic controls will be used to validate and protect the assumptions made in the safety analysis. Altogether, the safety analysis shows that the safety goal of the Aurora, to control the release of radionuclides to minimize the risk to the public and the environment, is achieved.

5.6.11 Control of reactivity

The underlying importance of reactivity control is that it is the means to control the generation of heat in the reactor. Imbalances between the heat generation and the heat removal in the reactor core lead to changes in core temperatures. As such, one means of limiting peak fuel temperature during the MCA is to control the reactivity of the core.

Reactivity is controlled in the Aurora through three distinct means: the shutdown rods, the control drums, and the inherent characteristics of the reactor. The control drums are solely responsible for compensating for slow reactivity changes due to fuel depletion and are not relied upon for any safety function during the transient overpower and loss of heat sink bounding events; they appear only in the transient overpower event solely to describe the challenge they present due to their malfunction. As such, they provide no required safety functions and are not further discussed in this section.

The shutdown rods, as their name implies, are the primary mechanism for achieving shutdown of the neutron chain reaction. The inherent characteristics of the reactor, primarily consisting of reactivity feedback coefficients of temperature, also serve to mitigate and control both planned and unplanned changes in core reactivity.

5.6.11.1 Shutdown rods

Only a single rod must insert fully in order for the neutron chain reaction to be shut down and the core made subcritical; this is the safety function of the shutdown rod system. The full stroke length of each rod is protected by a stainless-steel sheath that separates the rod from the surrounding reactor cells. As the core of the Aurora operates at near atmospheric pressure, there is no significant driving force that opposes rod insertion. Since the Aurora operates with a very low power density, it is relatively insensitive to any potential delay time that might elapse between the start of a transient until full rod insertion is achieved. As a result of this design, the shutdown rod system robustly provides its required safety function by inserting two out of three shutdown rods during the MCA.

5.6.11.2 Inherent characteristics

Inherent characteristics of reactivity control are a backstop to mitigate undesired transient behavior. They are not considered passive means because “failure” of these characteristics is nonphysical. A degradation of inherent characteristics is possible, but complete failure cannot occur. In the Aurora, changes in reactor power are inherently controlled and limited through two means: the physical core configuration, and the large negative temperature coefficient of reactivity. The Aurora is configured in such a way that it is in the most reactive configuration during normal operations. Any disruptions to the physical configuration of the core would lead the fuel to be in a less reactive state. The Aurora has an inherently large negative temperature coefficient of reactivity, due primarily to the large thermal expansion that the metal core materials, particularly the fuel, experience as their temperature increases. The negative temperature coefficient of reactivity is conservatively modeled in the MCA by assuming the reactor power stays constant when heat removal by the secondary system is lost.

5.6.12 Heat removal

Another means of limiting peak fuel temperatures is by removing heat from the reactor module. In the Aurora, heat removal is controlled by three distinct means: normal operation of the secondary system, thermal dissipation throughout the reactor materials, and residual heat rejection to the surrounding environment. Because the identified MCA involves a failure in the power conversion system piping, heat removal via this system is accordingly not included in this discussion on required safety functions.

Since the fuel is the heat generation source, the first order heat removal function during the MCA is to cool the fuel. The physical effect of interest is the conduction of heat from the metal fuel to other reactor components, the vast majority of which are also metal. Specifically, the sensible thermal mass of the reactor module is what initially drives the temperature response of the fuel following off-normal events. In other words, what is important for the Aurora's heat removal function is the amount of thermal mass readily available for the fuel to dissipate heat through, relative to the amount of heat generated. The amount of heat to be dissipated is limited by the low power density of the Aurora. Passive heat dissipation through the Aurora occurs through the heat pipes and inherent heat transfer parameters.

While the total decay heat of the Aurora is small relative to the sensible thermal mass available, it is indeed nonzero. As such, over longer time horizons, passive heat rejection from the reactor module to the surrounding environment must be considered to accurately capture the temperature response of the system. This residual heat rejection may be very small, but it is important to capture, as otherwise the system can only heat up over time.

5.6.12.1 Thermal dissipation

The Aurora operates at a low power density, significantly lower than that of commercial LWRs, generally one to two orders of magnitude smaller. It operates at a lower power density than liquid sodium cooled fast reactors, up to three orders of magnitude lower. The Aurora even operates at a lower power density than other reactors that rely primarily on conduction for decay heat removal such as high temperature gas reactors. The low power density of the Aurora thus serves to limit the amount of heat present in the fuel that must be dissipated to surrounding materials when normal cooling via the secondary system is decreased or lost.

Conduction is the dominant inherent contributor to heat dissipation from the fuel to other reactor components. For conduction to be an effective means of dissipating the heat from the fuel to other reactor core components, several physical parameters can be controlled. The fuel, and other core components, will be manufactured and assembled in such a way that an acceptable thermal contact will be established between the components to allow for good conduction capabilities.

Passive heat dissipation is accomplished by transferring heat from the fuel to cooler materials in the core; thermal contact between adjacent bodies, as well as the heat pipes, ensure effective heat transfer throughout the entire reactor module.

The net result of these passive and inherent characteristics of the Aurora reactor core is to provide a large thermal sink available for accepting heat from the fuel in scenarios where the nominal cooling pathway (from the fuel to the heat pipes and then to the secondary side working fluid) is hindered. This sink is both large relative to the heat generation (due to the Aurora's low power density), and available, since the materials of construction are either inherently very conductive (metals) or designed to present exceptionally small thermal resistance (heat pipes). This large thermal sink serves to limit the peak temperatures encountered in the fuel during the MCA.

5.6.12.2 Residual heat rejection

As the heat generated by the fuel is distributed throughout the core, some of this residual heat is rejected to the surrounding environment through natural convection to the air in the reactor cavity surrounding the module. Because of the characteristics of the Aurora core discussed in previous sections, namely its low power density, high thermal conductivity and high heat capacity, this heat rejection rate need not be large to have a very beneficial effect on limiting peak temperatures during the MCA.

5.6.13 Confinement

Confinement of radionuclides in the Aurora occurs through inherent properties of the fuel material and several structural barriers, including the reactor cell cans, the capsule, the module shell, the building basement, and the building first floor. However, the integrity of the fuel and the reactor cell cans is not challenged during the MCA, and as such, discussion of the other structural barriers is not presented here.

5.6.13.1 Fuel

The two contributing factors to the confinement safety function that are related to the fuel are the quantity of the material and the metal form.

5.6.13.1.1 Quantity of material

Because of the small size and power output of the Aurora, in comparison to a large LWR, the small amount of fuel mass limits the amount of radionuclides present throughout the life of the reactor, which limits the release scenario. Further, after a 20-year lifetime, the Aurora fuel has a burnup of less than 1 atom %. This small burnup means that very few radionuclides are generated in the fuel matrix during normal operation, which serves at all times to minimize the risk posed by challenges to the safety goal of the reactor, including during the MCA.

5.6.13.1.2 Metal form

The Aurora fuel is metal, in the form of a U-10Zr alloy, which has shown excellent performance to significantly higher burnups than the Aurora. Specifically, metal fuel used in the Experimental Breeder Reactor-II (EBR-II) was generally operated to a burnup of four atom% in the early years of operation, and later up to 10-15 atom%. Metal fuel, like other metals, is a relatively nonporous solid with a regular crystal lattice. As a result, the vast majority of fission products are retained within the fuel matrix. At approximately 1 atom% burnup, the voids caused by gaseous fission products within the fuel begin to interconnect and release fission gases and volatile elements from the fuel matrix and into the plenum. Nonetheless, the fission products that reach the plenum are a small fraction of the overall fission products. The safety limit is maintained during the MCA, and so the fuel performs its containment required safety function.

5.6.13.2 Reactor cell cans

The reactor cell can encloses the fuel and serves as its outer container. It is made of stainless steel and is designed to be leak-tight. Thus, it serves a mechanical function to physically separate the fuel material from objects exterior to the reactor cell itself. During normal operations, it serves as the sealed structural barrier that contains radionuclides generated during fuel depletion that are not retained within the fuel matrix; however, most radionuclides are expected to be retained in the fuel itself due to the low burnup of the Aurora fuel. The operational goal of maintaining reactor cell can integrity is satisfied during the MCA, such that the reactor cell cans perform their containment required safety function.

5.6.14 Summary

The required safety functions described in this chapter summarize the key design features of the Aurora reactor that ensure safety of the reactor. Chapter 2 describes the design bases and resulting design commitments that are taken to ensure that these required safety functions can be relied on. This chapter further describes how those design commitments are incorporated into the safety analysis to ensure that the as-modeled system is representative of the as-designed system. Finally, programmatic controls are put in place to demonstrate that the characteristics of the as-built system provide the required safety functions.

The remaining nonrequired safety functions shown in Figure 4-2 also contribute to the safety of the reactor and provide additional defense-in-depth. Note too that other design bases are present (such as those for the control drums), not to provide the required safety functions, but to minimize the severity of the challenges that the required safety functions respond to.

II.06 Fire protection

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6 FIRE PROTECTION

6.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(6) requires, "A description and analysis of the fire protection design features for the reactor necessary to comply with 10 CFR part 50, Appendix A, GDC 3, and § 50.48 of this chapter," passing the requirement to 10 CFR 50.48, "Fire protection." Specifically, 10 CFR 50.48(a)(1)-(3) applies and requires the following:

> (1) Each holder of an operating license issued under this part or a combined license issued under part 52 of this chapter must have a fire protection plan that satisfies Criterion 3 of Appendix A to this part. This fire protection plan must:
>
> > (i) Describe the overall fire protection program for the facility;
> >
> > (ii) Identify the various positions within the licensee's organization that are responsible for the program;
> >
> > (iii) State the authorities that are delegated to each of these positions to implement those responsibilities; and
> >
> > (iv) Outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage.
>
> (2) The plan must also describe specific features necessary to implement the program described in paragraph (a)(1) of this section such as—
>
> > (i) Administrative controls and personnel requirements for fire prevention and manual fire suppression activities;
> >
> > (ii) Automatic and manually operated fire detection and suppression systems; and
> >
> > (iii) The means to limit fire damage to structures, systems, or components important to safety so that the capability to shut down the plant safely is ensured.
>
> (3) The licensee shall retain the fire protection plan and each change to the plan as a record until the Commission terminates the reactor license. The licensee shall retain each superseded revision of the procedures for 3 years from the date it was superseded.

The remainder of 10 CFR 50.48 is not discussed in this application because it does not apply. Section 50.48(a)(4) of 10 CFR does not apply to combined license applications. Paragraph b of 10 CFR 50.48 does not apply to new plants and therefore to this application. Paragraph c of 10 CFR 50.48 is not utilized in this application. Paragraphs d and e of 10 CFR 50.48 are reserved. Paragraph e of 10 CFR 50.48 does not apply to this application.

Section 50.48 to 10 CFR refers to Criterion 3, “Fire protection,” of Appendix A, “General design criteria for nuclear power plants,” to 10 CFR Part 50. Appendix A to 10 CFR Part 50 does not apply to the Aurora, because the Aurora is not a light-water reactor. Instead, the parallel advanced reactor design criteria (ARDC) 3, “Fire protection,” from Regulatory Guide (RG) 1.232, “Guidance for developing principal design criteria for non-light-water-reactors,” Revision 0, issued April 2018, is used and is replicated as follows:

> Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and fire- resistant materials shall be used wherever practical throughout the unit, particularly in locations with structures, systems, or components important to safety. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

The purpose of this section is to describe how the requirements of 10 CFR 50.48(a)(1)-(3) and ARDC-3 to RG 1.232 are met.

6.1 Fire protection plan

In accordance with 10 CFR 50.48, each operating nuclear power plant must have a fire protection plan. The plan should establish the fire protection policy for the protection of structures, systems, and components at each plant and the procedures, equipment, and personnel required to implement the program at the plant site. The primary objectives of the Aurora Fire Protection Program (FPP) are to minimize both the probability of occurrence and the consequences of fire. To meet these objectives, the FPP is designed to provide reasonable assurance, through defense-in-depth, that a fire will not prevent the performance of necessary safe plant shutdown functions, and thereby will not increase the risk of radioactive releases to the environment.

For the Aurora, the structures, systems, and components of interest for protection against a fire are in the reactor module and in the control logic cabinetry. To support transients, the reactor trip system is also of interest for protection. It is important to highlight several design features of the Aurora that are conducive to inherently meeting the fire protection requirements, including:

- The reactor enclosures are filled with an inert gas, limiting potential for ignition,
- The only fluid circulating in the plant is carbon dioxide, which is traditionally used as a fire suppressant,
- Systems that may pose a fire threat are physically separated, and
- The site does not have a traditional switchyard, which is typically a comparatively high fire risk.

The Aurora FPP utilizes the concept of defense-in-depth in fire areas, with the following three objectives:

- Prevent fires from starting,
- Detect rapidly, control, and extinguish promptly those fires that do occur, and
- Provide protection for structures, systems, and components important to achieving a safe plant state.

II.07 Earthquake criteria

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7 EARTHQUAKE CRITERIA

7.0 Purpose

The purpose of this chapter is to meet Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(19), which requires “information necessary to demonstrate that the plant complies with the earthquake engineering criteria in 10 CFR Part 50, Appendix S.” Specifically, compliance with 10 CFR Part 50, Appendix S (IV)(a) is described in this chapter. Information regarding compliance with 10 CFR 100.23, “Geologic and seismic siting criteria,” as referenced by 10 CFR Part 50, Appendix S (IV)(b)-(c) is discussed in Chapter 1, “Site envelope and boundary.”

7.1 Seismic analysis

This section determines an extreme earthquake event that can be appropriately bounding for the U.S. and analyzes the Aurora against that event, in order to meet the underlying intent of 10 CFR Part 50 Appendix S (IV)(a)(1).

7.1.1 Overview of seismic analysis

7.1.1.1 Background

Seismic events have traditionally been considered the most bounding events for metal-fueled fast reactors, primarily due to the possibility of large positive reactivity insertions caused by control rod motion relative to the core lattice or reactor coolant sloshing. Reactivity challenges typically associated with seismic events are not of concern for the Aurora. These challenges typically include sloshing coolant, or oscillating control rods. Since the Aurora does not utilize a reactor coolant, there are no reactivity concerns associated with sloshing of the coolant. Additionally, since the Aurora does not operate with control rods, oscillation of rods is not of concern. The shutdown rods used in the Aurora are fully withdrawn a significant distance from the core during normal operation and would not pose a risk in the same manner as control rods which are inserted during normal operation. The other reactivity system is situated in the same structure as the reactor such that any oscillations would affect both the system and the reactor similarly or would place the reactor in a less reactive configuration.

The following design features of the Aurora ensure its resilience to earthquakes:

- A single powerhouse building eliminates concerns of differential displacement relative to connected structures in a seismic event.
- The rigidity of the reactor module results in minimal increased spectral acceleration.
- Passive decay heat removal is sufficient following reactor shutdown.
- No reactivity oscillations are possible from reactor coolant sloshing since heat pipes comprise the primary cooling pathway or from rod oscillations since the rods remain removed from the core during operations.

7.1.1.2 Analysis methodology

This seismic methodology considers an extreme earthquake to ensure the safety of the facility in a broad range of locations in the U.S. and is as follows:

1. Determine and describe the appropriately bounding earthquake for the U.S.
2. Determine the relevant structures, systems, or components (SSCs) that are potentially vulnerable to a seismic event and that require further analysis.
3. Analyze the relevant SSCs to determine impact of the appropriately bounding earthquake for the U.S.
4. Summarize the results of the seismic analyses to determine if the overall safety of the facility is impacted.

7.1.2 Description of a bounding earthquake magnitude

The purpose of this section is to describe the earthquake magnitude that adequately bounds the U.S. for the Aurora. The first step was to determine an appropriately bounding peak ground acceleration (PGA), which involved examining the ground motion response spectra (GMRS) of the existing U.S. operating nuclear fleet and accounting for other locations by the criteria in American Society of Civil Engineers (ASCE) 7. This examination resulted in a bounding PGA of 1.75 g, conservatively far greater than the specified maximum site-independent PGA of 1 g described in Regulatory Guide (RG) 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants", Revision 2, July 2014. Next, the bounding PGA was used in accordance with RG 1.60, to determine the appropriate spectral acceleration for the relevant components in the seismic analysis for the Aurora. Further information on this methodology is discussed in Appendix 7-A, "Determination of bounding earthquake," of this chapter.

7.1.3 Determination of relevant components

The purpose of this section is to determine which structures, systems, or components (SSCs) are potentially vulnerable during a seismic event. The primary consideration is whether a seismic event can result in an accident which would cause radioactive releases to exceed key regulatory limits. In accordance with the maximum credible accident (MCA), it was also considered if a seismic event could present a single cause for an accident with consequences which would exceed the consequences of the hypothesized MCA, presented in Chapter 5.1.

7.1.3.1 Progression of events

Following an extreme earthquake, the reactor trip system would receive a signal to release the shutdown rods. The reactor trip system monitors key reactor parameters, and, when appropriate, sends a reactor trip signal to the shutdown rod system, resulting in the insertion of the shutdown rods into the reactor core.

Shutdown is achieved by the shutdown rod system, which consists of three shutdown rods. Shutdown rods are composed of boron carbide powder within a stainless-steel cylinder. The rods are suspended above the core by electromagnets and insert via gravity drop when the electromagnets are de-energized, which introduces significant negative reactivity and consequently shuts the reactor down. It is important to note that only one shutdown rod is needed to ensure the reactor is in a sub-critical state, even at room temperature, and that no motor is required to allow for the rods to drop. As described in Chapter 2, "Description and analysis of structures, systems, and components," each shutdown rod travels vertically within a shutdown rod sleeve, which provides an unobstructed path for the rod to fully insert into the active core. The gap between the active portion of the shutdown rod and the walls of the sleeve, known as the channel, is 0.797 cm, which is sufficiently large to accommodate deformation of the sleeve without obstructing insertion of the rod.

Following a normal reactor shutdown, decay heat would be removed by the secondary system. In the event of an extreme earthquake, the secondary system is conservatively assumed to be inoperable. Decay heat rejection occurs through passive means to the air and structures in the powerhouse. These assumptions are consistent with the MCA analyzed and described in Chapter 5.1.

7.1.3.2 Characteristics of an extreme earthquake

The two characteristics, which could be associated with an extreme earthquake, that are considered for purposes of the seismic analysis are the resulting force from the ground acceleration and the potential collapse of structures. Specifically, analysis of the resulting force from the ground acceleration on the reactor module is of interest. Since the reactor module is densely packed of nearly all metal components,[16] a structural analysis of the reactor module is an appropriate indicator of the integrity of the internals. The complement to this analysis is the powerhouse collapse analysis to also analyze the structural integrity of the reactor module, specifically the module equipment housing.

7.1.3.3 Results of determination of relevant components

The relevant features of interest in the seismic analysis are the following:

- Ability for the shutdown rods to insert into the reactor
- Protection to the shutdown rod equipment, provided by the reactor module
- Integrity of the reactor module and internals

Ultimately, these three features can be analyzed through an evaluation of the reactor module integrity. If the reactor module does not reach structurally-challenging limits, then the reactor is shut down, and decay heat is passively rejected to the air and surrounding structures. These results would not exceed the consequences of the MCA.

7.1.4 Analysis assumptions and methodology

For purposes of this deterministic seismic analysis, this section is broken up into two subsections: (1) analysis of a large ground acceleration, and (2) analysis of a powerhouse collapse. The goal of the large ground acceleration analysis is to confirm that the reactor module integrity remains intact, which assures integrity of the internals. The goal of the powerhouse collapse is to analyze the reactor module integrity, specifically those portions that protect the shutdown rod equipment. If the reactor module integrity is upheld after an extreme ground acceleration and a full powerhouse building collapse, the safety of the reactor is unchallenged.

7.1.4.1 Ground acceleration analysis

The purpose of the ground acceleration analysis is to evaluate the structural effects on the reactor module, following a hypothetical extreme ground acceleration, as a result of a large earthquake. This analysis assumes a large earthquake occurs nearby the Aurora that disables heat removal by the secondary system and results in an automatic reactor trip. Therefore, this analysis focuses on showing that the structural limits of the reactor module are unchallenged.

7.1.4.1.1 Ground acceleration analysis assumptions

The ground acceleration analysis utilizes the response curve analysis based on the conservatively high PGA, as described in Appendix A, "External hazards evaluation.” This curve is then used as the basis for a modal analysis of the reactor module. The reactor module

[16] Further information of the reactor module components is located in Chapter 2.

is modeled as a single body, which is appropriate because of the similarity of materials present in the reactor module shell and internal components, as well as the rigidity of the internal components within the module. The shell model is determined to be an effective assumption because the width of the walls are less than 1/10 the diameter of the module. The total mass of the module, including the module shell and internal components, is 92.9 MT. The mass of the reactor module internals is accounted for by appropriately adjusting the mass density of the shell.

The reactor module is assumed to be rigidly mounted at the support flange. The location of the flange is essentially exactly at the foundation elevation of the building structure, such that no significant amplification of seismic accelerations is assumed. The analysis assumes a close coupling of the reactor module and the ground. The module is conservatively assumed to have the damping value of 5%, the smallest damping value outlined in RG 1.60. This assumption is conservative because there are site specific conditions that would dampen, or extremely reduce, the resulting forces from a ground acceleration on the reactor module. Therefore, soil structure interactions that result in additional damping are bound by this analysis.

The module shell is made of stainless-steel 304, and the mass of internal components is considered by adjusting the mass density of the body. Material properties used in the analysis assume a temperature of the module shell of 300 C, which is conservatively high in comparison to the temperatures expected during operation. As material properties, namely the elastic modulus and yield strength, degrade with increased temperature, the assumption of temperature is conservative and bounding. Properties used in the analysis are summarized in Table 7-1 [17].

Table 7-1: Material properties at 300 C

Property	Value
Mass, internals (kg)	81168
Volume, shell (m^3)	1.512
Density, shell (kg/m^3)	7794
Density, total (kg/m^3)	61494
Elastic modulus (GPa)	176
Poisson's ratio	0.31
Yield strength (MPa)	129

A three-dimensional finite element model is analyzed using modal and response spectrum analyses in ANSYS Mechanical. The reactor module is modeled with a refined mesh to adequately capture localized failure modes.

The response spectrum analysis considers two horizontal directional response and one vertical response. The horizontal response spectrum is based on the bounding spectrum defined in Section 7.1.2. The vertical response spectrum is conservatively assumed to be equal to the horizontal response spectrum. The three orthogonal responses are combined using the square-root-sum-of-the-squares method. Based on the results of the modal analysis, 300 modes are considered in the evaluation, resulting in 100 percent modal mass participation.

7.1.4.1.2 Ground acceleration results

The first step was to determine the dominant modes of the reactor module so the corresponding spectral acceleration could be applied. The dominant modes and associated frequencies are given in Table 7-2. Due to the simplicity of the model, most of the participating modal mass, greater than 67%, is within the first two mode shapes. The 18.8 Hz mode shapes correspond to cantilever, or flexural, mode shapes applicable to the reactor module. Higher mode shapes are primarily circumferential in nature.

Table 7-2: Dominant modes and associated frequencies of modal analysis

Mode	Frequency (Hz)
1	18.8
2	18.8
3	21.5
4	21.5
5	26.8
6	26.8
7	29.5
8	29.5
9	39.8
10	39.8

The two parameters of interest for the reactor module for the large ground acceleration analysis were the peak equivalent (von Mises) stress and horizontal displacement of the reactor module. The stresses in the reactor module were analyzed to assure that the reactor module did not reach material limits that could result in failure and to evaluate any concerns related to deflection of the module sufficient to distort internals so that the shutdown rods could not drop. The von Mises stress in the module is shown in Figure 7-1. The maximum stress in the shell is 40.6 MPa, which is substantially lower than the conservative yield strength of 129 MPa. The horizontal displacement of the module is shown in Figure 7-2. The maximum horizontal displacement, taken at the lowest tip of the module which experiences the greatest displacement, is 2.1 mm. Since the gap in the shutdown rod sleeve is approximately 7.9mm, a 2.1 mm maximum displacement of the entire module would not compromise the ability for the shutdown rods to drop.

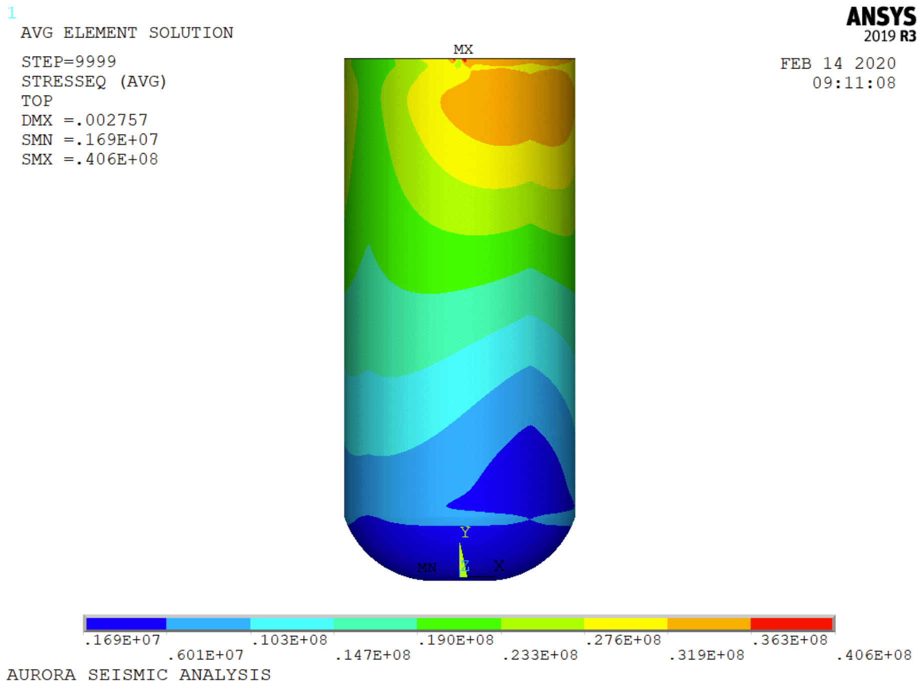

Figure 7-1: Equivalent (von Mises) stress under extreme earthquake

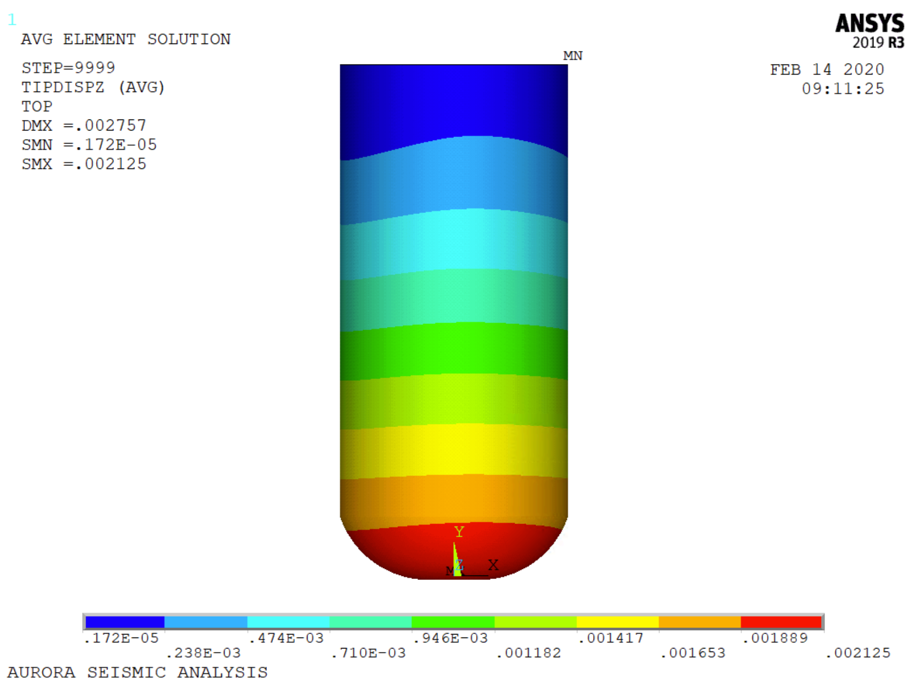

Figure 7-2: Horizontal displacement under extreme earthquake

This analysis concludes that the reactor module experiences mechanical loads within its material limits, confirming that the integrity of the module is maintained, and also that the shutdown rod sleeve integrity will not be compromised due to deflection.

7.1.4.2 Aurora powerhouse collapse analysis

The purpose of the Aurora powerhouse collapse analysis is to evaluate the structural effects on the module equipment housing following a hypothetical complete collapse of the powerhouse. This analysis assumes a full powerhouse collapse in order to bound all external hazards that could pose a challenge to the powerhouse. This analysis assumes a full powerhouse collapse disables the secondary system since the secondary system is located on the first floor of the powerhouse, directly underneath the powerhouse roof. The result of the secondary system being disabled is an automatic reactor trip, if not already triggered by another secondary failure. Therefore, this analysis focused on the ability of the reactor to be shut down by the shutdown rods, following an automatic reactor trip signal. To maintain shutdown functionality, the integrity of the reactor module must be upheld. Specifically, the subcomponent of the reactor module analyzed is the module equipment housing, which functions to protect equipment such as the shutdown rods. The goal of this analysis is to demonstrate the integrity of the module equipment housing through an impact analysis of the heavy powerhouse components.

7.1.4.2.1 Powerhouse collapse analysis assumptions

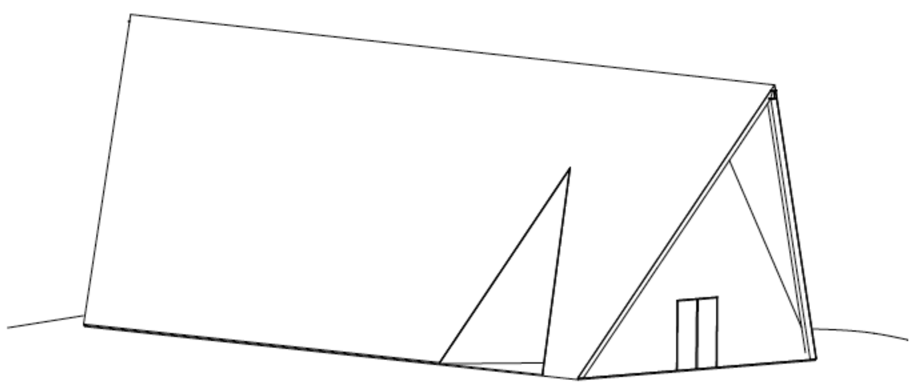

Figure 7-3: Aurora powerhouse

The Aurora powerhouse is an A-frame, as shown in Figure 7-3, which has a relatively small footprint of less than 5,000 sq. ft. The module equipment housing is a reactor module component and is located in the basement of the powerhouse as shown in Figure 7-4. One of the functions of the module equipment housing is to protect the shutdown rod equipment, which is the function of interest to this analysis. The module equipment housing, which is made of stainless-steel 304, is assumed to have an ultimate tensile strength of 517 MPa [17]. The considerations analyzed the impacts from a collapsed roof, crane, and floor to assess the deformation and penetration damage on the module equipment housing following a powerhouse building collapse.

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Figure 7-4: Module equipment housing location

The powerhouse includes several heavy components that could cause damage to the module equipment housing upon collapse. The falling objects considered were the heavy objects in the powerhouse that had a line of sight to the module equipment housing. These included the roof, the crane, and the floor, with the following general assumptions:

- The roof falling objects included several A-frame roof beams, which are arranged in a triangle formation and are supported at the base and top of the A-frame, and of standard steel roof construction.

- The crane falling object included a single girder crane type, which is underhung, and supported by a standard I-beam girder.

- The floor falling object included the first floor of the powerhouse, which is located directly above the module equipment housing, and was conservatively assumed to be thick concrete.

These falling objects can be seen in Figure 7-5.

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Figure 7-5: Schematic of falling objects considered

The frame roof beams have the following characteristics:

- W14 x 30 steel member
- 50 foot maximum length
- 42 foot maximum drop height

The crane beam has the following characteristics:

- W18 x 119 or S24 x 106 steel member[17]
- 25 foot maximum length
- 17 foot maximum drop height

[17] Assumed to govern the crane portion because it is the heaviest section.

The floor section has the following characteristics:

- 10 ton maximum weight
- 1.5 foot maximum drop height
- 15 foot by 10 foot smallest panel size

The most conservative assumptions were used, which include the following:

- The falling object (roof, crane, floor) has direct line of sight to the module equipment housing from the maximum height
- No protection of the module equipment housing is taken credit for, including the powerhouse structure, floor, or ground
- Full force of impact from the falling object, from maximum height, is assumed
- All impact loads scenarios are conservatively assumed to be from rigid sources
- All gravitation potential energy is conservatively assumed to be converted to an equivalent kinetic energy for impact load scenarios
- Impact load scenario sources are assumed to strike the module equipment housing in a way which produces conservative results (i.e. smallest cross-sectional area assumed as impact area but edges or sharp ends in geometry are not assumed since this would be overly conservative)

BC-TOP-9A, Revision 2, "Topical Report - Design of structures for missile impact," issued September 1974, is used to calculate the minimum module equipment housing thickness, following an impact from the falling objects.

7.1.4.2.2 Powerhouse collapse results

The first step was to determine which falling object to analyze to obtain maximum impact to the module equipment housing. From these falling objects, the roof (i.e., roof beams) were determined to cause the maximum impact damage to the module equipment housing and were used in this analysis.

Hand calculations are used using two different missile impact evaluation methods from BC-TOP-9A. Potential energies are determined for each missile scenario and then the Ballistic Research Laboratory (BRL) formula (i.e., equation 2-7 from BC-TOP-9A) and the Stanford Research Institute (SRI) formula (i.e., equation C-12 from BC-TOP-9A) are used to calculate the required module equipment housing thickness. The BRL formula presented a larger thickness that the SRI formula and governed the analysis.

Even following such an extreme collapse scenario, the thickness of the module equipment housing was found to be great enough such that no penetration was experienced from the falling roof beam. Therefore, the powerhouse collapse analysis found that the thickness of the module equipment housing was sufficient to withstand impact from falling objects due to a full building collapse.

7.1.5 Review of results

The purpose of the seismic analysis is to assure that the safe state of the reactor is not challenged by an extreme earthquake. The extreme earthquake is developed to sufficiently bound the U.S. for the Aurora in Appendix 7-A, "Determination of bounding earthquake," to this chapter. The seismic analysis examines the effects of a resulting force from the extreme ground acceleration and the effects from a building collapse. The seismic analysis demonstrates the robust design of the Aurora in its ability to maintain the integrity of the reactor module, ultimately concluding that the overall safe state of the reactor is not challenged by extreme seismic events. No SSCs are uniquely compromised by even a very conservatively large PGA and resultant response spectrum, and seismic does not present a safety challenge worse than the MCA analyzed. Because no SSCs need to be specifically designed to withstand a large seismic event, they are not further seismically classified and do not have required seismic testing.

7.2 Operating basis earthquake

7.2.1 Determination of operating basis earthquake

The purpose of this section is to determine the operating basis earthquake (OBE) in accordance with 10 CFR Part 50 Appendix S (IV)(a)(2). The operating basis earthquake (OBE) is set to one-third of the horizontal ground acceleration of the SSE, which in this analysis is interpreted as the bounding earthquake used; therefore, the requirements of Paragraph (a)(2)(i)(B)(I) are satisfied without explicit response or design analyses, in accordance with Paragraph (a)(2)(i)(A) of Appendix S to 10 CFR Part 50. The resulting OBE is 0.58 g PGA.

7.2.2 Required plant shutdown

The purpose of this section is to meet 10 CFR Part 50 Appendix S (IV)(a)(3):

> If vibratory ground motion exceeding that of the Operating Basis Earthquake Ground Motion or if significant plant damage occurs, the licensee must shut down the nuclear power plant. If systems, structures, or components necessary for the safe shutdown of the nuclear power plant are not available after the occurrence of the Operating Basis Earthquake Ground Motion, the licensee must consult with the Commission and must propose a plan for the timely, safe shutdown of the nuclear power plant. Prior to resuming operations, the licensee must demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public and the licensing basis is maintained

After a seismic event that registers near or exceeds the OBE, operating procedures provide guidance on evaluating possible structural impacts to the facility. If the seismic event exceeds the operating basis earthquake, the reactor is shut down prior to any inspections of structural impacts.

7.3 Seismic instrumentation

The purpose of this section is to meet to 10 CFR Part 50 Appendix S (IV)(a)(4):

> Suitable instrumentation must be provided so that the seismic response of nuclear power plant features important to safety can be evaluated promptly after an earthquake.

Seismic data is monitored based on onsite seismic instrumentation and is accessible to Onsite Monitors. Seismic instrumentation is expected to include triaxial acceleration sensor units connected to a time-history analyzer in order to determine the magnitude and duration of the ground acceleration.

APPENDIX 7-A: DETERMINATION OF BOUNDING EARTHQUAKE

Part of the seismic analysis for the Aurora is to determine an appropriately bounding extreme earthquake for the U.S. The purpose of this appendix is to define that appropriately bounding extreme earthquake for the U.S. through a set of conservative seismic design parameters.

The methodology presented here is novel but based on key concepts from guidance, as well as applying abundantly conservative margins, which are possible because of the simplicity of the Oklo Aurora design. According to Regulatory Guide (RG) 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants", Revision 2, July 2014, the determination of seismic design response spectra can be done in a site-independent manner. This begins by the determination of a peak ground acceleration and scaling response spectra. In RG 1.60, this maximum ground acceleration is given as 1 g. However, for defense-in-depth to ensure a truly bounding response spectra, and to enable use of data across the United States available from USGS and ASCE, this was compared with a range of site-specific Ground Motion Response Spectra (GMRS) from existing plants in areas with significant seismic considerations. Nuclear power plants collect site-specific seismic data with a frequency of seismic events with a risk target of once every 10,000 years. This data is only available for locations with sited nuclear power plants, and there is no readily-available seismic data for the entire U.S. at this fidelity. There is readily-available data for the majority of the U.S. from the American Society of Civil Engineers (ASCE) Standard, ASCE 7, issued in 2010, which corresponds to a risk target of structural collapse of 1% in 50 years. Since the risk targets between the nuclear power plants data and the ASCE 7 data are not the same, this appendix describes how the ASCE 7 data may be scaled to adequately encompass the U.S. at the same risk target as the data currently being collected by nuclear power plants. Further discussion on what conclusions may be drawn from these analyses is provided at the conclusion.

7-A.1: Seismic data

7-A.1.1: Seismic data from U.S. operating plants

After the March 2011 Fukushima Earthquake event, the U.S. Nuclear Regulatory Commission (NRC) issued a demand for information from all U.S. operating nuclear plants to re-evaluate their respective plants using the latest seismic hazard and regulatory guidance. As part of their responses, all plants provided site-specific GMRS to the NRC for screening purposes. GMRS are defined as a performance-based seismic ground motion that is developed in accordance with RG 1.208, issued March 2007, "A performance-based approach to define the site-specific earthquake ground motion."

7-A.1.2: Seismic data from American Society of Civil Engineers

ASCE 7 is a standard which guides development and use of an array of site-specific data to provide minimum load requirements for the design of buildings and other structures that are subject to building code requirements. Of those types of data, ASCE 7 provides detailed seismic maps and seismic data based on data from the USGS for a given site, including peak ground acceleration.

All data taken from ASCE 7 were used assuming the highest risk category, Risk Category IV, and the least damping soil type, i.e. the "hard rock" soil type.

7-A.2: Comparing the GMRS to ASCE 7

The first step in comparing the GMRS and ASCE 7 is to directly compare the data provided by each in a range of specific sites. This is done by examining the 10 operating plants with the highest associated seismicity, the locations of which are approximated with the Structural Engineers Association of California (SEAOC) interface,[18] and are shown in Figure 7A - 1.

[18] "Approximate locations" as used in this analysis are within 25 miles of the operating nuclear power plant. The SEAOC is available as a web interface through www.seismicmaps.org and links to United States Geologic Survey seismic data.

Figure 7A - 1: Map of the operating plants with the 10 highest GMRS

For these 10 operating nuclear power plants, ASCE 7 peak ground acceleration and peak spectral acceleration design values are also obtained; the parameters assume a Risk Category IV building structure and either Site Class B (rock) or Site Class D (stiff soil). Comparisons between the PGA and peak spectral acceleration values derived from the GMRS and ASCE 7 are shown in Table 7A - 1. The mean ratio of GMRS PGA to ASCE 7 PGA is 3.46 and the mean ratio of GMRS peak spectral acceleration to ASCE 7 peak spectral acceleration is 2.64. As can be seen by the range of ratios, GMRS PGA and ASCE PGA do not have a linear relationship, and the GMRS takes into account a design factor as described in RG 1.208. However, of note is that the maximum GMRS PGA of any plant is the Diablo Canyon plant, at 0.80 g. This is to be expected, because Diablo Canyon is located very near a large fault, in one of the most seismically active regions in the entire U.S.

Table 7A - 1: Comparison of ASCE 7 and GMRS seismic parameters

Selected plants	Approximate location	ASCE 7 PGA (g)	ASCE 7 S_s (g)	ASCE 7 S_s/PGA	GMRS PGA (g)	GMRS Peak S_a (g)	GMRS PGA/ASCE 7 PGA	GMRS Sa/ GMRS PGA
Columbia	Pasco, WA	0.17	0.39	2.35	0.25	1.45	1.51	5.80
Vogtle	Augusta, GA	0.15	0.29	1.95	0.44	1.09	2.95	2.48
Callaway	Jefferson City, MO	0.10	0.20	2.08	0.50	1.15	5.21	2.30
Pilgrim	Plymouth, MA	0.10	0.19	1.89	0.50	1.18	5.10	2.36
North Anna	Richmond, VA	0.10	0.19	1.82	0.57	1.26	5.59	2.21
Diablo Canyon	San Luis Obispo, CA	0.46	1.15	2.49	0.80	2.00	1.73	2.50
Seabrook	Portsmouth, NH	0.15	0.27	1.75	0.50	1.05	3.25	2.10
Indian Point	New York City, NY	0.17	0.28	1.66	0.40	0.85	2.40	2.13
Oconee	Greenville, SC	0.14	0.28	1.97	0.40	0.85	2.82	2.13
Peach Bottom	Lancaster, PA	0.10	0.18	1.83	0.40	0.97	4.00	2.43
Average							**3.46**	**2.64**

7-A.3: Seismic parameters for representative sites

The next step of this analysis is to estimate seismic parameters for U.S. locations where there are currently no sited operating plants. Nine sites are selected to conservatively evaluate the U.S., based on a wide range of locations. These sites included Alaska, Puerto Rico, St. Thomas, and Hawaii, as shown in Figure 7A - 2. The assumed ASCE 7 site condition for these sites was Class C (very dense soil and soft rock). The ASCE 7 PGA for these sites ranged from 0.27 g to 0.5 g and are shown in Table 7A - 2.

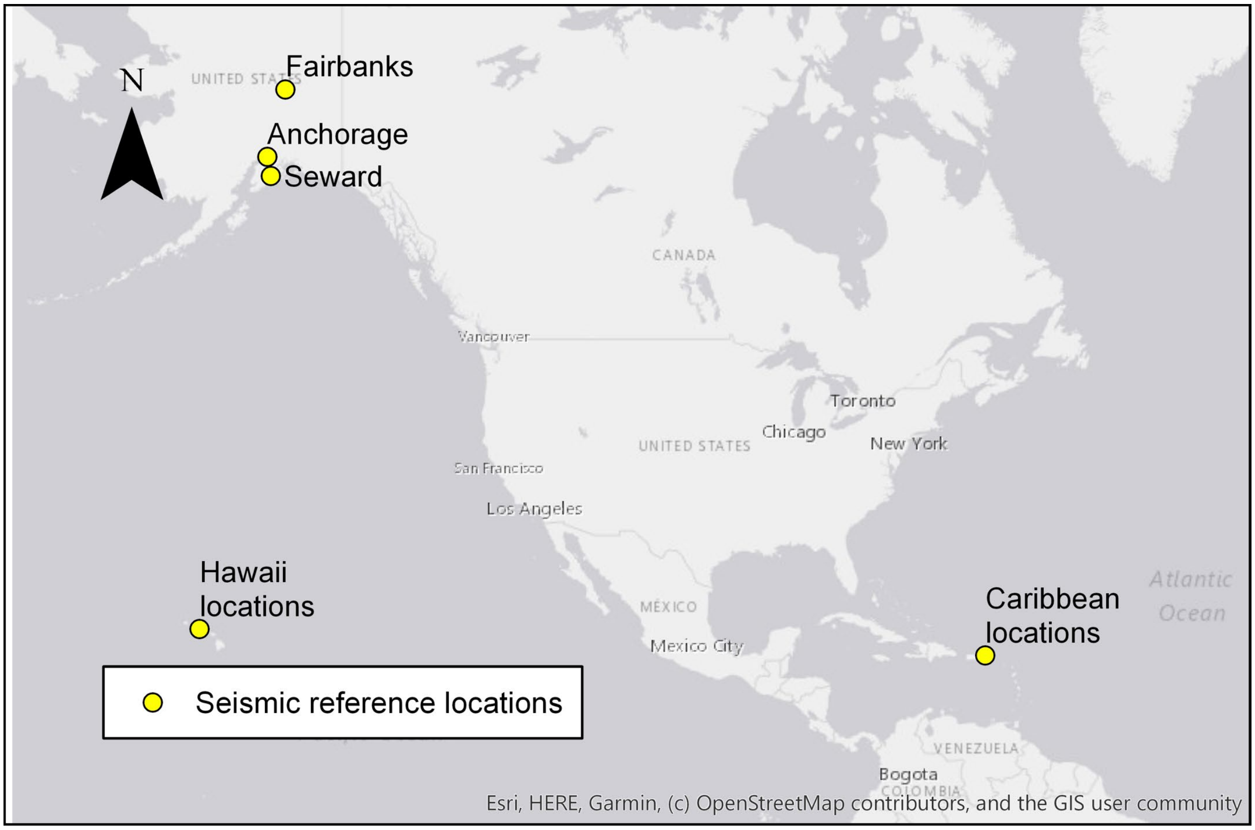

Figure 7A - 2: Representative U.S. sites

For estimating the GMRS-equivalent seismic values for the nine U.S. sites, the ASCE 7 PGA values are multiplied by the average scale factors derived in Table 7A - 1. The estimated GMRS-equivalent PGA for each site is found by applying a factor of 3.46 to the ASCE 7 estimated PGA. Similarly, for estimating peak spectral acceleration for each site, a factor of 2.64 is applied. As can be seen in Table 7A - 2, the maximum GMRS-equivalent PGA is found to be 1.73 g and the GMRS-equivalent peak spectral acceleration is found to be 4.55 g.

Table 7A - 2: Seismic parameters for representative U.S. sites

					Scaled values	
Representative location	ASCE 7 PGA (g)	ASCE 7 Ss (g)	PGA ratio (GMRS PGA /ASCE 7 PGA)	Sa ratio (GMRS Sa /GMRS PGA)	Calculated GMRS PGA (g) (ASCE 7 PGA * PGA ratio)	Calculated GMRS Peak Sa (g) (Calculated GMRS PGA *Sa ratio)
Fairbanks, AK	0.4	0.99	3.45	2.64	1.38	3.64
Anchorage, AK	0.5	1.5	3.45	2.64	**1.73**	**4.55**
Seward, AK	0.5	1.5	3.45	2.64	**1.73**	**4.55**
San Juan, Puerto Rico	0.41	0.99	3.45	2.64	1.41	3.73

Mayaguez, Puerto Rico	0.47	1.24	3.45	2.64	1.62	4.28
Charlotte Amalie, St. Thomas	0.49	1.23	3.45	2.64	1.69	4.46
St. Thomas, St. Thomas	0.49	1.24	3.45	2.64	1.69	4.46
Honolulu, Hawaii	0.27	0.58	3.45	2.64	0.93	2.46
Maui, Hawaii	0.37	1.03	3.45	2.64	1.28	3.37

7-A.4: Results

Since this analysis found that most U.S. sites fall below a GMRS-equivalent PGA of 1.73 g, the bounding earthquake PGA value is set to 1.75 g. This far exceeds the guidance in RG 1.60 regarding a maximum ground acceleration, i.e. PGA, of 1 g. Next the methodology in RG 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” Revision 2, July 2014, is applied to develop a design response spectrum to approximate the appropriate spectral accelerations of relevant components. This design response spectrum used in the seismic analyses is given in Table 7A - 3.

Table 7A - 3: Design response spectrum used in seismic analysis

Frequency (Hz)	Spectral acceleration (g)
0.10	0.11
0.25	0.70
0.50	1.23
1.00	2.63
1.50	3.50
2.50	5.48
9.00	4.57
33.00	1.75
100.00	1.75

This spectrum is then used in the modal analysis in a conservative manner as compared with guidance in RG 1.60 by assuming that this spectrum could be applied in both horizontal dimensions as well as the vertical dimension, by using the root mean of squares method as described.

II.08 Unresolved and generic safety issues

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8 UNRESOLVED AND GENERIC SAFETY ISSUES

8.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(20) requires the following:

> Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design;

The related regulatory guidance, NUREG-0933, "Resolution of generic safety issues," issued December 2011, contains a database of over 200 generic safety issues and is available online.

The purpose of this section is to describe the process used to determine if any generic safety issues, as identified in NUREG-0933, apply to the Aurora and to disposition any that do.

8.1 Review of NUREG-0933

A spreadsheet, "Generic issues dataset,"[19] from NUREG-0933 contains a complete set of the issues, titles, categorizations, and facility types. The spreadsheet was downloaded on February 27, 2020, which is within 6 months of the submission of the license application. The key header rows that allow the issues to be filtered and sorted are as follows:

- Facility types: assumed to specify the applicability of the issues to different facilities (e.g., light water reactors, pressurized water reactors, boiling water reactors, new reactors)
- Status/priority ranking: assumed to describe whether the issue has been resolved and its priority (which must be medium or high to necessitate consideration)

This allows the issues to be filtered down to only the ones that have not been stated as "resolved" and are deemed applicable to "new reactors" and "non light power reactors."

NUREG-0933, Appendix G, "Generic issues program current and historical procedures," further describes the prioritization of these issues as either high, medium, low, or drop. Only the high and medium generic safety issues require consideration for this application under 10 CFR 52.79(a)(20). Since the licensing issues are not medium or high priority and not directly related to protecting public health and safety or the environment, they are not considered for this application.

Issue 89, highlighted in orange, appears to be the only generic issue with a priority of medium or high that is applicable to new reactors and is further described in Table 8-1.

[19] Available for download at https://www.nrc.gov/sr0933/.

Table 8-1: Overview of generic issue 89

Issue title	Historical background	Description of safety significance
89: Stiff Pipe Clamps	This issue was identified following a staff evaluation of allegations that improper consideration of "stiff" pipe clamps in Class 1 piping systems could result in unsafe plant operation.	Type 1 seismic-induced pipe breaks, resulting in LOCA and/or reactor transients; and Type 2 pipe breaks in Class 1 piping, resulting from dynamic loads following LOCAs and transients.

Since there are no pipe clamps in the Aurora design, this generic issue does not apply.

The most updated version of NUREG-0933 has been considered, prior to submitting this combined license application. There are no unresolved or medium- and high-priority generic safety issues that are technically relevant to the Aurora design.

II.09 Emergency planning

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9 EMERGENCY PLANNING

9.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(21) requires, "emergency plans complying with the requirements of § 50.47 of this chapter, and 10 CFR part 50, Appendix E." The objective of the Emergency Plan is to provide a basis for action, to identify personnel and material resources, and to designate areas of responsibility for coping with any emergency at the Aurora Idaho National Laboratory (INL) site. The level of emergency preparedness provided in this plan is commensurate with the potential consequences to public health, safety, the common defense, and security at the Aurora INL site. The Emergency Plan is submitted as a separate document in Part VII, "Enclosures."

9.1 Background

Due to Oklo's unique market demands for a relatively small power source, the Aurora reactor has similar power level and quantities of nuclear material to a nonpower reactor. Although nonpower reactors vary in size, they are typically on the order of 10 Megawatts thermal (MWth) or less[20], which is two orders of magnitude less than a commercial power reactor. Similarly, the amount of nuclear material in the Aurora reactor is at least an order of magnitude less than a large light water reactor as shown in Table 9-1. This substantial reduction in radioactive material is one key to the inherent safety of the Aurora because it limits the maximum possible radionuclide release, in addition to the inherent safety characteristics shown in Chapter 2 and Chapter 5.1.

Table 9-1: Comparison of current large light water reactor to Aurora

	Current large light water reactors	Aurora
Power output (MWth)	1600-4400	<5
Refueling cycle (years)	1.5-2	None
Radionuclide inventory (metric tons)	100-150	<5
System pressure (atm)	150	Near atmospheric
Hydrogen explosion risk	Yes	No
Cooling	Loop with low thermal inertia	Passive heat pipes
Electric power dependence	Relies on offsite power or emergency diesel generation	No safety-related electric power dependence
Negative reactivity coefficient	Yes	Yes

9.2 Guidance

In the development of the Emergency Plan and the size of the emergency planning zone (EPZ), Oklo has taken the following guidance into consideration:

[20] Nonpower reactors vary in power level from 0.000005 MWth to 20 MWth. Roughly 94% of nonpower reactors are licensed at or below 5 MWth according to the NRC in the publicly available dataset, "Operating U.S. Nuclear Research and Test Reactors - Regulated by the NRC." This data set was last updated on July 1, 2016.

- NUREG-0654, Revision 2, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” issued December 2019

- NUREG-0396, Revision 0, “Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants,” issued December 1978

- DG-1350, “Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production and Utilization Facilities”

- ANSI/ANS-15.16-2015, “Emergency Planning for Research Reactors”

As specified in NUREG-0654, this Emergency Plan provides for the following:

- Suitable measures are taken to protect any people located within the EPZ.

- Responsibilities during an emergency are assigned to properly trained personnel.

- Emergency action levels (EALs) are defined such that events can be appropriately classified assisting in fast and accurate decision making during an emergency.

- Sufficient equipment and data are available to personnel with emergency responsibilities to assess the appropriate EAL.

- Sufficient training, equipment, and procedures are in place to detect and mitigate the consequences of an onsite emergency.

As specified in DG-1350, this Emergency Plan provides for the following:

- The EPZ boundary provides public protection from dose levels above a 1 rem total effective dose equivalent threshold.

As specified in ANSI/ANS-15.16, this Emergency Plan provides for the following:

- The EAL radiological thresholds are in line with the limitations established for nonpower reactors.

- The general structure of this emergency plan comes from ANSI/ANS-15.16.

This Emergency Plan provides information regarding emergency preparedness and response planning for the Aurora reactor by the Materials and Fuels Complex site at the Aurora INL site. The provided information addresses organizational responsibilities, capabilities, actions, and guidelines for collaborating with community response organizations during an onsite emergency. Oklo Power LLC (Oklo Power) will be working closely with the Idaho Department of Energy (DOE-ID) for emergency support services as per the Site Use Permit.

9.3 Emergency planning zone

9.3.1 Background for emergency planning zones

As part of emergency planning, an EPZ must be identified. Postulated radioactive releases from credible accidents provide the basis for determining the size of the EPZ. NUREG-0654 defines an EPZ as "the areas for which planning is needed to assure that prompt and effective actions can be taken to protect the public in the event of an accident." NUREG-0654, authored by a task force comprised of representatives from the Federal Emergency Management Agency and the Nuclear Regulatory Commission, includes the following language:

> The primary objective of radiological emergency planning is to provide dose savings for a spectrum of radiological incidents that have the potential to produce offsite doses in excess of the current Federal protective action guides (PAGs).

The Environmental Protection Agency (EPA) determines the PAGs, which are the basis for emergency response. The precedent, set by all currently licensed reactors, bases the plume exposure pathway EPZ size to the distance which a radioactive plume could result in possible dose to the public in excess of the PAGs. The PAGs dose level concerns for determining the EPZ has been the precedent for determining radiological harm to the public since 1979 when the U.S. Nuclear Regulatory Commission (NRC) incorporated NUREG-0396 guidance through a policy statement[21]. In NUREG-0396, the combined NRC EPA task force concluded that "the objective of emergency response plans should be to provide dose savings for a spectrum of accidents that could produce offsite doses in excess of the PAGs." The early phase limits for the public taking protective actions is 1-5 rem projected dose over four days as given in the PAG Manual, "Protective Action Guides and Planning Guidance for Radiological Incidents," published January 2017. As the basis in NRC emergency planning guidance is for the size of the EPZ to be based on the PAGs, an offsite emergency preparedness plan needs to exist if there is a possibility of an accident which would result in a 1 rem projected dose[22] to a member of the public.

The Emergency Plan provides sufficient planning to ensure appropriate preparation for responding to credible events through onsite emergency event planning. There are two types of EPZs considered: (1) the plume exposure pathway and (2) the ingestion exposure pathway EPZ. The plume exposure pathway EPZ is primarily concerned with limiting the radiation exposure to the public and the inhalation of airborne radioactive contamination. The primary concern of the ingestion exposure pathway EPZ is the ingestion of food and drink which has been contaminated by radioactivity.

9.3.2 Emergency planning zone boundary and goal

For the Aurora, the plume exposure and ingestion exposure pathway comprise the same EPZ, which is limited to the exterior boundary of the Aurora powerhouse. As there is no radiological release associated with the maximum credible accident (MCA), the PAGs are met through an

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[21] The NRC's policy statement incorporating NUREG-0396 was released October 23, 1979 in 44 FR 61123.

[22] As stated in the PAG Manual "Projected dose is the sum of the effective does from external radiation exposure (e.g. groundshine and plume submersion) and the committed effective dose from inhaled radioactive material."

EPZ limited to the Aurora powerhouse. The MCA is discussed in Chapter 5.1, "Transient analysis."

The lower limit of the PAGs, 1 rem projected over 4 days, provide the EPZ boundary goal for the Emergency Plan. Due to the small size of the EPZ, there is no offsite emergency planning necessary and emergency planning is limited to onsite only. Therefore, the Emergency Plan for the Aurora contains only one emergency class, a Notification of Unusual Events, and describes onsite emergency response only.

9.3.3 Emergency Plan description

In accordance with 10 CFR 50.47, " Emergency plans," and Appendix E, "Emergency planning and preparedness for production and utilization facilities," to 10 CFR Part 50, the Emergency Plan includes the following:

- Description of the overall emergency planning for the facility
- Identification of the various positions and their authorities within the organization that are responsible for the program
- Description of the emergency classification system
- Description of the onsite emergency response, including, activation of the organization, response to the emergency, and onsite evacuations
- Description of the emergency facility and equipment
- Description of the recovery actions needed to be taken following an emergency
- Description of the administrative controls for the maintenance of the Emergency Plan

II.10 Emergency planning with state and local governments

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10 EMERGENCY PLANNING WITH STATE AND LOCAL GOVERNMENTS

10.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(22) requires that:

> (i) All emergency plan certifications that have been obtained from the State and local governmental agencies with emergency planning responsibilities must state that:
>
> > (A) The proposed emergency plans are practicable;
> >
> > (B) These agencies are committed to participating in any further development of the plans, including any required field demonstrations; and
> >
> > (C) These agencies are committed to executing their responsibilities under the plans in the event of an emergency;
>
> (ii) If certifications cannot be obtained after sustained, good faith efforts by the applicant, then the application must contain information, including a utility plan, sufficient to show that the proposed plans provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site.

The purpose of this section is to show compliance with 10 CFR 52.79(a)(22).

10.1 Evaluation

Because the Aurora EPZ is limited to the powerhouse building itself, the Emergency Plan, as submitted and described in Part VII, “Enclosures,” has been generalized for applicability across the U.S. and is generally not specific to any one State or locality.

It is important to develop and maintain strong relationships with State and local governmental agencies, including security, medical, ambulance, and fire services. For the Aurora Idaho National Laboratory (INL) site, emergency response is augmented by the national laboratory and does not necessitate any further relationships with the state or local governments. Intent of this relationship is documented in the Site Use Permit (NO. DE-NE700105).

1. The Site Use Permit between the U.S. Department of Energy (DOE) and Oklo requires a memorandum of agreement (MOA) to address access and control procedures to the Aurora powerhouse. The MOA will define items such as badging and access controls, emergency response procedures, security, and any activities that may require coordination with DOE.

2. The Site Use Permit also defines a Site Services Agreement which will describe which, if any, services Oklo expects or desires the DOE or its contractors to provide or supply. This Site Services Agreement will describe the service and the cost structure for such services. Examples of these services include, but are not limited to, security, emergency response, transportation, power, sanitation, and roads maintenance.

Oklo Power will continue to work closely with INL in establishing joint emergency response plans and operations.

II.11 Prototype operational conditions

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11 PROTOTYPE OPERATIONAL CONDITIONS

11.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR*)* Section 52.79(a)(24) requires that paragraph e of 10 CFR 50.43, "Additional standards and provisions affecting class 103 licenses and certifications for commercial power," be met and requires that applicants demonstrate their safety features through either prototype testing or sufficient analysis, testing and experimentation for designs that differ significantly from pre-1997 light water reactor designs or use passive safety means to accomplish safety functions. Section 50.43(e) to 10 CFR states the following:

> Applications for a design certification, combined license, manufacturing license, operating license, or standard design approval that propose nuclear reactor designs which differ significantly from light-water reactor designs that were licensed before 1997. Or use simplified, inherent, passive, or other innovative means to accomplish their safety functions will be approved only if:
>
> (1)(i) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;
>
> (ii) Interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof; and
>
> (iii) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions; or
>
> (2) There has been acceptable testing of a prototype plant over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If a prototype plant is used to comply with the testing requirements, then the NRC may impose additional requirements on siting, safety features, or operational conditions for the prototype plant to protect the public and the plant staff from the possible consequences of accidents during the testing period.

The purpose of this section is to explain how the Aurora design meets the requirements of 10 CFR 50.43(e).

11.1 Evaluation

The safety of the Aurora design is based largely on analysis that utilizes operational experience data. Therefore, a prototype reactor, under 10 CFR 50.43(e)(2), is not intended to be constructed; instead, compliance is shown with 10 CFR 50.43(e)(1). The Aurora reactor and associated systems are described in Chapter 2 "Description and analysis of structures, systems, and components." The safety analysis that corresponds to those systems is in Chapter 5.1, "Transient analysis." Subsequent testing is expected as part of preoperational and startup testing and is detailed in Chapter 14, "Preoperational testing and initial operations."

II.12 Quality Assurance Plan Description

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12 QUALITY ASSURANCE PLAN DESCRIPTION

12.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(25) requires that the final safety analysis report includes:

> "A description of the quality assurance program, applied to the design, and to be applied to the fabrication, construction, and testing, of the structures, systems, and components of the facility. Appendix B to 10 CFR part 50 sets forth the requirements for quality assurance programs for nuclear power plants. The description of the quality assurance program for a nuclear power plant must include a discussion of how the applicable requirements of Appendix B to 10 CFR part 50 have been and will be satisfied, including a discussion of how the quality assurance program will be implemented"

12.1 Overview

Oklo has separately previously submitted its Quality Assurance Plan Description (QAPD) to the NRC as a topical report. This QAPD is for the design and construction scope. Once the SER is issued for this QAPD scope, which is anticipated in the near term, the modifications to the QAPD for the operations scope will be submitted as a topical report revision.

II.13 Organizational structure for operations

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13 ORGANIZATIONAL STRUCTURE FOR OPERATIONS

13.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(26) requires that "the applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements for operation" be provided.

This chapter describes the organizational structure of Oklo Power LLC (Oklo Power) management, its plant operations, the associated functions and responsibilities, and the personnel qualifications requirements for operation. Oklo Power expects to act as both the owner and operator of its plants. As the Aurora is expected to operate nearly automatically, many of the operational roles of traditional reactors are unnecessary. Onsite personnel do not perform any credited operator actions.

13.1 Organizational structure for operations overview

13.1.1 Commitment to the community

The first priority of each member of the Oklo Power staff throughout the life of a unit is improving the lives of the community. Safety is a core aspect of this goal. Decision-making for onsite activities is performed in a conservative manner with expectations of this core value. Chains of command and lines of communication are clearly and unambiguously established to promote effective operations.

13.1.2 Management structure

At all times, Oklo Power Management is responsible for the overall operations of the plant. Oklo Power Management oversees the site and ensures that its priorities are maintained. The Director of Reactor Operations is responsible for overseeing the operations of Oklo Power's nuclear power plants.

Oklo Power Management is supported by the Plant Organization, led by the Plant Manager. The Plant Manager is responsible for overseeing the operations of a specific plant. During normal operation, the site is staffed with two onsite personnel: a Primary Site Monitor and a Secondary Site Monitor, jointly referred to as Onsite Monitors. Oklo Power does not have any licensed operators through 10 CFR Part 55, "Operators' licenses."

The Plant Organization is supported by the Technical Support Organization, which includes Startup Operators, Radiation Protection Personnel, and Technicians. The Technical Support Organization is expected to include additional technical support, including engineering support, that is not described in detail in this section. Unless otherwise specified, personnel in the Technical Support Organization are not expected to be present onsite.

Oklo Power expects to hire several contractors and vendors to perform activities during the life of the plant, primarily during site preparation and maintenance. During off-normal operation, including site preparation, construction, startup, and maintenance, additional personnel may be present onsite. Depending on the situation, onsite personnel may include Site Preparation Personnel, Startup Operators, Radiation Protection Personnel, and Technicians.

13.1.3 Qualification

While reasonable levels of qualification and education are expected for each role, an individual's ability to successfully complete the Training Program and satisfy all job performance requirements are the determining factors for appointment to a position.

13.1.4 Delegation of responsibilities

It is important to note that specific responsibilities may be delegated from one Oklo Power employee to another Oklo Power employee. In these cases, the delegating employee is responsible for properly training the designated employee to complete the task(s) required, and the delegating employee maintains ultimate responsibility for ensuring that the task(s) are completed.

13.1.5 Management

At all times, Oklo Power Management is responsible for the overall operations of the plants and oversees the sites and ensures that the priorities are maintained. The organizational chart is shown in Figure 13-1.

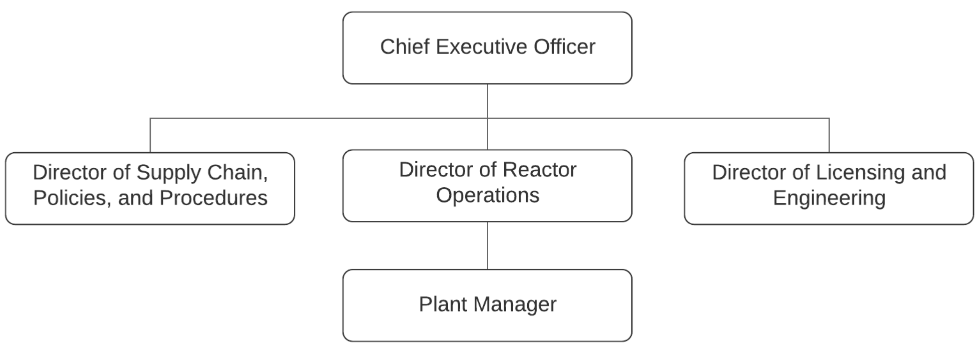

Figure 13-1: Management organizational chart

13.1.5.1 Chief Executive Officer

The Chief Executive Officer (CEO) is responsible for all aspects of operating reactors. The CEO is also responsible to provide technical and administrative support activities to Oklo Power, and responsible for contractors and vendors. The CEO directs Oklo Power Management in the fulfillment of their responsibilities.

13.1.5.2 Director of Reactor Operations

The Director of Reactor Operations is responsible for all matters related to the operations of the power plants, and reports to the CEO. The Director of Reactor Operations is responsible for several operational programs, including the following duties:

- Implement and maintain the Training Program
- Oversee and maintain the Fitness-for-Duty (FFD) Program

- Oversee and maintain the Initial Test Program and approve associated administrative and technical procedures
- Oversee and approve the development and implementation of the Radiation Protection Program, as well as approving exceeding administrative control levels for external dose
- Own the Fire Protection Program
- Oversee and approve the development and implementation of the Physical Security Plan
- Update the Emergency Plan and maintain agreements with the Community Emergency Response Organizations

During emergencies, the Director of Reactor Operations serves as the Headquarters Emergency Coordinator and supports the emergency response, as per the Emergency Plan.

13.1.5.3 Director of Licensing and Engineering

The Director of Licensing and Engineering is responsible for all matters related to regulatory and licensing activities, as well as design and engineering analysis of operating reactors. The Director of Licensing and Engineering reports to the CEO.

13.1.5.4 Director of Supply Chain, Policies, and Procedures

The Director of Supply Chain, Policies, and Procedures is responsible for all matters regarding supply chain and Oklo Power policies and procedures. The Director of Supply Chain, Policies, and Procedures reports to the CEO.

13.1.6 Plant Organization

Oklo Power Management is supported by the Plant Organization, led by the Plant Manager. The Plant Manager is responsible for overseeing the operations of a specific plant. During normal operation, the site is staffed with two onsite personnel: a Primary Site Monitor and a Secondary Site Monitor, jointly referred to as Onsite Monitors. Oklo Power does not have any licensed operators through 10 CFR Part 55, “Operators’ licenses.” The organizational chart is shown in Figure 13-2.

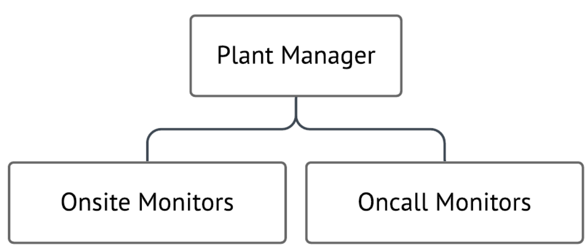

Figure 13-2: Plant Organization organizational chart

13.1.6.1 Plant Manager

The Plant Manager is responsible for the overall operations of the plant. The responsibilities of the Plant Manager include managing site preparation, construction, startup, Initial Test Program, normal operation, and maintenance. The Plant Manager is not expected to be onsite at all times. The Plant Manager reports to the Director of Reactor Operations.

Several additional responsibilities lie with the Plant Manager, including the following:

- Witness and determine Inspection, Testing, Analyses, and Acceptance Criteria (ITAAC)
- Assign shifts to Onsite Monitors and Startup Operators
- Ensure onsite personnel, including Oklo Power employees, contractors, and vendors, have sufficient training, qualifications, and certifications
- Communicate at least monthly with each Onsite Monitor to ensure psychological health
- Perform necessary duties as per the Radiation Protection Program
- Perform necessary duties as per the Emergency Plan
- Perform necessary duties as per the Physical Security Plan
- Perform necessary duties as per the Fire Protection Program

Additional responsibilities during emergencies, including decisions regarding and oversight of recovery and re-entry modes and management of shift changes lie with the Plant Manager and are described in the Emergency Plan.

The Plant Manager is expected to have a bachelor's level degree.

13.1.6.2 Onsite Monitors

During normal operation, the site is staffed with two onsite personnel: a Primary Site Monitor and a Secondary Site Monitor, jointly referred to as Onsite Monitors. The Onsite Monitors report to the Plant Manager.

Onsite Monitors do not have any credited operator actions. The control logic of the Aurora ensures that the reactor trips when necessary. Onsite Monitors have the ability to manually initiate a reactor trip and may be instructed to do so by the Plant Manager.

During normal operation, the responsibilities for Onsite Monitors include the following:

- Monitor key parameters during normal operation
- Ensure the reactor is operating within the technical specifications
- Perform rounds to ensure proper operation of equipment
- Perform necessary duties as per the Radiation Protection Program
- Occupy the Monitoring Room and perform duties as per the Physical Security Plan

During emergencies, the Primary Site Monitor begins to act as the Onsite Emergency Coordinator and the Secondary Site Monitor begins to act as the Onsite Emergency Supporter. The Onsite Monitors carry out the necessary actions as dictated by the Emergency Plan.

Individuals assigned to Onsite Monitor roles are expected to have a high-school diploma or have successfully completed a General Equivalency Development (GED) test.

13.1.6.3 Oncall Monitor

During normal operations, Oncall Monitor(s) are available to support the Plant Organization if determined by the Plant Manager or Onsite Monitors. Oncall Monitors are a subset of the Onsite Monitors that are not on shift at the time but are expected to be fit for duty and able to respond in the case of an emergency, as per the Emergency Plan. Oncall Monitors report to the Plant Manager.

13.1.7 Technical Support Organization

The Plant Organization is supported by the Technical Support Organization, which includes Startup Operators, Radiation Protection Personnel, and Technicians. The Technical Support Organization is expected to include additional technical support, including engineering support, that is not described in this chapter. Unless otherwise specified, personnel in the Technical Support Organization are not expected to be present onsite.

13.1.7.1 Startup Operators

Startup Operators are responsible for the startup of the reactor and for performing startup tests within the Initial Test Program. Startup Operators are expected to be onsite the first time the reactor is started up but will likely not be onsite during normal operations. Startup Operators report to the Director of Reactor Operations.

Responsibilities during initial reactor startup for Startup Operators include the following:

- Perform startup tests within ITP
- Initiate reactor startup
- Perform reactivity changes
- Monitor and control key unit parameters

Startup Operators are expected to have a high-school diploma or have successfully completed a GED test.

13.1.7.2 Radiation Protection Personnel

Radiation Protection Personnel are responsible for overseeing and executing the Radiation Protection Program. Radiation Protection Personnel are led by a Certified Health Physicist (CHP). The leader of the Radiation Protection Personnel reports to the Director of Reactor Operations. Radiation Protection Personnel are not expected to be present onsite during normal operations but are expected to be present during maintenance and decommissioning.

As requested during an emergency under the Emergency Plan, a CHP may be dispatched from Oklo Power headquarters to the site to ensure appropriate protections are in place in the case of a radiological emergency.

Radiation Protection Personnel are expected to have a high-school diploma or have successfully completed a GED test.

13.1.7.3 Instrumentation and Control Technician

Instrumentation and Control (I&C) Technicians are responsible for servicing I&C equipment during site preparation, construction, and maintenance; calibrating I&C equipment; and performing preoperational testing as directed by the Plant Manager. I&C Technicians report to the Plant Manager.

I&C Technicians are expected to have a high-school diploma or have successfully completed a GED test.

13.1.7.4 Mechanical Technician

Mechanical Technicians are responsible for servicing mechanical equipment during site preparation, construction, and maintenance, and performing preoperational testing as directed by the Plant Manager. Mechanical Technicians report to the Plant Manager.

Mechanical Technicians are expected to have a high-school diploma or have successfully completed a GED test.

13.1.7.5 Electrical Technician

Electrical Technicians are responsible for servicing electrical equipment during site preparation, construction, and maintenance, and performing preoperational testing as directed by the Plant Manager. Electrical Technicians report to the Plant Manager.

Electrical Technicians are expected to have a high-school diploma or have successfully completed a GED test.

13.1.8 Contractors and vendors

Oklo Power expects to hire several contractors and vendors to perform activities during the life of the plant, primarily during site preparation, maintenance, and the Initial Test Program. While many of these individuals are not employed full-time by Oklo Power, individuals may be subject to Oklo Power's Fitness-for-Duty Program, described in Chapter 23. Similarly, as contractors and vendors are not subject to the Oklo Power Training Program, described in Chapter 17, onsite personnel may be subject to training to ensure safe practices while onsite. The training and qualification requirements are determined by the Plant Manager prior to access to the plant.

13.1.8.1 Site Preparation Personnel

Site Preparation Personnel are responsible for preparing the site and powerhouse and installation of structures, systems, and components. Site Preparation Personnel report to the Plant Manager.

13.1.8.2 Fitness-for-Duty Personnel

Fitness-for-Duty Personnel are responsible for supporting the implementation of the FFD Program described in Chapter 23; responsibilities may include collecting specimens for drug and alcohol testing, performing drug and alcohol testing, performing behavioral observation, and providing input to a determination of fitness. FFD Personnel report to the Director of Reactor Operations.

II.14 Preoperational testing and initial operations

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14 PREOPERATIONAL TESTING AND INITIAL OPERATIONS

14.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(28) requires the following to be submitted, "Plans for preoperational testing and initial operations." The purpose of this chapter is to describe the Initial Test Program (ITP) for the Aurora and to meet the requirements of 10 CFR 52.79(a)(28).

14.1 Introduction

The Aurora is significantly smaller in size, complexity, and thermal power than existing light water reactors (LWRs). Compared to a typical LWR ITP, the Aurora ITP will require participation of fewer personnel, as described in Section 14.3, and will take less time, as described in Section 14.4. Although the Aurora has significantly reduced complexity, it does include design features that have not been previously reviewed by the U.S. Nuclear Regulatory Commission (NRC) that necessitate first-of-a-kind (FOAK) tests.

14.1.1 Effects of reduced complexity

Most of the Aurora structures, systems, and components (SSCs) are fabricated, assembled, and tested offsite. This ITP focuses on testing that is conducted onsite, after the SSCs are delivered and installed. Because there are few SSCs to test, the Aurora ITP requires the participation of a small group of personnel and can be accomplished in a matter of months. The Aurora ITP is similar to a factory acceptance test for a piece of industrial equipment.

14.1.2 First-of-a-kind testing

First-of-a-kind tests for the Aurora are tests that verify new or unique design features, which have corresponding design bases in Chapter 2, "Description and analysis of structures, systems, and components," and are being reviewed by the NRC for the first time. The new or unique design features in the Aurora include passive decay heat removal and a metal fuel operated in the fast spectrum. The ITP tests that are FOAK tests are indicated along with the test identifier in Table 14-11 and in Section 14.10.4.

Despite differences between the Aurora design and a typical LWR, the objectives, methods, and acceptance criteria of most of the preoperational and startup tests are similar.

14.1.3 Guidance reviewed

The following regulatory guides (RGs) were reviewed and informed the content of this ITP:

- RG 1.20, "Comprehensive vibration assessment program for reactor internals during preoperational and initial startup testing," Revision 4, issued June 2013
- RG 1.68, "Initial test program for water-cooled nuclear power plants," Revision 4, issued June 2013

- RG 1.68.2, “Initial startup test program to demonstrate remote shutdown capability for water cooled nuclear power plants,” Revision 2, issued April 2010
- RG 1.69, “Concrete radiation shields and generic shield testing for nuclear power plants,” Revision 1, issued May 2009
- RG 1.118, “Periodic testing of electric power and protection systems,” Revision 3, issued April 1995
- RG 1.206, “Combined license applications for nuclear power plants,” issued June 2007
- RG 8.38, “Control of access to high and very high radiation areas in nuclear power plants,” Revision 1, issued May 2006
- NUREG-0554, “Single-failure-proof cranes for nuclear power plants,” published May 1979
- NUREG-0612, “Control of heavy loads at nuclear power plants: resolution of generic technical activity A 36,” issued July 1980
- NUREG-1537 “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content,” issued February 1996

14.2 Summary of ITP and objectives

The purpose of this section is to describe the ITP that will be performed during initial startup of the plant. The objective of the ITP is to demonstrate that the plant has been constructed as designed and that the individual systems and the overall plant function as expected based on the plant design.

The major phases of the ITP are the following:

- Preoperational testing
- Startup plan, which includes:
    - Initial fuel loading and pre-criticality testing
    - Initial criticality testing
    - Low-power testing
    - Power-ascension testing

14.2.1 Preoperational testing objectives

Preoperational tests are performed before initial fuel loading. The main objective of these tests is to demonstrate that individual SSCs are installed and function correctly based on the design bases, described in Chapter 2.[23] objectives of the preoperational tests are as follows:

- Document the as-installed location, condition, and configuration of SSCs and demonstrate that the as-installed SSCs meet the expectations of the design
- Demonstrate that engineered protection features and other SSCs function as expected based on the design
- Generate initial test and operating data for components and systems for future reference and to validate analytical models
- Demonstrate readiness for fuel loading and startup testing

Additional preoperational testing[24] will be performed in parallel with the ITP, to verify that all systems, even those that do not have design bases in Chapter 2, are ready for operation. This additional testing is not part of the ITP and is not further described.

Preoperational tests are performed in a specific order to prevent relying on an SSC before that SSC has been tested. After the performance of individual components have been demonstrated, components are integrated, and the performance of the integrated systems are tested. Completion of the preoperational tests and the relevant ITAAC indicates that initial fuel loading can begin, pending the NRC 10 CFR 52.103(g) finding.

14.2.2 Startup plan objectives

The startup plan begins after ITAAC have been successfully completed and the 10 CFR 52.103(g) NRC finding has been made. The objectives of the startup plan are the following:

- Load fuel safely
- Finish reactor assembly
- Perform a series of tests to verify design commitments to ensure the design bases are met

[23] Design bases are the characteristics of a system that ensure the safe operation of the Aurora reactor. Most major systems in the reactor have at least one design basis, but some systems do not have any design bases. Each design basis has one or more design commitments, which are the specific commitments made to ensure that the design basis is met. Each design commitment has one or more preoperational tests or startup tests that are used to verify that the commitment is met. The Physical Security Plan, Emergency Plan, and Radiation Protection Plan do not have specific design bases but have preoperational tests to ensure the associated components are installed and function correctly.

[24] For example, the power conversion system is not required for safe operation of the Aurora and testing of the power conversion system is not included in the ITP.

The design commitments verified during startup testing determine that SSCs are installed and function correctly and determine important operating characteristics of the reactor. After reactor assembly is completed, subcritical tests, initial criticality, and low-power tests are performed. Measurements of important reactor operating characteristics are compared to predicted values to ensure the behavior of the reactor is well understood. The final test is power ascension to full-rated power.

Additional startup testing will be performed in parallel with the ITP, to verify that all systems, even those that do not have design bases in Chapter 2, are ready for operation. This additional testing is not part of the ITP and is not further described.

14.3 Organization and staffing for the ITP

Oklo Power will be both the owner and the operator of its plants. Oklo Power Management is responsible for the plant at all times.[25] Oklo Power Management is supported by the Plant Organization. During the ITP, the Plant Organization is led by the Plant Manager and supported by the Technical Support Organization, other employees, contractors, and vendors. ITP personnel refers to those individuals that are onsite during the ITP; these individuals are not expected to be present onsite during normal operations. The Plant Manager is also ultimately responsible for ensuring ITP personnel are adequately trained.

14.3.1 ITP personnel

14.3.1.1 Plant Manager

The Plant Manager is the leader of the Plant Organization, responsible for the normal operation of a specific plant, and reports to the Director of Reactor Operations. The Plant Manager is responsible for managing the ITP, including the following responsibilities:

- Coordinating between construction activities and the ITP
- Managing, supervising, and scheduling the onsite ITP personnel
- Developing and implementing a detailed ITP schedule
- Implementing and supervising the approved administrative and technical procedures associated with the ITP
- Confirming that personnel and visitors onsite during the ITP have adequate training, qualifications, and certifications
- Witnessing important tests and inspections
- Managing vendor support and contracts associated with the ITP

[25] The organizational structure of Oklo Power is described in Chapter 13, “Organizational structure for operations,” and the relevant portions are summarized in this section.

- Coordinating with utility companies, as a consumer, to establish utility services to the plan
- Coordinating with utility companies as a provider of electricity and process heat
- Coordinating with external organizations to schedule inspections or site visits
- Performing initial review of test results and forwarding results to the correct Oklo Power employee for evaluation and final review
- Providing periodic progress updates to the Director of Reactor Operations that identify overall progress and potential challenges
- Acting as the initial point of contact between ITP personnel and those personnel not already involved in the ITP
- Involving Oklo Power personnel in preoperational and startup activities, when practical, to provide the personnel with relevant experience and knowledge

14.3.1.2 Technical Support Organization

The Technical Support Organization includes Startup Operators, Radiation Protection Personnel, and Technicians. Members of the Technical Support Organization personnel will be onsite during the relevant parts of the ITP.

Startup Operators are responsible for the startup of the reactor and report to the Director of Reactor Operations. Radiation Protection Personnel are responsible for overseeing the Radiation Protection Program[26] and report to the leader of the Radiation Protection Personnel. The leader of the Radiation Protection Personnel reports to the Director of Reactor Operations. Technicians provide electrical, mechanical, instrumentation, and control support during the ITP and report to the Plant Manager.

The Technical Support Organization personnel are responsible for the following duties:

- Implementing technical procedures related to the ITP
- Documenting and reporting the results of procedures
- Complying with administrative procedures

14.3.1.3 Other employees

Other employees involved may include Oklo Inc. or Oklo Power employees such as engineers that are intimately familiar with the Aurora system but are not part of reactor operations group. These employees may provide ITP support from onsite or offsite. The Plant Manager is the initial point of contact to request additional support from other employees not already involved in the ITP.

[26] The Radiation Protection Program is submitted under Part VII, "Enclosures."

The administrative and technical procedures associated with the preoperational and startup phases of the ITP are developed and reviewed by Oklo Power personnel from the licensing and engineering group, the reactor operations group, and the supply chain, policies, and procedures group. Personnel who designed or are responsible for the satisfactory performance of a system or design feature cannot also be solely responsible for formulating or conducting the test activities associated with the system or design feature. The administrative and technical procedures associated with the preoperational and startup phases of the ITP are approved by the Director of Reactor Operations. The Director of Reactor Operations is responsible for overseeing and maintaining the ITP.

14.3.1.4 Contractors and vendors

Oklo Power may hire contractors to perform activities and may hire vendors to provide goods and services during the ITP. The onsite activities of contractors and vendors are scheduled by the Plant Manager. While these individuals are not full-time employees at Oklo Power, these individuals are supervised by the Plant Manager while onsite.

14.3.2 ITP training and qualification

Personnel onsite for the ITP must be qualified and trained to ensure that the personnel are prepared to safely participate in the ITP. ITP personnel includes employees, contractors, and vendors. Visitors also require training before being allowed onsite during the ITP. The Plant Manager is responsible to confirming that ITP personnel, including contractors and vendors, have adequate training, qualifications, and certifications.

14.3.2.1 Training and qualification for employees

The training program for plant personnel is described in Chapter 17, "Training Program description." The Training Program includes qualification verification, medical evaluation, general training, job-specific training, certifications, retraining, and documentation and records requirements. Participation in the Training Program is mandatory for plant personnel.

Employees that are not required to participate in the Training Program may be required to complete part or all of the Training Program before participating in ITP activities onsite. The training and qualification requirements depend on the onsite activities to be performed by the personnel and the expected conditions of the facility while the personnel are onsite. The training and qualification requirements are determined by the Plant Manager.

At the discretion of the Plant Manager, an employee that does not meet the training requirements may be escorted while onsite by an employee that does meet the training and qualification requirements.

14.3.2.2 Training and qualification for contractors and vendors

The training and qualification requirements for contractors and vendors depend on the onsite activities to be performed by the personnel, the type of access needed by the personnel, and the conditions of the facility while the personnel are onsite. The Plant Manager is responsible for coordinating with the contractor or vendor to arrange a work schedule that does not result in additional training or qualification requirements as a result of conditions changing at the facility.

The training and qualification requirements for contractors and vendors is determined by the Plant Manager and should be established before the personnel arrive onsite. Training and qualification activities should be performed before personnel arrive onsite, with the possible exception of site-specific training. Documentation of training and qualification records should be submitted to and approved by the Plant Manager before the personnel arrive onsite.

At the discretion of the Plant Manager, a contractor or vendor that does not meet the training requirements may be escorted while onsite by an employee that does meet the training and qualification requirements.

14.3.2.3 Training and qualification for visitors

Visitors are not expected to participate in onsite ITP activities. The training and qualification requirements for visitors depends on the conditions of the facility while the visitors are onsite. The Plant Manager is responsible for coordinating with the visitor to schedule visits. The training and qualification requirements for visitors is determined by the Plant Manager but will include site-specific training. Documentation of training and qualification records should be submitted to and approved by the Plant Manager before the visitor arrives onsite. Visitor access will be restricted at the discretion of the Plant Manager.

At the discretion of the Plant Manager, a visitor that does not meet the training requirements may be escorted while onsite by an employee that does meet the training and qualification requirements.

14.4 Test procedures

Most SSCs are fabricated, assembled, tested offsite, and delivered onsite. Consequently, the procedures developed for the preoperational and startup tests are similar to a factory acceptance test for a piece of industrial equipment.

Preoperational and startup tests are performed using test procedures. For each test, the test procedure specifies the following items:

- Objectives for performing the test
- Prerequisites that must be completed before starting the test
- Equipment, materials, and personnel required to perform the test
- Special precautions required to protect the safety of personnel or equipment
- Initial conditions under which the test is to be started
- Instructions describing how the test is to be performed, including planned modifications to other SSCs
- Specification of the required data to be collected
- Data analysis methods, if appropriate
- Criteria for evaluating test results

The objectives and acceptance criteria for each test are based on the design commitment that is being verified. Most design commitments are verified by a preoperational test or startup test. Some design commitments are verified by multiple tests to be performed at different stages of the ITP. The Emergency Plan, Physical Security Plan, and Radiation Protection Plan do not have specific design bases but have preoperational tests to ensure the associated components are installed and function correctly.[27] Each test is referenced by a unique test identifier that is cross-referenced in Chapter 2. Test descriptions that include the objective, prerequisites, methods, and acceptance criteria are provided in Section 14.9 and Section 14.10.

14.4.1 Individual test procedure generation

Test procedures are developed, reviewed, and approved by personnel with appropriate technical backgrounds and experience. Test procedures are typically written assuming personnel from the Plant Organization or the Technical Support Organization will be performing the test.

Test procedures are reviewed and revised as needed between personnel in the three Oklo Power groups. A finished test procedure must be approved by an Oklo Power employee from the licensing and engineering group, the reactor operations group, and the supply chain, policies, and procedures group. The approved procedure is then reviewed by the Director of Reactor Operations for final approval.

14.4.2 Conduct of the test program

The Plant Manager is responsible for managing the conduct and schedule of the ITP. All personnel onsite are responsible for understanding and following appropriate safety practices. The Plant Manager organizes the schedule of tests such that the safety of personnel or equipment does not depend on untested components or systems and to minimize plant modifications needed to perform tests. Before the start of each shift, the onsite personnel meet to assess progress from the previous shift, assign goals for the upcoming shift, address any concerns, and handoff information between shifts.

It is the responsibility of the ITP personnel to adhere to approved test procedures and to report deviations to the Plant Manager. Oklo Power employees are responsible for ensuring contractors or vendors adhere to approved test procedures.

14.4.2.1 Safety and authority to stop work

Oklo Power is responsible for providing a safe work environment. Oklo Power provides for a safe work environment through proper education, training, use of protective equipment, and by following safety rules, regulations, standards, and laws. Every person onsite is responsible for understanding and practicing appropriate safety procedures.

At any time during the ITP, all onsite Oklo Power employees, contractors, vendors, and visitors have the right and responsibility to stop work when they encounter an unsafe condition. The affected personnel will stop work, address the situation and, if the affected personnel are in agreement that the unsafe condition has been resolved, work will resume. If the affected employees cannot agree on a resolution to the condition, the Plant Manager has authority to

[27] These plans are submitted under Part VII and are also described in Chapter 9, "Emergency planning," Chapter 18, "Security plans," and Chapter 20, "Radiation Protection Program description."

make a final determination. In the event a person still believes the condition is unsafe, he or she will be assigned to another job with no retribution. No retribution will follow a stop work action initiated in good faith even if the stop work action is deemed unnecessary. Stop work actions should be discussed and documented at the following shift change meeting.

14.4.2.2 Individual tests

The Plant Manager assigns one or more personnel to perform an individual test and provides a copy of the approved test procedure for their review. The personnel responsible for performing the test determine the status of the prerequisites, equipment, and materials that are necessary to perform the test. Before starting the test, any plant modifications necessary to perform the test are discussed with the Plant Manager and may result in additional personnel being assigned to assist with the task. Before approving the start of a test, the Plant Manager determines if the prerequisites, special precautions, and initial conditions are satisfied and that the required equipment, materials, and personnel are available.

After the test is complete, test results and data are provided to the Plant Manager as described in Section 14.4.3. The Plant Manager performs an initial review of the test results and then forwards the data and results to the correct personnel for review, as described in Section 14.4.3.

14.4.2.3 Modifications to approved test procedures

Oklo Power personnel, contractors, and vendors must adhere to approved test procedures. In the event that it is impossible or impractical to adhere to an approved test procedure, the procedure must be modified, and the modification must be approved before continuing the test. Minor modifications to the instructions describing how the test is to be performed can be approved by the Plant Manager. Any approved modification to a test procedure must be reported to the Director of Reactor Operations.

Modifications that are more substantial than minor modifications to test instructions must be reviewed by personnel with appropriate technical backgrounds and experience, and then approved by an Oklo Power employee from the licensing and engineering group and the reactor operations group. The modified procedure is then reviewed by the Director of Reactor Operations for final approval.

14.4.2.4 Modifications and maintenance during the ITP

Temporary plant modifications or maintenance that affect previously tested SSCs may be required during the ITP. The test procedures are designed, and the test schedule is arranged, to minimize temporary plant modifications and maintenance. Modifications and maintenance are performed or supervised by Oklo Power personnel. Testing may be performed and documented to maintain the validity of previously performed tests.

14.4.2.4.1 Planned temporary modifications

Procedures for temporarily modifying SSCs, restoring the SSCs to the original condition, and testing to confirm the operation of the modified SSCs are documented in the test procedure. The scope of post-modification testing for planned temporary plant modifications is determined, reviewed, and documented by personnel with appropriate technical backgrounds and experience in the process of generating and reviewing the test procedure. Planned temporary modifications that are performed according to the test procedure do not invalidate previously performed tests.

14.4.2.4.2 Unplanned temporary modifications

Unplanned temporary modifications to previously tested SSCs may be necessary. Procedures for modifying the components, restoring the components to the original condition, and testing to confirm the operation of the modified components are discussed, reviewed, and documented. The scope of post-modification testing for unplanned modifications is reviewed and approved by one or more Oklo Power personnel with appropriate technical backgrounds and experience. If the modification invalidates a previously completed test, then that test is performed again. Procedures for unplanned temporary modifications that affect previously performed tests can be approved by the Plant Manager or the Director of Reactor Operations.

14.4.2.4.3 Planned and unplanned maintenance

Maintenance activities may affect previously tested SSCs. Planned maintenance is included in the ITP schedule by the Plant Manager. Planned maintenance is performed according to a procedure that documents the necessary testing to confirm the operation of the affected components. The scope of post-maintenance testing is determined, reviewed, and documented in the process of generating and reviewing the maintenance procedure. Planned maintenance that is performed according to the maintenance procedure does not invalidate previously performed tests.

Unscheduled maintenance is added to the ITP schedule, as necessary, by the Plant Manager. Unscheduled maintenance on an SSC that is performed according to a planned maintenance procedure does not invalidate tests previously performed on the affected SSC. Unplanned maintenance activities do not have an existing maintenance procedure. If an unplanned maintenance activity affects a previously tested SSC, the scope of post-maintenance testing is determined, reviewed, and documented by Oklo Power employees with appropriate technical backgrounds and experience. Procedures for unplanned maintenance that affect previously performed tests can be approved by the Plant Manager or the Director of Reactor Operations.

14.4.2.5 Completion of and transition between test program phases

The Plant Manager coordinates between construction and ITP activities. Construction is defined in 10 CFR 51.4, "Definitions," and includes driving of piles, subsurface preparation, placement of backfill, concrete, or permanent retaining walls within an excavation, installation of foundations, or in-place assembly, erection, fabrication, or testing, which are for the following items for the Aurora:

- SSCs whose failure could prevent safety-related SSCs from fulfilling their safety-related function
- SSCs necessary to comply with 10 CFR Part 73, "Physical protection of plants and materials"
- SSCs necessary to comply with 10 CFR Part 50.48, "Fire protection"
- SSCs necessary to comply with 10 CFR Part 50.47, "Emergency plans," and 10 CFR Part 50, Appendix E, "Emergency planning and preparedness for production and utilization facilities"

The Plant Manager reports to the Director of Reactor operations when all construction is complete. Construction should be complete to the degree that outstanding construction items could not be expected to affect the validity of test results of the ITP. The Director of Reactor Operations must approve the start of the ITP to allow the ITP to proceed under the direction of the Plant Manager.

Preoperational tests can be performed prior to the completion of construction with the approval of the Plant Manager and the Director of Reactor Operations. Construction and construction related inspections and tests associated with the SSC being tested must be completed prior to performing preoperational tests on the SSC. Test evaluation reports for preoperational tests are submitted to the Director of Reactor Operations as the evaluation reports are completed. After the completion of all of the preoperational tests, the ITP continues to startup testing after the NRC makes the 10 CFR 52.103(g) finding.

After the NRC 10 CFR 52.103(g) finding, the ITP continues with startup testing, beginning with fuel loading. Test evaluation reports are submitted to the Director of Reactor Operations as the reports are completed. After fuel loading, the ITP proceeds under the direction of the Plant Manager with the exception that the Director of Reactor Operations must approve the start of initial criticality testing and the start of power-ascension testing.

14.4.3 Review and evaluation of test results

After the completion of a preoperational or startup test, the test results and test data are provided to the Plant Manager. The Plant Manager, or delegated employee with appropriate technical background and experience, performs an initial review of the test results. As part of the initial review, a test evaluation report is started. The test evaluation report is used to track the review and evaluation of the test results within the organization. The test evaluation reports include the following items:

- Test procedure
- Identity of the test result evaluators
- Location where test results and test data are stored
- Comparison of applicable test data with the related acceptance criteria
- Justifications for acceptance of SSCs that do not conform with the acceptance criteria, if applicable
- Conclusions about the SSC adequacy or deficiency
- A pass or fail indication for the test

After initiating the test evaluation report and performing the initial review, the Plant Manager forwards the test results, data, and report to the correct personnel, as indicated in the test procedure, for evaluation. For some tests, the Plant Manager is allowed to evaluate the results, determine if the test passed or failed, complete the test evaluation report, and send the test evaluation report to the Director of Reactor Operations. For most tests, one or more Oklo Power personnel are required to evaluate the test results, determine that the test passed or failed, complete the test evaluation report, and send the test report to the Plant Manager and the Director of Reactor Operations.

If a test passes, the ITP proceeds under the direction of the Plant Manager. If a test fails, the Oklo Power personnel that evaluated and reviewed the results discuss the failure with the Plant Manager and Director of Reactor Operations, as appropriate. Oklo Power personnel with appropriate technical backgrounds and experience are assigned to develop a plan to address the test failure. The plan to address the test failure must be approved by the Director of Reactor Operations. After the cause of the test failure has been addressed, the SSC is retested, and the results of the test are reevaluated. ITP tests that list another ITP test as a prerequisite cannot be started until the prerequisite test is evaluated as passing.

14.4.4 Test records

ITP test procedures, test results, test data, and reports are retained as part of the plant's historic record in accordance with the Oklo Quality Assurance Program. Test results, test data, and test evaluation reports are retained regardless of whether the test was evaluated as passing or failing.

After the completion of the preoperational testing, a preoperational test report is generated that is a summary of the individual preoperational test evaluation reports. After the completion of the startup testing, a startup report is generated that is a summary of the individual startup test reports.

14.5 Utilization of reactor operating and testing experiences

The Aurora is a compact fast reactor that produces 4 megawatts thermal and has many design features that make it significantly different than water cooled power reactors. Despite the design differences, previous operating and testing experience from other reactors is relevant to the Aurora ITP. The Aurora design includes design features which have not been previously reviewed by the NRC that necessitate FOAK tests, as described in Section 14.1.2. The FOAK tests will only be performed on the first Aurora and will not be repeated after experience is gained testing and operating the first Aurora. First-plant-only tests are described in Section 14.5.2.

14.5.1 Information reviewed and effect on the ITP

RG 1.68 provides ITP guidance specifically for water-cooled nuclear power plants. The Aurora is not water-cooled but RG 1.68, and the regulatory guides and standards referenced in RG 1.68, were reviewed. Despite the fundamental design differences between the Aurora and an LWR, many of the preoperational tests, startup tests, and guidance were relevant to the Aurora. The guidance documents referenced by RG 1.68 that were relevant to the Aurora are listed in Section 14.1.3. The relevant operating and testing experience from these documents was used to inform preoperational and startup tests to be completed under the scope of the ITP and also the Oklo Power tests that will be completed in parallel with the ITP.

A literature search was preformed to find published operating and testing experience for reactors with similar characteristics to the Aurora. Research and test reactors with similar thermal power, similar fuel type, or other design similarities to the Aurora were identified. EBR-II reports were reviewed to identify tests or inform the ITP [1][2]. Documents describing the assembly and startup of the Massachusetts Institute of Technology Reactor (MITR) were reviewed to identify tests or inform the ITP [47][48]. Experience from the MITR and EBR-II startup tests will also be used in the generation of the Aurora startup test procedures.

14.5.2 First-plant-only tests

The Aurora includes design features which have not been previously reviewed by the NRC and necessitate FOAK tests. FOAK tests are performed during the ITP to verify new or unique design features. Following the successful completion of these tests in the first plant, these FOAK tests are not required for following plants. Table 14-1 provides a list of the FOAK tests with the test identifier and the table in which the test abstract is provided.

Table 14-1: List of first-plant-only tests

Test objective	Test identifier	Reference to abstract
Verify the net power coefficient of reactivity of the reactor core system is negative.	SUT.RXS.03.A (FOAK)	Section 14.10.4
Verify the reactor can be cooled passively by conduction through the surrounding systems to the environment.	SUT.RXS.04.B (FOAK)	Section 14.10.4

14.6 Trial use of plant operating and emergency procedures

To the extent practical, plant operating, emergency, and surveillance test procedures will be developed and reviewed prior to the ITP. During the preoperational portion of the ITP, these procedures will be tested, and corrections and updates will be made to the procedures, as needed. These procedures will be reused for subsequent plant installations. Relevant training will be performed with onsite personnel prior to beginning the startup portion of the ITP.

As the Aurora is expected to operate nearly automatically, many of the operational roles of traditional reactors are unnecessary for the Aurora. During normal operation, the site is staffed with two Onsite Monitors. During normal operations, the Onsite Monitors do not perform any credited operator actions. The plant operating, emergency, and surveillance test procedures will be used to train Onsite Monitors during the startup portion of the ITP, as needed. Onsite Monitors will be included, as appropriate, during portions of the startup activities to provide the personnel with hands-on experience and knowledge.

14.7 Startup plan

The startup plan for the Aurora is described in this section and the startup test abstracts are provided in Section 14.10. Startup test abstracts are only provided for tests that are used to confirm design commitments made in Chapter 2.

The startup tests ensure that the operating characteristics of the reactor are well understood and validate the predicted behavior of the reactor. Measurements of selected parameters are compared to calculated values to verify the design commitments made in Chapter 2. The acceptance criteria for the startup tests ensures that the reactor is functioning within the bounds for which it was designed and analyzed.

The startup plan starts after the finding under 10 CFR 52.103(g) is made. The startup plan follows these general steps:

1. Fuel is loaded.
2. Pre-critical test is performed to verify the as-installed reactor core system.
3. Reactor assembly and verification testing is completed.
4. Pre-critical tests are performed to determine the worth of shutdown rods and control drums.
5. Final reactor assembly and associated verification tests are completed.
6. Initial criticality and low-power testing are performed to determine important reactor parameters for comparison to predicted performance and acceptance criteria.
7. Power-ascension to full power is completed.

A list of the startup tests to be performed during the ITP is provided in Table 14-11.

Pre-critical testing, initial criticality, low-power testing, and power ascension will be conducted on a one-shift basis so that transfer of information between personnel from shift-to-shift is not needed. Engineers intimately familiar with the Aurora will oversee the startup activities.

14.7.1 Fuel loading

Initial fuel loading is conducted cautiously to prevent inadvertent criticality. Predictions of core reactivity are prepared in advance to aid in evaluating the measured responses to specified fuel loading increments. Neutron count-rate and reactor period reactor trips are configured and tested before starting initial fuel loading. Neutron flux is continuously monitored during fuel loading to monitor the subcritical multiplication factor. Each reactor cell is inspected before being loaded into the reactor.

14.7.2 Pre-critical testing and final reactor assembly

Pre-critical tests begin after fuel loading. Pre-critical testing includes verification of core assembly, additional reactor assembly, subcritical multiplication tests, and final core assembly. Test abstracts for these tests are shown in Sections 14.10.1, 14.10.2, and 14.10.3.

Pre-critical tests are conducted cautiously to prevent inadvertent criticality. Neutron flux is continuously monitored to measure the subcritical multiplication factor. Automatic reactor trips are configured to prevent inadvertent criticality.

After fuel loading is complete, a startup test is performed to inspect the assembly of the core to ensure components have been installed correctly. After verification of the core assembly, assembly of the reactor continues with installation of shielding components and reactor enclosure components. Startup tests are completed to verify the installation of each system is correct, as shown in Section 14.10.1. The final reactor assembly is not completed until after subcritical measurements of shutdown rod worth and control drum worth are completed.

Subcritical multiplication measurements are performed to determine important reactor parameters for comparison against predicted performance and acceptance criteria. Reactor parameters that are determined during pre-critical testing include cold-core integral shutdown rod worth and cold-core integral control drum worth, as shown in the abstracts in Section 14.10.2. Determination of these parameters allow an improved calculation of excess core reactivity. If necessary, the excess reactivity is adjusted by adjusting the absorber cells located in the center of the core. If adjustments to the absorber cells are made, the pre-critical testing shown in Section 14.10.2 is repeated. Some of these measurements may be performed again after initial power ascension activities are completed.

Final reactor assembly includes installation of heat exchanger components, shielding components, instrumentation components, and reactor enclosure components. Startup tests are completed to verify the installation of each system is correct, as shown in Section 14.10.3. After final reactor assembly and any associated tests are completed and evaluated as passing, the startup plan continues with initial criticality.

14.7.3 Initial criticality

Initial criticality is approached cautiously to achieve criticality in a safe and controlled manner. Neutron flux is continuously monitored during the approach to criticality. The reactor achieves initial criticality with the absorber portion of the three control drums turned into the core and by slowly withdrawing the three shutdown rods from the core. The initial approach to criticality is performed using the same methods and procedure that will be used for subsequent cold startups, but with additional hold points and slower removal of the shutdown rods. Because there are not any design commitments that are verified during the initial criticality, there are not any startup tests to be completed during the initial criticality. Should finer control motion be desired, the drums can be placed in a partial absorber in position so drum control can be used as well.

The following items are necessary to be completed before starting the approach to initial criticality:

- Critical shutdown rod positions have been predicted
- Control drums are rotated to the minimum reactivity position
- Shutdown rods are fully inserted into the core
- Nuclear instruments are calibrated
- Neutron count-rate and reactor period automatic trip limits are conservatively set
- A neutron count rate of at least 0.5 counts/sec should register on the startup channels before startup begins, and the signal to noise ratio must be known to be greater than 2

14.7.4 Low-power testing

After initial criticality, low-power testing is performed to test additional aspects of the Aurora design that could not be tested previously. Low-power testing abstracts are provided in Section 14.10.4. Low-power test abstracts are only provided for tests that are used to confirm design commitments made in Chapter 2. Low-power testing is conducted cautiously. Neutron flux is continuously monitored, and the neutron count-rate and reactor period automatic trip limits are conservatively set.

Low-power testing is performed for the following reasons:

- Verify the negative reactivity coefficient of the reactor core
- Verify the ability of the reactor to passively conduct heat to the surrounding environment starting from nominal operating temperature
- Measure the reactivity worth of the shutdown rods at operating temperature
- Verify shutdown rod insertion time at elevated core temperature
- Verify adequate overlap of source and intermediate range neutron instrumentation
- Measure neutron flux distribution
- Measure neutron and gamma radiation dose rates in the facility

14.7.5 Power-ascension testing

Power-ascension testing is performed after the results of the low-power testing have been completed and evaluated as passing. The completion of the low-power startup tests completes the programmatic controls used to verify the design commitments described in Chapter 2. Because there are not any design commitments that are verified during power ascension, there are not any startup tests associated with power-ascension testing.

The purpose of power-ascension testing is to achieve full power in a safe and controlled manner, confirm that the plant operates in accordance with the design at normal steady state conditions, and confirm the functionality of the automatic control system. In addition, the dynamic behavior of the plant during and following anticipated transients will be determined, to the extent practical. During power-ascension, power is increased gradually with specific tests being performed at power levels of approximately 10 percent, 25 percent, 50 percent, 75 percent, and 100 percent. Measured values are compared to predicted responses to validate the analytical models.

14.8 Test program schedule

The construction and construction related tests should be completed before starting the ITP. The first phase of the ITP is preoperational testing, including the analysis and review of test results and is expected to take between 1 month and 3 months. Based on this schedule, most ITAAC will be completed within 225 days before the scheduled loading of fuel. Preoperational tests can be performed prior to the completion of construction, in accordance with the ITP. Following the completion of preoperational testing, the second phase of the ITP may start and is the startup program. Startup testing, beginning with loading fuel and including the analysis and review of test results, is expected to take between 1 month and 3 months.

14.9 Preoperational test abstracts

As described in Section 14.2.1, the preoperational tests are based on the design commitments made to ensure the design bases of the Aurora reactor are met. The following tables list the preoperational test objectives, including test identifiers that cross-reference with Chapter 2:

- Table 14-2 – List of building and auxiliary system (BAS) preoperational tests
- Table 14-3 – List of control drum system (CDS) preoperational tests
- Table 14-4 – List of reactor enclosure system (RES) preoperational tests
- Table 14-5 – List of reactor system (RXS) preoperational tests
- Table 14-6 – List of shutdown rod system (SRS) preoperational tests
- Table 14-7 – List of instrumentation and control system (ICS) preoperational tests
- Table 14-8 – List of physical security (PS) preoperational tests
- Table 14-9 – List of radiation protection (RP) preoperational tests
- Table 14-10 – List of emergency plan (EP) preoperational tests

The individual preoperational tests are divided into the following groups that are generally based on common prerequisite dependencies and similar testing activities:

- Section 14.9.1 – Instrumentation and control system installation test group
- Section 14.9.2 – Instrumentation and control system configuration test group
- Section 14.9.3 – Reactor trip system functionality test group (1 of 2)
- Section 14.9.4 – Reactor trip system functionality test group (2 of 2)
- Section 14.9.5 – Fixed reactor component installation test group
- Section 14.9.6 – Pre-fuel loading reactor component installation test group
- Section 14.9.7 – Reactor ICS and SRS installation and functionality test group
- Section 14.9.8 – Control drum system actuator functionality test group
- Section 14.9.9 – Fire detection and suppression test group
- Sections 14.9.10, 14.9.11, and 14.9.12 – Physical security test group
- Section 14.9.13 – Radiation protection test group
- Section 14.9.14 – Emergency plan test group

Test abstracts that describe test frequency, purpose, objectives, methods, and acceptance criteria are included in each group of tests. Test abstracts were generated based on the design commitments, knowledge of the SSCs, and knowledge of assembly requirements. These test abstracts will be used as the basis for generating test procedures.

Individual tests appear in order in each group of tests, but that order does not necessarily determine the order in which the tests need to be performed and does not preclude tests from being performed in parallel. The prerequisites included with each group of tests determine when a test can be performed relative to other tests and facility status.

Table 14-2: List of building and auxiliary system preoperational tests and objectives

Test identifier	Design basis	Objective
POT.BAS.01.A	DB.BAS.01	Verify the construction of the reactor emplacement against design documents referenced by the test procedure.
POT.BAS.02.A	DB.BAS.02	Verify the critical components and cabling, as identified by the test procedure, are installed in the correct locations, according to design documents referenced by the test procedure.
POT.BAS.02.B	DB.BAS.02	Verify that openings and penetrations through fire barriers are protected according to design documents referenced by the test procedure.
POT.BAS.03.A	DB.BAS.03	Verify the functionality of manual fire pull-stations and individual fire detectors.
POT.BAS.03.B	DB.BAS.03	Verify the fire protection system provides for manual fire fighting capabilities in each fire area.

Table 14-3: List of control drum system preoperational tests and objectives

Test identifier	Design basis	Objective
POT.CDS.01.A	DB.CDS.01	Verify the maximum angular rate of rotation of each control drum.
POT.CDS.01.C	DB.CDS.01	Verify that stepper motors are used as the control drum system actuators.

Table 14-4: List of reactor enclosure system preoperational tests and objectives

Test identifier	Design basis	Objective
POT.RES.01.A1	DB.RES.01	Verify the critical components of the as-installed reactor enclosure system, including the module shell and capsule, are installed correctly.
POT.RES.01.A2	DB.RES.01	Verify the critical components of the as-installed reactor enclosure system, including the capsule lid and module lid, are installed correctly.

Table 14-5: List of reactor system preoperational tests and objectives

Test identifier	Design basis	Objective
POT.RXS.05.A	DB.RXS.05	Verify the critical components of the as-installed reflector system are installed correctly.
POT.RXS.06.A	DB.RXS.06	Verify the critical components of the as-installed shielding system are installed correctly.

Table 14-6: List of shutdown rod system preoperational tests and objectives

Test identifier	Design basis	Objective
POT.SRS.02.A	DB.SRS.02	Verify the shutdown rods release and insert within the allowed time.

Table 14-7: List of instrumentation and control system preoperational tests and objectives

Test identifier	Design basis	Objective
POT.ICS.01.A1	DB.ICS.01	Verify each flux detector is installed in the correct location in the reactor core.
POT.ICS.01.A2	DB.ICS.01	Verify each control drum absolute position sensor is installed in the correct location.
POT.ICS.01.B1	DB.ICS.01	Verify each process limit monitor is connected in the correct location in the junction box.
POT.ICS.01.B2	DB.ICS.01	Verify the process limit monitors are configured with the correct scaling information and limit setpoints.
POT.ICS.01.C1	DB.ICS.01	Verify each flux detector is connected in the correct location in the junction box.
POT.ICS.01.C2	DB.ICS.01	Verify each control drum absolute position sensor is connected in the correct location in the junction box.
POT.ICS.01.D	DB.ICS.01	Verify that each reactor trip system process limit monitor sends a fault signal when the measured value exceeds a limit.
POT.ICS.02.A	DB.ICS.02	Verify the time between the exceedance of a limit setpoint and the reactor trip signal is less than the specified time.
POT.ICS.03.A	DB.ICS.03	Verify the functionality of each of the manual reactor trip buttons installed in the facility.
POT.ICS.04.A	DB.ICS.04	Verify that a reactor trip signal causes the reactor trip system to latch in the tripped state and that a deliberate action must be performed to reset the system from the tripped state.
POT.ICS.05.A	DB.ICS.05	Verify the reactor trip system does not use any digital computers or custom software.
POT.ICS.05.B	DB.ICS.05	Verify the reactor trip system is isolated from computer networks.
POT.ICS.05.C	DB.ICS.06	Verify the control cabinets and instrumentation cabinets are installed in an access-controlled area.
POT.ICS.05.D	DB.ICS.05	Verify the process limit monitors are configured to require a password before limit setpoints, scaling information, or other configuration can be changed.
POT.ICS.06.A	DB.ICS.06	Verify that loss of AC power to each control cabinet activates a time-delay relay that results in a reactor trip after the expiration of the time-delay if power is not restored.
POT.ICS.06.B	DB.ICS.06	Verify that loss of DC power to the reactor trip circuit or to the aggregation logic in the control cabinet causes a reactor trip signal.
POT.ICS.06.C	DB.ICS.06	Verify that disconnecting a sensor from a process limit monitor causes the process limit monitor to send a fault signal.
POT.ICS.06.D	DB.ICS.06	Verify the redundant reactor trip system components, including the control and instrumentation cabinets, are installed in the correct locations, according to design documents referenced by the test procedure.

Table 14-8: List of physical security preoperational tests and objectives

Test identifier	Objective
POT.PS.01	Verify that access control points are established to control personnel access into the protected area and detect prohibited items. This capability remains operable from an uninterruptible power supply in the event of the loss of normal power.
POT.PS.02	Verify that an access control system with numbered picture badges is installed for use by individuals who are authorized to access the protected area without escort.
POT.PS.03	Verify that emergency exits through the protected area perimeter are locked, alarmed, and equipped with a crash bar to allow for emergency egress.
POT.PS.04	Verify that penetrations through the protected area barrier are secured and monitored as per the relevant implementing procedure of the Physical Security Plan.
POT.PS.05	Verify the protected area is locked and alarmed with active intrusion detection systems that annunciate in the alarm station upon intrusion into the protected area. This capability remains operable from an uninterruptible power supply in the event of the loss of normal power.
POT.PS.06	Verify that the intrusion detection system is capable of detection and surveillance of unauthorized penetration or activities in the protected area independent of the presence or absence of natural light, and this capability remains operable from an uninterruptible power supply in the event of the loss of normal power.
POT.PS.07	Verify that security alarm devices are tamper indicating and self-checking.
POT.PS.08	Verify that alarm annunciation indicates the type of alarm and location.
POT.PS.09	Verify that equipment exists to record onsite security alarm annunciation, including the location of the alarm, date, time, alarm circuit, type of alarm, false alarm, alarm check, and tamper indication. This capability remains operable from an uninterruptible power supply in the event of the loss of normal power.
POT.PS.10	Verify that alarm station is located inside the protected area and the interior of the alarm station is not visible to persons outside the protected area.
POT.PS.11	Verify that the intrusion detection and assessment system provides visual displays and audible annunciation of alarms to the alarm station, and concurrently transmits visual display and alarm data to Headquarters and the appropriate Community Emergency Response Organization(s). This capability remains operable from an uninterruptible power supply in the event of the loss of normal power.
POT.PS.12	Verify that security alarm annunciation and surveillance video data is displayed in the alarm station and is transmitted to Headquarters and the appropriate Community Emergency Response Organization(s), and that the video image recording with real time playback capability enables assessment of activities before and after each alarm annunciation. This capability remains operable from an uninterruptible power supply in the event of the loss of normal power.

Table 14-9: List of radiation protection preoperational tests and objectives

Test identifier	Objective
POT.RP.01	Verify that all radiation monitoring equipment required by the Radiation Protection Program implementing procedures is functional and installed in the correct locations.
POT.RP.02	Verify that the Monitoring Room displays the readings from the relevant radiation monitoring equipment as required by the Radiation Protection Program implementing procedures.
POT.RP.03	Verify that appropriate signage is posted to indicate radiation areas as required by the Radiation Protection Program implementing procedures.
POT.RP.04	Verify that a secure check source storage cabinet is installed as required by the Radiation Protection Program implementing procedures.

Table 14-10: List of emergency plan preoperational tests and objectives

Test identifier	Objective
POT.EP.01	Verify that sufficient data to identify the current Emergency Action Level, including the PCS pressure, core temperature, core power, and basement radiation levels, are accessible in the Monitoring Room, and are transmitted to Headquarters and to the relevant Community Emergency Response Organization(s). This capability remains operable in the event of the loss of normal power.
POT.EP.02	Verify that the Monitoring Room has redundant communication methods capable of continuous communication with Headquarters, the appropriate Community Emergency Response Organization(s), and the NRC.
POT.EP.03	Verify the Monitoring Room is sufficiently sized for six occupants.
POT.EP.04	Verify that portable radiation detectors are accessible and functional for screening personnel for contamination and for assessing the source of unusual radiation levels.

14.9.1 Instrumentation and control system installation test group

Frequency	These tests are required to be performed once per reactor.
Purpose	Completion of the following tests verifies that the tested components are installed correctly.
Prerequisites	Installation of each component must be completed prior to inspecting or testing the component.
Test identifier	**POT.ICS.06.D**
objective	Verify the redundant reactor trip system components, including the control and instrumentation cabinets, are installed in the correct locations, according to design documents referenced by the test procedure.
method	Visual inspection to identify the component, visual identification of the installed location, and comparison to referenced design documents.
acceptance criteria	Redundant reactor trip system logic is installed in separate fire areas to prevent fire-induced failure of the reactor trip system.
Test identifier	**POT.BAS.02.A**
objective	Verify the critical components and cabling, as identified by the test procedure, are installed in the correct locations, according to design documents referenced by the test procedure.
method	Visual inspection and measurements of the critical components and cabling, and comparison to referenced design documents.
acceptance criteria	Components and cabling that could adversely impact an automatic reactor trip and initiate a loss of heat sink will be separated from each other by fire barriers.
Test identifier	**POT.BAS.02.B**
objective	Verify that openings and penetrations through fire barriers are protected according to design documents referenced by the test procedure.
method	Visual inspection and measurements of the components installed to protect fire barrier openings and penetrations, and comparison to referenced design documents.
acceptance criteria	Openings and penetrations through fire barriers are protected by components (e.g. fire doors, fire dampers, or penetration seals) having fire resistance equivalent to those of the barrier.
Test identifier	**POT.ICS.05.C**
objective	Verify the control cabinets and instrumentation cabinets are installed in an access-controlled area.
method	Confirmation that access-control features are in place to protect the control and instrumentation cabinets from unauthorized access.
acceptance criteria	The process limit monitors are installed in an access-controlled area to prevent changes to limit setpoints, scaling information, or other configuration by unauthorized personnel.

14.9.2 Instrumentation and control system configuration test group

Frequency These tests are required to be performed once per reactor.

Purpose Completion of these tests verifies the tested instrumentation and control system components are installed and/or configured correctly.

Prerequisites 1. POT.ICS.06.D, POT.BAS.02.A, POT.BAS.02.B, and POT.ICS.05.C test results must be evaluated as passing before starting these tests.
2. POT.ICS.05.D test results must be evaluated as passing before performing POT.ICS.05.A, POT.ICS.05.B, and POT.ICS.01.B2.

Test identifier	**POT.ICS.05.D**
objective	Verify the process limit monitors are configured to require a password before limit setpoints, scaling information, or other configuration can be changed.
method	Confirmation by attempted access to configuration menus on each process limit monitor without entering the device password.
acceptance criteria	The process limit monitors are password protected to prevent changes to limit setpoints, scaling information, or other configuration by unauthorized personnel.
Test identifier	**POT.ICS.05.A**
objective	Verify the reactor trip system does not use any digital computers or custom software.
method	Confirmation by visual inspection that the reactor trip system is installed according to design documents referenced by the test procedure. Confirmation that the firmware on each process limit monitor matches the firmware installed by the manufacturer, as recorded during the procurement quality assurance process.
acceptance criteria	The reactor trip system does not use any digital computers or custom software.
Test identifier	**POT.ICS.05.B**
objective	Verify the reactor trip system is isolated from computer networks.
method	Confirmation by visual inspection that the reactor trip system is installed according to design documents referenced by the test procedure.
acceptance criteria	The reactor trip system is isolated from computer networks to prevent changes to limit setpoints, scaling information, or other configuration by unauthorized personnel.
Test identifier	**POT.ICS.01.B2**
objective	Verify the process limit monitors are configured with the correct scaling information and limit setpoints.
method	Confirmation by visual inspection that the scaling information and limit setpoints on each process limit monitor in the reactor trip system match the values in design documents referenced by the test procedure.
acceptance criteria	The reactor trip system process limit monitors are connected to the correct locations, and are configured with the correct sensor scaling information and limit setpoints.

14.9.3 Reactor trip system functionality test group (1 of 2)

Frequency	These tests are required to be performed once per reactor.
Purpose	Completion of these tests verifies the tested reactor trip system components function correctly.
Prerequisites	1. POT.ICS.01.B2 test results must be evaluated as passing before starting these tests. 2. Installation of each component must be completed prior to inspecting or testing the component.
Test identifier	**POT.ICS.03.A**
objective	Verify the functionality of each of the manual reactor trip buttons installed in the facility.
method	Confirmation by two persons that actuation of each manual reactor trip button results in the shutdown rod electromagnets de-energizing.
acceptance criteria	Manual reactor trip buttons send a reactor trip signal when pushed.
Test identifier	**POT.ICS.04.A**
objective	Verify that a reactor trip signal causes the reactor trip system to latch in the tripped state and that a deliberate action must be performed to reset the system from the tripped state.
method	Confirmation by visual inspection that the reactor trip system maintains the tripped state until the reset control is used to reset the system to the not-tripped state.
acceptance criteria	A reactor trip signal causes the reactor trip system to latch in the tripped state. After the condition that caused the reactor trip has been resolved, a control must be toggled to reset the trip system from the tripped state.
Test identifier	**POT.ICS.01.D**
objective	Verify that each reactor trip system process limit monitor sends a fault signal when the measured value exceeds a limit.
method	Use a sensor simulator to source a known value into each of the sensor inputs. Simulate values above and below the limit setpoints, as applicable, and verify the over-limit and/or under-limit fault status was correctly received at each control cabinet.
acceptance criteria	The reactor trip system process limit monitors send a fault signal when a process variable exceeds a limit.
Test identifier	**POT.ICS.02.A**
objective	Verify the time between the exceedance of a limit setpoint and the reactor trip signal is less than the specified time.
method	Simultaneously measure the analog process variable associated with the limit setpoint and the voltage to the shutdown rod electromagnets to determine the time between the process variable exceeding the limit setpoint and the de-energization of the shutdown rod electromagnets. Repeat the test three times.
acceptance criteria	The reactor trip system detects the exceedance of a limit setpoint and sends a reactor trip signal within 6 seconds.

14.9.4 Reactor trip system functionality test group (2 of 2)

Frequency	These tests are required to be performed once per reactor.
Purpose	Completion of these tests verifies the tested reactor trip system components function correctly.
Prerequisites	1. POT.ICS.01.B2 test results must be evaluated as passing before starting these tests. 2. Installation of each component must be completed prior to inspecting or testing the component.
Test identifier	**POT.ICS.06.C**
objective	Verify that disconnecting a sensor from a process limit monitor causes the process limit monitor to send a fault signal.
method	Use a sensor simulator to source a known value inside the process variable operational limits into each of the sensor inputs. Verify the process limit monitor is not sending a fault signal. Disconnect the simulated sensor and verify the process limit monitor sends a fault signal.
acceptance criteria	Detection of a disconnected sensor causes the associated process limit monitor to send a fault signal.
Test identifier	**POT.ICS.06.A**
objective	Verify that loss of AC power to each control cabinet activates a time-delay relay that results in a reactor trip after the expiration of the time-delay if power is not restored.
method	With the reactor trip system in the not-tripped state, disconnect the control cabinet from facility power. Confirm by visual inspection that the reactor trip system tripped after the expiration of the time-delay. Perform this test on each control cabinet individually.
acceptance criteria	Loss of AC power to one or both control cabinets activates a time-delay that results in a reactor trip signal if power is not restored within five minutes.
Test identifier	**POT.ICS.06.B**
objective	Verify that loss of DC power to the reactor trip circuit or to the aggregation logic in the control cabinet causes a reactor trip signal.
method	With the reactor trip system in the not-tripped state, disconnect the DC power to the reactor trip circuit and aggregation logic in one control cabinet, and confirm by visual inspection that the reactor trip system tripped. Perform this test on each control cabinet individually, and with both cabinets simultaneously.
acceptance criteria	Loss of DC power to the reactor trip circuit or the aggregation logic in one or both control cabinets causes a reactor trip signal.

14.9.5 Fixed reactor component installation test group

Frequency	These tests are required to be performed once per reactor.
Purpose	Completion of the following tests verifies that the tested components are installed correctly.
Prerequisites	1. Installation of each component must be completed prior to inspecting or testing the component. 2. POT.BAS.01.A test results must be evaluated as passing before starting POT.RES.01.A1.
Test identifier	**POT.BAS.01.A**
objective	Verify the construction of the reactor emplacement against design documents referenced by the test procedure.
method	The as-built dimensions of the reactor cavity are measured and compared to design documents referenced by the test procedure.
acceptance criteria	The critical components of the reactor module, as identified in the appropriate procedure, are installed in the reactor module emplacement as described in the design documents referenced by the procedure.
Test identifier	**POT.RES.01.A1**
objective	Verify the critical components of the as-installed reactor enclosure system, including the module shell and capsule, are installed correctly.
method	Identification of critical components, measurement of critical dimensions, and comparison to design documents referenced by the procedure.
acceptance criteria	The critical components of the reactor enclosure system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.
Test identifier	**POT.RXS.06.A**
objective	Verify the critical components of the as-installed shielding system are installed correctly.
method	Identification of critical components, measurement of critical dimensions, and comparison to design documents referenced by the procedure.
acceptance criteria	The critical components of the shielding system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.
Test identifier	**POT.RXS.05.A**
objective	Verify the critical components of the as-installed reflector system are installed correctly.
method	Identification of critical components, measurement of critical dimensions, and comparison to design documents referenced by the procedure.
acceptance criteria	The critical components of the reflector system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.

14.9.6 Pre-fuel loading reactor component installation test group

Frequency	These tests are required to be performed once per reactor, except for POT.RES.01.A2. POT.RES.01.A2 is repeated as SUT.RES.01.A during startup testing.
Purpose	Completion of the following tests verifies that the tested components are installed and/or function correctly.
Prerequisites	1. POT.RES.01.A1, POT.RXS.06.A, and POT.RXS.05.A test results must be evaluated as passing before starting these tests. 2. POT.ICS.01.A1 test results must be evaluated as passing before starting POT.RES.01.A2.
Test identifier	**POT.ICS.01.A1**
objective	Verify each flux detector is installed in the correct location in the reactor core.
method	Visual identification of the neutron flux detector prior to installation, visual confirmation that the detector cable is correctly labeled, visual identification of the installation location in the reactor, comparison to design documents referenced by the procedure, and confirmation with another person followed by installation of the flux detector into the identified position in the reactor.
acceptance criteria	The reactor trip system sensors are installed in the correct locations.
Test identifier	**POT.RES.01.A2**
objective	Verify the critical components of the as-installed reactor enclosure system, including the capsule lid and module lid, are installed correctly.
method	Identification of critical components, measurement of critical dimensions, and comparison to design documents referenced by the procedure.
acceptance criteria	The critical components of the reactor enclosure system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.

14.9.7 Reactor ICS and SRS installation and functionality test group

Frequency	These tests are required to be performed once per reactor, except for POT.SRS.02.A, which is repeated as SUT.SRS.02.A during startup testing.
Purpose	Completion of these tests verifies the tested components are installed and/or function correctly.
Prerequisites	1. Installation of each component must be completed prior to inspecting or testing the component. 2. POT.ICS.01.A2 test results must be evaluated as passing before starting POT.ICS.01.C2.
Test identifier	**POT.ICS.01.A2**
objective	Verify each control drum absolute position sensor is installed in the correct location.
method	Visual confirmation that the sensors installed at each location are correct according to design documents referenced by the procedure. Visual confirmation that the cable connected to each sensor is correctly labeled according to design documents referenced by the test procedure.
acceptance criteria	The reactor trip system sensors are installed in the correct locations.
Test identifier	**POT.ICS.01.B1**
objective	Verify each process limit monitor is connected in the correct location in the junction box.
method	Use a sensor simulator to source a known value into each sensor input in the junction box. Use the known simulated value to identify the process limit monitor connected to the sensor input. Verify the correct limit monitor is connected to each input according to design documents referenced by the test procedure.
acceptance criteria	The reactor trip system process limit monitors are connected to the correct locations, and are configured with the correct sensor scaling information and limit setpoints.
Test identifier	**POT.ICS.01.C1**
objective	Verify each flux detector is connected in the correct location in the junction box.
method	Visual inspection of the label on the termination end of the sensor cable and the labeling in the junction box, comparison to design documents referenced by the test procedure, and confirmation by a second person.
acceptance criteria	The reactor trip system sensors are connected to the correct process limit monitors.
Test identifier	**POT.ICS.01.C2**
objective	Verify each control drum absolute position sensor is connected in the correct location in the junction box.
method	Visual inspection of the label on the termination end of the sensor cable and the labeling in the junction box, comparison to design documents referenced by the test procedure, and confirmation by a second person.
acceptance criteria	The reactor trip system sensors are connected to the correct process limit monitors.
Test identifier	**POT.SRS.02.A**
objective	Verify the shutdown rods release and insert within the allowed time.
method	Simultaneously record the reactor trip signal and the position of each shutdown rod. With the shutdown rods fully withdrawn, initiate a reactor trip and measure the time required for each rod to fully insert.
acceptance criteria	The shutdown rod system fully inserts shutdown rods within 4 seconds of receiving a trip signal.

14.9.8 Control drum system actuator functionality test group

Frequency	These tests are required to be performed once per reactor.
Purpose	Completion of these tests verifies that the tested components are installed and/or function correctly.
Prerequisites	1. Installation of each component must be completed prior to inspecting or testing the component. 2. The control drum absolute position instrumentation is calibrated before starting POT.CDS.01.A.
Test identifier	**POT.CDS.01.A**
objective	Verify the maximum angular rate of rotation of each control drum.
method	Command each drum at the maximum control speed to an angular position in the direction of increasing reactivity. Measure the time required for the drum to rotate to the commanded position. Repeat the test using each control cabinet to provide the command signal.
acceptance criteria	The maximum rotation speed of the drums is limited to 1×10-2 deg/sec.
Test identifier	**POT.CDS.01.C**
objective	Verify that stepper motors are used as the control drum system actuators.
method	Visual inspection of the control drum stepper motors.
acceptance criteria	The control drum actuators use stepper motors to eliminate the possibility of a hot-short induced unintentional rotation.

14.9.9 Fire detection and suppression test group

Frequency	These tests are required to be performed once per reactor.
Purpose	Completion of these tests verifies that the tested components are installed and/or function correctly.
Prerequisites	Installation of each component must be completed prior to inspecting or testing the component.
Test identifier	**POT.BAS.03.A**
objective	Verify the functionality of manual fire pull-stations and individual fire detectors.
method	Manually actuate each of the pull-stations and each fire detector and verify the response according to design documents referenced by the test procedure.
acceptance criteria	Manual pull stations or individual fire detectors provide fire detection capability and can be used to initiate fire alarms.
Test identifier	**POT.BAS.03.B**
objective	Verify the fire protection system provides for manual fire fighting capabilities in each fire area.
method	Confirmation by visual inspection that the fire fighting equipment is installed according to design documents referenced by the test procedure.
acceptance criteria	The fire protection system provides for manual fire fighting capabilities in each fire area.

14.9.10 Physical security test group (1 of 3)

Frequency	These tests are required to be performed once per reactor.
Purpose	Completion of these tests verifies that the physical security components are installed and function correctly.
Prerequisites	Installation of each component must be completed prior to inspecting or testing the component.
Test identifier	**POT.PS.01**
objective	Verify that access control points are established to control personnel access into the protected area and detect prohibited items. This capability remains operable from an uninterruptible power supply in the event of the loss of normal power.
method	Test, inspection, or a combination of tests and inspections of the installed systems will be performed.
acceptance criteria	The access control points control personnel access into the protected area, detect prohibited items, and remain operable in the event of loss of normal power.
Test identifier	**POT.PS.02**
objective	Verify that an access control system with numbered picture badges is installed for use by individuals who are authorized to access the protected area without escort.
method	Test, inspection, or a combination of tests and inspections of the installed system will be performed.
acceptance criteria	The access control system includes numbered picture badges and can identify and authorize protected area access only to those personnel with unescorted access authorization.
Test identifier	**POT.PS.03**
objective	Verify that emergency exits through the protected area perimeter are locked, alarmed, and equipped with a crash bar to allow for emergency egress.
method	Test, inspection, or a combination of tests and inspections of the emergency exits through the protected area boundaries will be performed
acceptance criteria	Emergency exits through the protected area perimeter are locked and secured by locking devices that allow prompt egress during an emergency and opening the emergency exit actuates an alarm in the alarm station. This capability remains operable in the event of the loss of normal power.
Test identifier	**POT.PS.04**
objective	Verify that penetrations through the protected area barrier are secured and monitored as per the relevant implementing procedure of the Physical Security Plan.
method	Inspections will be performed of penetrations through the protected area barrier.
acceptance criteria	Penetrations through the protected area barrier are secured and monitored.
Test identifier	**POT.PS.05**
objective	Verify the protected area is locked and alarmed with active intrusion detection systems that annunciate in the alarm station upon intrusion into the protected area. This capability remains operable from an uninterruptible power supply in the event of the loss of normal power.
method	An inspection of the as-built protected area and alarm stations are performed.
acceptance criteria	Protected areas are alarmed with active intrusion detection systems and intrusion is detected and annunciated in the alarm station.

14.9.11 Physical security test group (2 of 3)

Frequency	These tests are required to be performed once per reactor.
Purpose	Completion of these tests verifies that the physical security components are installed and function correctly.
Prerequisites	Installation of each component must be completed prior to inspecting or testing the component.
Test identifier	**POT.PS.06**
objective	Verify that the intrusion detection system is capable of detection and surveillance of unauthorized penetration or activities in the protected area (PA) independent of the presence or absence of natural light, and this capability remains operable from an uninterruptible power supply in the event of the loss of normal power.
method	Testing of detection and surveillance capability in the PA will be performed.
acceptance criteria	The intrusion detection system is capable of detection and surveillance of unauthorized penetration or activities in the PA in the presence or absence of natural light and in the event of loss of normal power.
Test identifier	**POT.PS.07**
objective	Verify that security alarm devices are tamper indicating and self-checking.
method	Confirmation by visual inspection that the security alarm devices and associated transmission lines are installed according to documents referenced by the test procedure.
acceptance criteria	The security alarm devices are tamper indicating and self-checking.
Test identifier	**POT.PS.08**
objective	Verify that alarm annunciation indicates the type of alarm and location.
method	Functional testing of the security alarm devices to verify alarm annunciation indicates the type and location of the alarm.
acceptance criteria	The alarm annunciation indicates the type of alarm and location.
Test identifier	**POT.PS.09**
objective	Verify that equipment exists to record onsite security alarm annunciation, including the location of the alarm, date, time, alarm circuit, type of alarm, false alarm, alarm check, and tamper indication. This capability remains operable from an uninterruptible power supply in the event of the loss of normal power.
method	Test, inspection, or a combination of tests and inspections of the installed system will be performed.
acceptance criteria	Equipment to record onsite security alarm annunciation details is functional and remains so in the event of loss of normal power.
Test identifier	**POT.PS.10**
objective	Verify that alarm station is located inside the protected area and the interior of the alarm station is not visible to persons outside the protected area.
method	Inspection of the alarm station will be performed.
acceptance criteria	The alarm station is located inside the protected area and the interior of the alarm station is not visible from the perimeter of the protected area.

14.9.12 Physical security test group (3 of 3)

Frequency	These tests are required to be performed once per reactor.
Purpose	Completion of these tests verifies that the physical security components are installed and function correctly.
Prerequisites	Installation of each component must be completed prior to inspecting or testing the component.
Test identifier	**POT.PS.11**
objective	Verify that the intrusion detection and assessment system provides visual displays and audible annunciation of alarms to the alarm station, and concurrently transmits visual display and alarm data to Headquarters and the appropriate Community Emergency Response Organization(s) (CEROs). This capability remains operable from an uninterruptible power supply in the event of the loss of normal power.
method	Test, inspection, or a combination of tests and inspections of the installed system will be performed.
acceptance criteria	The intrusion detection and assessment system provides visual displays and audible annunciation to the alarm station and transmits visual and alarm data to Headquarters and the appropriate CERO(s), even in the event of loss of normal power.
Test identifier	**POT.PS.12**
objective	Verify that security alarm annunciation and surveillance video data is displayed in the alarm station and is transmitted to Headquarters and the appropriate Community Emergency Response Organization(s) (CEROs), and that the video image recording with real time playback capability enables assessment of activities before and after each alarm annunciation. This capability remains operable from an uninterruptible power supply in the event of the loss of normal power.
method	Test, inspection, or a combination of test and inspections of the installed systems will be performed.
acceptance criteria	Security alarm annunciation and surveillance video data is displayed in the alarm station and at Headquarters and the appropriate CEROs, and the video image recording with real time playback capability enables assessment of activities before and after each alarm annunciation.

14.9.13 Radiation protection test group

Frequency	These tests are required to be performed once per reactor.
Purpose	Completion of these tests verifies that the radiation protection components are installed and function correctly.
Prerequisites	Installation of each component must be completed prior to inspecting or testing the component.
Test identifier	**POT.RP.01**
objective	Verify that all radiation monitoring equipment required by the Radiation Protection Program implementing procedures is functional and installed in the correct locations.
method	Test, inspection, or a combination of tests and inspections of the installed systems will be performed.
acceptance criteria	Each radiation monitor listed in the appropriate implementing procedure is installed in the correct location and tested to verify functionality.
Test identifier	**POT.RP.02**
objective	Verify that the Monitoring Room displays the readings from the relevant radiation monitoring equipment as required by the Radiation Protection Program implementing procedures.
method	Test, inspection, or a combination of tests and inspections of the installed systems will be performed.
acceptance criteria	The Monitoring Room displays the readings from the radiation monitoring equipment.
Test identifier	**POT.RP.03**
objective	Verify that appropriate signage is posted to indicate radiation areas as required by the Radiation Protection Program implementing procedures.
method	Inspection of the installed signage will be performed.
acceptance criteria	Appropriate radiation area signage is posted.
Test identifier	**POT.RP.04**
objective	Verify that a secure check source storage cabinet is installed as required by the Radiation Protection Program implementing procedures.
method	Inspection of the installed check source storage cabinet will be performed.
acceptance criteria	A secure check source storage cabinet is installed.

14.9.14 Emergency plan test group

Frequency	These tests are required to be performed once per reactor.
Purpose	Completion of these tests verifies that the emergency plan components are installed and function correctly.
Prerequisites	Installation of each component must be completed prior to inspecting or testing the component.
Test identifier	**POT.EP.01**
objective	Verify that sufficient data to identify the current Emergency Action Level (EAL), including the PCS pressure, core temperature, core power, and basement radiation levels, are accessible in the Monitoring Room, and are transmitted to Headquarters and to the relevant Community Emergency Response Organization(s) (CEROs). This capability remains operable in the event of the loss of normal power.
method	Test, inspection, or a combination of tests and inspections of the installed display systems will be performed.
acceptance criteria	Data sufficient to determine the EAL is displayed in the Monitoring Room, and is transmitted to Headquarters and to the relevant CEROs, and this capability remains operable in the event of loss of normal power.
Test identifier	**POT.EP.02**
objective	Verify that the Monitoring Room has redundant communication methods capable of continuous communication with Headquarters, the appropriate Community Emergency Response Organization(s), and the NRC.
method	Test, inspection, or a combination of tests and inspections of the installed communication systems will be performed.
acceptance criteria	The Monitoring Room has redundant communication methods capable of continuous communication with all appropriate offsite parties, and they remain operable in the event of loss of normal power.
Test identifier	**POT.EP.03**
objective	Verify the Monitoring Room is sufficiently sized for six occupants.
method	Inspect the Monitoring Room.
acceptance criteria	The size of the Monitoring Room is as specified in the design of the Aurora powerhouse.
Test identifier	**POT.EP.04**
objective	Verify that portable radiation detectors are accessible and functional for screening personnel for contamination and for assessing the source of unusual radiation levels.
method	Test, inspection, or a combination of tests and inspections of the portable radiation detector systems will be performed.
acceptance criteria	The required portable radiation detectors are accessible and functional.

14.10 Startup test abstracts

As described in Section 14.2.2, the startup tests are based on the design commitments made to ensure the design bases of the Aurora reactor are met. A list of the startup test objectives, including test identifiers and design bases that cross-reference with Chapter 2, is shown in Table 14-11.

The individual startup tests are divided into the following groups generally based on common prerequisite dependencies and similar testing activities:

- Post-fuel loading reactor component installation test group (Section 14.10.1)
- Pre-critical cold-core subcritical multiplication test group (Section 14.10.2)
- Heat exchanger installation and final reactor assembly test group (Section 14.10.3)
- Low-power test group (Section 14.10.4)

Test abstracts that describe test frequency, purpose, objectives, methods, and acceptance criteria are included in each group of tests. Test abstracts were generated based on the design commitments, knowledge of the SSCs, and knowledge of assembly requirements. These test abstracts will be used as the basis for generating test procedures.

Individual tests appear in order in each group of tests, but that order does not necessarily determine the order in which the tests need to be performed. The prerequisites included with each group of tests determine when a test can be performed relative to other tests and facility status.

Table 14-11: List of startup tests and objectives

Test identifier	Design basis	Objective
SUT.CDS.01.B	DB.CDS.01	Determine the integral control drum worth of each control drum with a cold core.
SUT.HXS.01.A	DB.HXS.01	Verify the critical components of the as-installed heat exchanger system are installed correctly.
SUT.ICS.01.A	DB.ICS.01	Verify each thermocouple is installed in the correct location.
SUT.ICS.01.C	DB.ICS.01	Verify each thermocouple is terminated in the correct location in the junction box.
SUT.RES.01.A	DB.RES.01	Verify the critical components of the as-installed reactor enclosure system, including the capsule lid and module lid, are installed correctly.
SUT.RXS.03.A (FOAK)	DB.RXS.03	Verify the net power coefficient of reactivity of the reactor core system is negative.
SUT.RXS.04.A	DB.RXS.04	Verify the critical components of the as-installed reactor core system, including the fuel, are installed correctly.
SUT.RXS.04.B (FOAK)	DB.RXS.04	Verify the reactor can be cooled passively by conduction through the surrounding systems to the environment.
SUT.RXS.06.A1	DB.RXS.06	Verify the critical components of the as-installed shielding system, including the heat exchanger shield, are installed correctly.
SUT.RXS.06.A2	DB.RXS.06	Verify the critical components of the as-installed shielding system, including the top shield, are installed correctly.
SUT.SRS.01.A1	DB.SRS.01	Determine the integral shutdown rod worth of each shutdown rod with a cold core.
SUT.SRS.01.A2	DB.SRS.01	Determine total shutdown rod worth of individual shutdown rods with the core at operating temperature.
SUT.SRS.02.A	DB.SRS.02	Verify the shutdown rods release and insert into the core within the allowed time with the core at operational temperature.

14.10.1 Post-fuel loading reactor component installation test group

Frequency	These tests are required to be performed once per reactor.
Purpose	Completion of these tests verifies that the tested components are installed and/or function correctly.
Prerequisites	1. SUT.RXS.04.A may be performed in parallel with fuel loading. 2. SUT.RXS.04.A test results must be evaluated as passing before starting SUT.RXS.06.A1.
Test identifier	**SUT.RXS.04.A**
objective	Verify the critical components of the as-installed reactor core system, including the fuel, are installed correctly.
method	Identification of critical components, measurement of critical dimensions, and comparison to design documents referenced by the procedure.
acceptance criteria	The critical components of the reactor core system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.
Test identifier	**SUT.RXS.06.A1**
objective	Verify the critical components of the as-installed shielding system, including the heat exchanger shield, are installed correctly.
method	Identification of critical components, measurement of critical dimensions, and comparison to design documents referenced by the procedure.
acceptance criteria	The critical components of the shielding system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.

14.10.2 Pre-critical cold-core subcritical multiplication test group

Frequency	These tests are required to be performed once per reactor. These tests must be repeated after adjustments are made to reconfigurable reflector cells, shutdown rods are replaced, fixed reflectors are replaced, or reactor cells are replaced.
Purpose	Completion of these tests provide measurements of the reactivity worth of the control drums and the shutdown rods with a cold core.
Prerequisites	1. Neutron flux instrumentation, neutron flux limits, and reactor period limits are functional and can automatically trip a reactor shutdown. 2. The reactor core is loaded with fuel and the layout of the reactor core has been verified. 3. Area radiation monitors are operational. 4. The background neutron source term has been quantified. 5. The neutron source is installed in the reactor.
Test identifier	**SUT.CDS.01.B**
objective	Determine the integral control drum worth of each control drum with a cold core.
method	Subcritical multiplication test with each control drum rotated at angular increments from the minimum reactivity position to the maximum reactivity position.
acceptance criteria	The total reactivity worth of the drums is less than 700 pcm at all operating conditions.
Test identifier	**SUT.SRS.01.A1**
objective	Determine the integral shutdown rod worth of each shutdown rod with a cold core.
method	Subcritical multiplication test with each rod incrementally removed from the core.
acceptance criteria	The worth of each shutdown rod will be greater than 1400 pcm, where 1400 pcm is greater than the total of: the reactivity worth associated with the temperature decrease from hot full power conditions to cold zero power conditions, and an additional margin of 500 pcm.

14.10.3 Heat exchanger installation and final reactor assembly test group

Frequency	These tests are required to be performed once per reactor.
Purpose	Completion of these tests verifies that the tested components are installed and/or function correctly.
Prerequisites	1. SUT.ICS.01.A may be completed before installing the heat exchangers in the core. 2. Installation of each component must be completed prior to inspecting or testing the component.
Test identifier	**SUT.ICS.01.A**
objective	Verify each thermocouple is installed in the correct location.
method	Visual inspection of the label on junction end of the thermocouple, visual identification of the installed location, and comparison to design documents referenced by the test procedure.
acceptance criteria	The reactor trip system sensors are installed in the correct locations.
Test identifier	**SUT.HXS.01.A**
objective	Verify the critical components of the as-installed heat exchanger system are installed correctly.
method	Identification of critical components, measurement of critical dimensions, and comparison to design documents referenced by the test procedure.
acceptance criteria	The critical components of the heat exchanger system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.
Test identifier	**SUT.RXS.06.A2**
objective	Verify the critical components of the as-installed shielding system, including the top shield, are installed correctly.
method	Identification of critical components, measurement of critical dimensions, and comparison to design documents referenced by the test procedure.
acceptance criteria	The critical components of the shielding system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.
Test identifier	**SUT.RES.01.A**
objective	Verify the critical components of the as-installed reactor enclosure system, including the capsule lid and module lid, are installed correctly.
method	Identification of critical components, measurement of critical dimensions, and comparison to design documents referenced by the procedure.
acceptance criteria	The critical components of the reactor enclosure system, as identified in the appropriate procedure, are installed as described in the design documents referenced by the procedure.
Test identifier	**SUT.ICS.01.C**
objective	Verify each thermocouple is terminated in the correct location in the junction box.
method	Visual inspection of the label on the termination end of the sensor cable, visual inspection of the labeling in the junction box, comparison to approved design documents, and confirmation with another person.
acceptance criteria	The reactor trip system sensors are connected to the correct process limit monitors.

14.10.4 Low-power test group

Frequency	The tests identified as FOAK are performed once for the Aurora design. SUT.SRS.01.A2 is required to be performed once per reactor and must be repeated after adjustments are made to reconfigurable reflector cells, shutdown rods are replaced, fixed reflectors are replaced, or reactor cells are replaced. SUT.SRS.02.A is required to be performed once per reactor and must be repeated after a shutdown rod or a shutdown rod sleeve is replaced.
Purpose	These tests measure the reactivity worth of the shutdown rods with the core at operating temperature. In addition, the following tests verify the negative temperature coefficient of reactivity, the ability of the reactor to passively dissipate decay heat, and the shutdown rod drop time.
Prerequisites	Before SUT.SRS.01.A2 is performed, temporary modifications must be made to the instrumentation and control system to allow one shutdown rod to be inserted independently from the other shutdown rods.
Test identifier	**SUT.RXS.03.A (FOAK)**
objective	Verify the net power coefficient of reactivity of the reactor core system is negative.
method	The reactor is operated at a low power with minimal cooling from the power conversion system. Nuclear heating increases the temperature of the core and causes the reactor to go sub-critical.
acceptance criteria	The net power coefficient of reactivity of the reactor core system is negative.
Test identifier	**SUT.RXS.04.B (FOAK)**
objective	Verify the reactor can be cooled passively by conduction through the surrounding systems to the environment.
method	The reactor is operated at low power with minimal cooling from the power conversion system to achieve nominal operating temperature. The power conversion system is disabled, the reactor is shut down, and the thermal performance of the reactor is monitored.
acceptance criteria	The reactor core system can be cooled by conduction through the surrounding systems (reflector system, shielding system, heat exchanger system, and reactor enclosure system) and subsequent convection from the module shell after shutdown.
Test identifier	**SUT.SRS.02.A**
objective	Verify the shutdown rods release and insert into the core within the allowed time with the core at operational temperature.
method	Simultaneously record the reactor trip signal and the position of each shutdown rod. Initiate a reactor trip with the shutdown rods fully withdrawn and measure the time required for each rod to fully insert into the core.
acceptance criteria	The shutdown rod system fully inserts shutdown rods within 4 seconds of receiving a trip signal.
Test identifier	**SUT.SRS.01.A2**
objective	Determine total shutdown rod worth of individual shutdown rods with the core at operating temperature.
method	Step insertion of each shutdown rod starting with a critical core at nominal operating temperature.
acceptance criteria	The worth of each shutdown rod will be greater than 1400 pcm, where 1400 pcm is greater than the total of: the reactivity worth associated with the temperature decrease from hot full power conditions to cold zero power conditions, and an additional margin of 500 pcm.

II.15 Operational plans

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15 OPERATIONAL PLANS

15.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(29) requires that applicants provide the following:

> (i) Plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components;
>
> (ii) Plans for coping with emergencies, other than the plans required by § 52.79(a)(21)

The purpose of this section is to provide an overview of the operational programs applicable to the Aurora reactor and serve as a summary of the locations of the programs in this application and their implementation milestones. The operational plans that do not have a regulatory requirement for implementation will be implemented as per the proposed license conditions in Part VI, "Proposed license conditions."

15.1 Evaluation

Table 15-1 lists the operational programs that are applicable to the Aurora design.

Table 15-1: Operational programs

Operational program	Program source (required by)	FSAR chapter	Implementation: Milestone	Implementation: Requirement
Fire Protection Program	10 CFR 50.48	21	Prior to receipt of fuel onsite	License condition
Radiation Protection Program	10 CFR 20.1101	20	Prior to initial receipt of byproduct or source for portions that relate to handling of byproduct or source; Prior to receipt of fuel onsite for full program implementation	License condition
Training Program	10 CFR 50.120	17	At least 3 months prior to the first test being conducted for the Initial Test Program	License condition
Emergency Plan	10 CFR 50.47; 10 CFR Part 50, Appendix E	9	Prior to receipt of fuel onsite	License condition
Physical Security Plan	10 CFR 73.55	18	Prior to receipt of fuel onsite	License condition
Quality assurance program - Operation	10 CFR Part 50, Appendix B	12	30 days prior to the scheduled date for the initial loading of fuel	10 CFR 50.54(a)(1)
Initial Test Program	10 CFR 52.79(a)(28)	14		
Preoperational test program		14	Prior to the first preoperational test for the Initial Test Program	License condition
Startup test program		14	Prior to the first startup plan test for the Initial Test Program	License condition
Fitness-for-Duty Program	10 CFR 52.79(a)44	23	At least 3 months prior to the start construction activities, as defined in 10 CFR 51.4	License condition

II.16 Technical qualifications of the applicant

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16 TECHNICAL QUALIFICATIONS OF THE APPLICANT

16.0 Purpose

Title 10 of the *Code of Federal Regulations* Section 52.79(a)(32) requires that the technical qualifications of the applicant be provided.

Oklo Inc. (Oklo) is the designer of the Aurora plant. Oklo Power LLC (Oklo Power) is a subsidiary of Oklo and will be the applicant, owner and operator of the plant applied for in this application. Oklo and Oklo Power utilize subcontractors for site preparation activities, as well as engineering consultant firms to support the design and application efforts. These partners are listed and the partnerships are described in 16.1.3.

16.1 Evaluation

16.1.1 Oklo Inc.

Oklo Inc. (Oklo) is a privately-funded fission technology developer. Oklo was founded in 2013 and a majority of its employees have technical degrees.

Oklo qualifications include its sources of funding, testimonies, industry leadership roles, partnerships, awards, and ability to prepare and submit a combined operating license application.

Oklo is funded entirely by private investors. In 2014 and 2015, the company completed fundraising rounds. {

}The chair of the Oklo board of directors is Sam Altman. Sam Altman has experience which includes serving as the CEO of OpenAI, and president of Y Combinator, leading it to a combined portfolio value of approximately $100 billion. Oklo's other institutional investors have over $3 billion in assets under management, and Oklo has raised funds from high net worth individuals, including founders and executives of leading technology companies, such as LinkedIn and Facebook.

Oklo's CEO, Jacob DeWitte, has testified before the House of Representatives' Committee on Science, Space, and Technology (2017), the Senate's Committee on Energy and Natural Resources (2016), and spoke at the White House Summit on Nuclear Energy (2015). He serves as the Chair of the Fast Reactor Working Group, the largest and most diverse advanced reactor technology group, and on the Nuclear Energy Institute Board of Directors. Oklo's COO, Caroline Cochran, served on the Department of Energy's Nuclear Energy Advisory Committee from 2018-2019, and serves on the Karnfull Advisory Board, a Swedish company providing 100% carbon-free nuclear power options to consumers. Oklo serves on the University of Michigan Department of Nuclear Engineering Advisory Board, and on the Technology Inclusive Contents of Applications Project (TICAP), among other committee support and involvement .

Oklo has won the following awards: the top MIT team at the MIT Clean Energy Prize (2013), the winner of the energy track at the MIT 100k (2013), finalist at MassChallenge (2013), and winner of the MassChallenge Gold Award (2013). Oklo was also invested in by the selective investor and accelerator, Y Combinator (2014). Oklo was awarded the Trailblazer Award by the Nuclear Industry Council in 2020.

Oklo has had three Gateway for Accelerated Innovation in Nuclear (GAIN) vouchers awarded to the national labs for Oklo projects, totaling over $1.6 million in value to the national labs, including $300,000 in cost share from Oklo to these projects. The vouchers provide access to technical expertise at Department of Energy laboratories including Argonne National Laboratory, Idaho National Laboratory, and Sandia National Laboratory. These awards demonstrate a recognition of Oklo's qualifications and serve as a means of leveraging the technical qualifications of the laboratory system.

In 2016, Oklo began paid pre-application work with the NRC and was assigned a project manager. There have been an array of activities, reports, planning, and meetings with the assigned NRC staff core team in this time.

In 2017, Oklo signed a Memorandum of Understanding (MOU) with the U.S. Department of Energy to support the demonstration of Oklo's advanced fission technology in the early 2020s. Key highlights of the MOU include DOE making fuel resources available for an Oklo reactor, DOE providing site access and data for potential locations for an Oklo reactor, and Oklo licensing and demonstrating its first reactor.

Oklo was selected for a Site Use Permit in 2019 through the process established by INL in 2018. Oklo was also awarded fuel material by INL through a competitive process in 2019.

16.1.2 Oklo Power LLC

Oklo Power is the applicant for this license and the operating subsidiary of Oklo for operations of the Aurora plant being applied for. Oklo Power is wholly owned by Oklo. More information regarding the financial and business relationship between the entities is discussed in Part I, "Financial Qualifications."

16.1.3 Other contractors and participants

Under the direction of Oklo, a number of highly qualified organizations have provided design and analysis in support of the Aurora reactor. Each organization has specific responsibilities to Oklo as defined in various contracts and agreements. The major contributors are identified in this section.

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II.17 Training Program description

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17 TRAINING PROGRAM DESCRIPTION

17.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(33) requires that a description of the training program required by 10 CFR 50.120, “Training and qualification of nuclear plant personnel,” and its implementation be provided.

The purpose of this chapter is to describe the Training Program, as required by 10 CFR 50.120. The Training Program will be implemented in accordance with Chapter 15, “Operational plans,” and Part VI, “Proposed license conditions.” Oklo Power LLC (Oklo Power) is seeking an exemption from the requirement in 10 CFR 50.120(b)(1) to establish, implement, and maintain its Training Program at least 18 months prior to initial fuel load. The relevant exemption is included in Part V, “Non-applicabilities and requested exemptions.”

17.1 Training Program description

Although the function of nuclear plant personnel is very different in the Aurora plant than any prior nuclear plant, and most of this function would not be considered under the regulatory purview, an overview of the training program is described here. A systematic approach to the training process is used to establish and maintain the Training Program.

As the Aurora is expected to operate nearly automatically, many of the operational roles of traditional reactors are unnecessary. It is important to note that onsite personnel do not perform any credited operator actions.

Because there is minimal operating experience for commercial reactors with similar operating structures, the Training Program is informed by ANSI/ANS-15.4-2016, “Selection and Training of Personnel for Research Reactors.” For some roles, the Training Program is informed by NEI 06-13A, Revision 2, “Template for an Industry Training Program Description.”

As described in Chapter 13, “Organizational structure for operations,” Oklo Power is not licensing any reactor operators through 10 CFR Part 55, “Operators’ License.” Individuals who are properly qualified and trained for duty will receive certification through Oklo Power.

In accordance with 10 CFR 50.120, the Training Program includes the following:

- Description of qualification, which ensures that personnel have the appropriate background to qualify for training
- Description of medical evaluation, which ensures that personnel have adequate physical and mental health to perform the required duties
- Description of general training, which ensures that personnel have general training on the site, operations, safety, and Oklo Power policies and procedures
- Description of specific training, which ensures that personnel have the job-specific training to perform the required duties

- Description of certification, which ensures that personnel have demonstrated satisfactory levels of qualification, medical status, and comprehension of training materials
- Description of retraining, which ensures that personnel have received training within an appropriate timeframe
- Methods to ensure the Training Program is properly reviewed and maintained
- Methods to ensure documentation is retained for certified personnel

II.18 Security plans

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18 SECURITY PLANS

18.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(35) requires the following:

> (i) A physical security plan, describing how the applicant will meet the requirements of 10 CFR part 73 (and 10 CFR part 11, if applicable, including the identification and description of jobs as required by § 11.11(a) of this chapter, at the proposed facility). The plan must list tests, inspections, audits, and other means to be used to demonstrate compliance with the requirements of 10 CFR parts 11 and 73, if applicable;
>
> (ii) A description of the implementation of the physical security plan;

Additionally, 10 CFR 52.79(a)(36) requires the following:

> (i) A safeguards contingency plan in accordance with the criteria set forth in Appendix C to 10 CFR part 73. The safeguards contingency plan shall include plans for dealing with threats, thefts, and radiological sabotage, as defined in part 73 of this chapter, relating to the special nuclear material and nuclear facilities licensed under this chapter and in the applicant's possession and control. Each application for this type of license shall include the information contained in the applicant's safeguards contingency plan. (Implementing procedures required for this plan need not be submitted for approval.)
>
> (ii) A training and qualification plan in accordance with the criteria set forth in Appendix B to 10 CFR part 73.
>
> (iii) A cyber security plan in accordance with the criteria set forth in § 73.54 of this chapter;
>
> (iv) A description of the implementation of the safeguards contingency plan, training and qualification plan, and cyber security plan; and
>
> (v) Each applicant who prepares a physical security plan, a safeguards contingency plan, a training and qualification plan, or a cyber security plan, shall protect the plans and other related Safeguards Information against unauthorized disclosure in accordance with the requirements of § 73.21 of this chapter.

The objective of the Physical Security Plan (PSP) is to establish the programmatic elements necessary to maintain physical security throughout the life of the Aurora. The level of security required through the PSP is commensurate with the size of the Aurora and the correspondingly small potential impact to public health. The PSP is submitted as a separate document in Part VII, “Enclosures.”

Contingency measures are executed through the PSP implementing procedures and training and qualification of personnel with security roles is executed through the Training Program. The Training Program is submitted as a separate document in Part VII.

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18.1 Background

Due to the unique market demands for a relatively small power source, the Aurora reactor has similar power level and quantities of nuclear material to a nonpower reactor. The amount of nuclear material in the Aurora reactor is at least an order of magnitude less than a large light water reactor (LWR) as shown in Table 18-1.

Table 18-1: Comparison of current large light water reactor to Aurora

	Current large LWRs	Aurora
Power output (MWth)	1600-4400	<5
Refueling cycle (years)	1.5-2	None
Radionuclide inventory (metric tons)	100-150	<5
System pressure (atm)	150	Near atmospheric
Hydrogen explosion risk	Yes	No
Cooling	Loop with low thermal inertia	Completely passive thermal superconductors
Electric power dependence	Relies on offsite power or emergency diesel generation	No safety-related electric power dependence
Negative reactivity coefficient	Yes	Yes

The information presented in the PSP is largely dependent on the design and largely independent of site-specific characteristics.

This PSP provides information regarding the Aurora reactor by the Materials and Fuels Complex (MFC) site at Idaho National Laboratory (INL), referred to as the Aurora INL site.

18.2 Category of material used

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18.3 Target set identification

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18.4 Physical Security Plan description

In accordance with the relevant portions of 10 CFR 73.55, the PSP includes the following:

- Description of the overall protective strategy for the facility
- Identification of the various positions and their authorities within the organization that are responsible for the PSP
- Description of the detection and assessment systems used
- Description of access controls present on the site and the access control systems used
- Description of the access authorization process for unescorted and escorted access
- Description of the response to a security threat and the relationship to emergency preparedness
- Description of the administrative controls for the maintenance of the PSP

II.19 Incorporation of operational insights

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19 INCORPORATION OF OPERATIONAL INSIGHTS

19.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(37) requires that the information necessary to demonstrate how operating experience insights have been incorporated into the plant design be provided.

The purpose of this section is to explain how the requirements of 10 CFR 52.79(a)(37) are met.

19.1 Discussion

Throughout its design process, Oklo Inc. (Oklo) has considered the operating experience of past reactors and incorporated the successful aspects of those reactors. Consideration of operating experience insights includes searching over multiple reactor designs, including:

- Generic operating experience that affects all nuclear reactors
- Metal fueled fast reactor operating experience
- Compact reactor operating experience and analytical methods
- Light water reactor operating experience.

For example, Oklo's decision to use metal fuel was driven largely by the successful operating experience of Experimental Breeder Reactor II (EBR-II), including its demonstration of inherent safety and the ability to shut down the reactor passively. The operating experience of EBR-II and other metal fueled fast reactors provided an opportunity to iterate on metal fuel design, informing material composition, geometry, and operating limits [25]. The improvements that developed from this experience resulted in substantial gains in performance and reliability over time. The fuel design directly incorporates the insights gained from this experience.

More broadly, past operating experience is incorporated throughout the Aurora, which was designed with simplicity, inherent safety, and ease of operation in mind. As a result, the Aurora utilizes a vastly different and simpler control scheme than past reactors.

II.20 Radiation Protection Program description

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20 RADIATION PROTECTION PROGRAM

20.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(39) requires, "A description of the radiation protection program required by § 20.1101 of this chapter and its implementation," passing the requirements to 10 CFR 20.1101, "Radiation protection programs," which states:

(a) Each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of this part. (See § 20.2102 for recordkeeping requirements relating to these programs.)

(b) The licensee shall use, to the extent practical, procedures, and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

(c) The licensee shall periodically (at least annually) review the radiation protection program content and implementation.

(d) To implement the ALARA requirements of § 20.1101 (b), and notwithstanding the requirements in § 20.1301 of this part, a constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, shall be established by licensees other than those subject to § 50.34a, such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from these emissions. If a licensee subject to this requirement exceeds this dose constraint, the licensee shall report the exceedance as provided in § 20.2203 and promptly take appropriate corrective action to ensure against recurrence.

The implementation of the Radiation Protection Program is done by acknowledging and incorporating provisions in 10 CFR Part 20, "Standards for protection against radiation," into the development of procedures and engineering controls that ensure occupational doses and doses to members of the public are ALARA and within the exposure limits set in 10 CFR Part 20.

The following guidance was reviewed in the development of the Radiation Protection Program:

- Regulatory Guide 1.206, "Combined license applications for nuclear power plants," issued in June 2007
- NEI 07-03A, Revision 0, "Generic FSAR [final safety analysis report] template guidance for radiation protection program description," issued in May 2009
- NEI 07-08A, Revision 0, "Generic FSAR template guidance for ensuring that occupational radiation exposures are as low as is reasonably achievable (ALARA), issued in October 2009

The purpose of this chapter is to describe the Radiation Protection Program, as required by 10 CFR 20.1101.

20.1 Radiation Protection Program description

In accordance with 10 CFR 20.1101, the Radiation Protection Program includes the following:

- Descriptions of possible sources of radiation exposure
- Commitment from management to maintain occupational and public doses ALARA
- Description of practices, procedures, and operational methodology to accomplish ALARA goals
- The organization and responsibilities of various personnel and their training requirements
- Description of the work control program authorizing work in radiation areas and the associated radiological controls
- Description of occupational dose monitoring including personnel dosimetry
- Description of public dose monitoring including environmental dosimeters
- Explanation of expected actions for license termination
- Explanation of design features, equipment, and procedures used to minimize contamination
- Description of fixed, portable, and other instrumentation used for surveying and monitoring for radiation in the facility and associated documentation
- Explanation of access controls used for restricted areas
- Description of the implementation of respiratory protection and controls
- Description of receipt, labeling, and storage of radioactive material
- Compliance to record and report keeping as outlined in 10 CFR Part 20

II.21 Fire Protection Program description

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21 FIRE PROTECTION PROGRAM DESCRIPTION

21.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(40) requires, "A description of the fire protection program required by § 50.48 of this chapter and its implementation," passing the requirement to 10 CFR 50.48, "Fire protection." Specifically, 10 CFR 50.48(a)(1)-(3) apply and requires the following:

> (1) Each holder of an operating license issued under this part or a combined license issued under part 52 of this chapter must have a fire protection plan that satisfies Criterion 3 of Appendix A to this part. This fire protection plan must:
>
> > (i) Describe the overall fire protection program for the facility;
> >
> > (ii) Identify the various positions within the licensee's organization that are responsible for the program;
> >
> > (iii) State the authorities that are delegated to each of these positions to implement those responsibilities; and
> >
> > (iv) Outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage.
>
> (2) The plan must also describe specific features necessary to implement the program described in paragraph (a)(1) of this section such as—
>
> > (i) Administrative controls and personnel requirements for fire prevention and manual fire suppression activities;
> >
> > (ii) Automatic and manually operated fire detection and suppression systems; and
> >
> > (iii) The means to limit fire damage to structures, systems, or components important to safety so that the capability to shut down the plant safely is ensured.
>
> (3) The licensee shall retain the fire protection plan and each change to the plan as a record until the Commission terminates the reactor license. The licensee shall retain each superseded revision of the procedures for 3 years from the date it was superseded.

The remainder of 10 CFR 50.48 is not discussed in this application because it does not apply. Section 50.48(a)(4) of 10 CFR does not apply to combined license applications. Paragraph b of 10 CFR 50.48 does not apply to new plants and therefore this application. Paragraph c of 10 CFR 50.48 is not utilized in this application. Paragraphs d and e of 10 CFR 50.48 are reserved. Paragraph e of 10 CFR 50.48 does not apply to this application.

Section 50.48 of 10 CFR refers to Criterion 3, "Fire protection," of Appendix A, "General design criteria for nuclear power plants," of 10 CFR Part 50. Appendix A to 10 CFR Part 50 does not apply to the Aurora, because it does not apply to nonlight water reactors. Instead, the parallel advanced reactor design criteria 3, “Fire protection,” from Regulatory Guide 1.232, “Guidance for developing principal design criteria for non-light-water-reactors,” Revision 0, issued April 2018, is used and is replicated as follows:

> Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and fire- resistant materials shall be used wherever practical throughout the unit, particularly in locations with structures, systems, or components important to safety. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

The purpose of this section is to describe the Aurora Fire Protection Program (FPP), as required by 10 CFR 50.48.

21.1 Fire Protection Program description

In accordance with 10 CFR 50.48, the Aurora FPP includes the following:

- Description the overall FPP for the facility.
- Identification of the various positions and their authorities within Oklo Power LLC (Oklo Power)’s organization that are responsible for the program.
- Description of the administrative controls and personnel requirements for fire protection and manual fire suppression activities.
- Description of the automatic and manually operated fire detection and suppression systems.
- Description of the means to limit fire damage to structures, systems, and components to ensure the ability to achieve a safe plant state.

The FPP is documented and maintained in an overall Fire Hazard Analysis, which contains the following:

- Evaluation of the potential in-situ and transient fire hazards.
- Determination of the effects of a fire in any location in the plant including the impact on the ability to achieve a safe state and minimizing the risk of radioactive release to the environment.
- Determination of the appropriate measures for fire prevention, fire detection, fire suppression, and fire containment for each area containing structures, systems, and components necessary for achieving a safe state.

II.22 Criticality accidents

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22 CRITICALITY ACCIDENTS

22.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(43) requires information be provided demonstrating compliance with the requirements for criticality accidents as prescribed in 10 CFR 50.68, "Criticality accident requirements." The requirements within 10 CFR 50.68 were established for light water reactors (LWRs). In contrast to LWRs, the Aurora reactor is very small, requiring roughly about 5% of the total fuel of an LWR, and has few auxiliary systems to maintain criticality over plant life. These design features are fundamental to the minimization of concerns related to criticality accidents.

The objective of this section is to meet the requirements of 10 CFR 50.68, specifically of 10 CFR 50.68(b)[28]. This section provides a description of how nuclear fuel is handled for the Aurora plant, to ensure that criticality accidents are prevented, as well as a relevant description of the facility. This section also provides results of bounding analyses that show that criticality accidents are not credible at the Aurora facility.

22.1 Handling of special nuclear material

Fuel handled during initial loading at the beginning of core life is expected to be used for the duration of the plant operating period, removing the need for refueling events. By removing the need for refueling, the probability of reaching criticality outside normal operating conditions is greatly minimized.

Moving and handling of fuel during core life is not anticipated and is not considered normal operations. In the event that fuel would need to be moved during core life, the reactor cells would be handled using a fuel handling tool and only a single reactor cell would be handled at a time. The removed reactor cell would be immediately placed in a storage cask and promptly removed from the site. Storage of fuel onsite, outside of the reactor, is not anticipated during plant life.

Lastly, fuel is handled during decommissioning, when all reactor cells are individually removed from the core and placed in storage casks. These casks would subsequently be transported offsite.

[28] Compliance with 10 CFR 70.24, "Criticality accident requirements," is not sought in lieu of 10 CFR 50.68.

22.2 Facility description

The entire core inventory can be stored in storage casks adjacent to each other in the basement of the powerhouse while reasonably maintaining criticality below requirements of paragraph b to 10 CFR 50.68. The quantity of nuclear fuel onsite cannot reach criticality outside of the reactor module. Therefore, the storage casks do not require fuel loading patterns or zones to ensure safe storage. In addition, the design of the storage casks include neutron absorbing materials, further impeding the probability of reaching inadvertent criticality.

22.3 Criticality analyses

The Aurora reactor is designed and constructed such that its most reactive state is the fully assembled core configuration at the beginning of core life. The reactor module is designed to incorporate reflectors to maximize and optimize fission for the fuel configuration over the core lifetime. Any disruption or change to this configuration results in a less reactive state.

The criticality analyses presented below assume that the reactor cells are placed in the original core configuration outside of the reactor module in a room with concrete floors, ceilings, and walls, modeled after the basement of the Aurora powerhouse. These analyses were done using Serpent Monte Carlo simulations that were run at room temperature using ENDF-VIII.0 cross section libraries.

The two most conservative cases are presented below. Both cases assume all of the reactor cells are adjacent to each other, are confined in a single room, are outside of their storage casks, do not have absorbers, and do not have reflectors. In other words, all of the reactor cells are analyzed in their normal in-reactor configuration, minus any other material.

22.3.1 Case 1

For the first case, the reactor cells were modeled in the same configuration as they are loaded in the core, in the center of an empty room, without any other components (e.g., reactor enclosures, shutdown rods, absorbers, reflectors). { }The simulation geometry is depicted in Figure 22-1 and Figure 22-2. The room was assumed to be filled with dry air. The purpose of modeling this configuration was to analyze the possibility of criticality without any intervening materials or procedures that diminish reactivity levels.

This simulation yielded multiplication factors well below 0.95, maintaining subcriticality as required by 10 CFR 50.68. Results for this case are shown in Table 22-1.

Table 22-1: Case 1, multiplication factors

			95/95 confidence interval	
	Multiplication factor	Relative error	Min	Max
Analog k-eff	7.92629E-01	0.00031	7.92147E-01	7.9311E-01
Implicit k-eff	7.92610E-01	0.00021	7.92284E-01	7.9294E-01
Collision k-eff	7.92851E-01	0.00020	7.92540E-01	7.9316E-01
Absorption k-eff	7.92610E-01	0.00021	7.92284E-01	7.9294E-01

{

}

Figure 22-1: Case 1, X-Y geometry for Aurora criticality accident analysis

{

}

Figure 22-2: Case 1, axial geometry for Aurora criticality accident analysis

22.3.2 Case 2

The second case that was analyzed, like case 1, has all the reactor cells arranged in their original operating configuration, but placed in a smaller room. { } The reactor cells were placed in the center of a room filled with dry air, without casks, shutdown rods, absorbers, or reflectors. The simulation geometry is depicted in Figure 22-3 and Figure 22-4. Like case 1, this simulation was done to show that even with the maximum amount of fuel from the core, stored in a single room in a tight configuration, subcriticality is still maintained.

This simulation also yielded multiplication factors below 0.95, as shown in Table 22-2. The multiplication factors are closer to criticality than in case 1 due to the increased reflection from the concrete resulting from the close proximity of the walls to the fuel. However, even with the increase in neutron reflection, this case still shows that tightly stored reactor cells without any criticality mitigating features, do not reach criticality and maintain the limits provided in the regulations.

Table 22-2: Case 2, multiplication factors

			95/95 confidence interval	
	Multiplication factor	Relative error	Min	Max
Analog k-eff	8.97983E-01	0.00029	8.97473E-01	8.9849E-01
Implicit k-eff	8.97803E-01	0.00020	8.97451E-01	8.9815E-01
Collision k-eff	8.97774E-01	0.00022	8.97387E-01	8.9816E-01
Absorption k-eff	8.97803E-01	0.00020	8.97451E-01	8.9815E-01

{

}

Figure 22-3: Case 2, X-Y geometry for Aurora criticality accident analysis

{

}

Figure 22-4: Case 2, axial geometry for Aurora criticality accident analysis

22.4 Conclusion

These analyses used extreme scenarios that would bound all fuel storage configurations and fuel handling situations. Storing all the reactor cells in a room, in a tight configuration, without reactivity limiting materials or procedures, resulted in multiplication factors that are well below the regulatory limits set in 10 CFR 50.68. The reason this is possible in the Aurora reactor is because criticality is achieved by secondary design features that optimize and preserve neutron population, not simply due to enrichment levels and fuel quantities alone.

No reactor cells are planned to be stored outside of the reactor at any time during the plant lifetime. The analyses provided in this section used extreme assumptions to show that criticality accidents are prevented by the nature of the design of the Aurora facility not to reflect what would be standard procedure at the facility.

II.23 Fitness-for-Duty Program description

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23 FITNESS-FOR-DUTY PROGRAM DESCRIPTION

23.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(44) requires that a description of the fitness-for-duty program required by 10 CFR Part 26, "Fitness for duty programs," and its implementation be provided.

The purpose of this section is to describe the Oklo Power LLC (Oklo Power) Fitness-for-Duty (FFD) Program. The Oklo Power FFD Program will be implemented in accordance with Chapter 15, "Operational plans," and Part VI, "Proposed license conditions." The relevant requested exemptions related to FFD are included in Part V, "Non-applicabilities and requested exemptions."

23.1 Fitness-for-Duty Program description

The Oklo Power FFD Program applies to onsite personnel who are responsible for activities that directly impact the safety or security of the plant. The FFD Program will establish, implement, and maintain written policies and procedures to ensure that the overall goals of the FFD Program are clearly communicated to all employees who are subject to the FFD Program. Consistent with 10 CFR 26.23, "Performance objectives," the overall goals of the FFD Program are the following:

1. Provide reasonable assurance that individuals are trustworthy and reliable as demonstrated by the avoidance of substance abuse.
2. Provide reasonable assurance that individuals are not under the influence of any substance, legal or illegal, or mentally or physically impaired from any cause, which in any way adversely affects their ability to safely and competently perform their duties.
3. Provide reasonable measures for the early detection of individuals who are not fit to perform the duties that require them to be subject to the FFD Program.
4. Provide reasonable assurance that the Aurora site is free from the presence and effects of illegal drugs and alcohol.
5. Provide reasonable assurance that the effects of fatigue and degraded alertness on individuals' abilities to safely and competently perform their duties are managed commensurate with maintaining public health and safety.

The FFD Program includes the following:

- Description of the FFD Program policies and procedures
- Description of the activities that are applicable to the FFD Program during each expected phase of operation
- Description of drug and alcohol testing procedures and cutoff levels
- Description of behavioral observation

- Methods to ensure the FFD Program is properly reviewed and maintained
- Methods to ensure documentation and records are retained

II.24 Probabilistic risk assessment summary

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24 PROBABILISTIC RISK ASSESSMENT SUMMARY

24.0 Purpose

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.79(a)(46) requires "A description of the plant-specific probabilistic risk assessment (PRA) and its results." The purpose of this chapter is to provide a description of a PRA for the Aurora and its results.

The PRA summarized in this chapter considers internal initiating events over all operating conditions. External events are excluded from the PRA because they are treated in a deterministic manner, as described in Chapter 1, "Site envelope and boundary." Hazards associated with accidents resulting from purposeful human-induced security threats (e.g., sabotage, terrorism) and risks associated with accidental radiological exposures to onsite personnel are explicitly excluded from the PRA.

24.1 Definitions

challenge condition: A phenomenon that could potentially pose damage to the Aurora facility.

event family: A method of grouping events according to a common challenge condition.

event tree: A graphical representation of the possible sequence of events that might occur following an initiating event.

initiating event: A failure that causes a challenge condition and starts the chain of events represented in an event tree.

shutdown: A full insertion of one or more of the three shutdown rods. The reactor is subcritical and only decay heat generation occurs following a shutdown of the Aurora.

shutdown failure frequency (SFF): The estimated frequency of occurrence (per reactor-year) of a failure to achieve shutdown.

shutdown PRA: The mature PRA developed for the Aurora that incorporates a metric of shutdown failure frequency and focuses on characterizing the potential challenges to successful shutdown.

24.2 Introduction

Probabilistic risk assessment has been used as a tool throughout the development of the Aurora to ensure the creation of a safe and robust system. Initially, PRA techniques were used to help identify the range of possible off-nominal initiating events that historically have challenged nuclear reactors. With this range of historic events in mind, and with the goal of providing small-scale, safe, and reliable power to potential customers, the Aurora design was developed.

The design characteristics of the Aurora feature passive system operation and inherent means, including:

1. The Aurora is approximately 1000 times smaller than currently operating nuclear power plants, with a substantially smaller overall radionuclide inventory, thereby placing an upper bound on the potential dose consequences of an off-nominal event.

2. The Aurora never reaches burnups higher than 1 atom percent, minimizing material degradation challenges and retaining most fission products in the fuel matrix.

3. The Aurora uses metal fuel, which has inherent safety characteristics such as high thermal conductivity, negative reactivity feedback with thermal expansion, and has a robust response to temperature changes, without concerns of cracking or pulverizing. These inherent safety characteristics limit undesired power increases and any associated fuel damage.

4. The Aurora operates in a fast neutron spectrum, which results in a linear and very small reactivity letdown with burnup, which in turn allows control mechanism reactivity worths to be correspondingly small.

5. The Aurora is cooled by heat pipes, which operate passively and do not rely on electricity; therefore, the Aurora is not challenged by a station blackout, and loss-of-coolant accidents are not possible.

6. The Aurora reactor operates at near-atmospheric pressure, eliminating challenges posed by having high pressures in the reactor.

7. The inherent thermal characteristics of the Aurora, including its small power level, the excess thermal mass and heat capacity in the materials in and surrounding the core, and the presence of passive heat removal via natural convection from the outer surface of the reactor module allow for easy decay heat removal post-shutdown.

8. The Aurora has redundant means of achieving shutdown, and the primary means of shutdown is through a fail-safe, gravity driven shutdown rod system whose rods are positioned external to the core during normal operation. This eliminates challenges associated with rod ejections, rod insertion motor malfunctions, or reactivity insertions due to rod oscillations.

9. The Aurora instrumentation and controls system is designed to be both redundant and fail safe, increasing the probability of a successful rod insertion.

Early PRA of the Aurora included identification of the range of possible off-nominal initiating events that might challenge the reactor, and estimated system responses to these events. Adjustments to the design were made in response to the identified events, to eliminate or reduce the severity of the challenge they posed. Later, dynamic PRA was performed to evaluate the key passive and inherent features of the Aurora design.

Given the inherent thermal characteristics of the Aurora, once the reactor is shut down by the insertion of at least a single shutdown rod, no further challenge to safety exists. As a result, the mature PRA developed for the Aurora and summarized in this chapter focused on identifying, characterizing, and quantifying challenges to successful insertion of a single shutdown rod. As such, this PRA is referred to as a "shutdown PRA." This terminology echoes the historic labeling of PRAs according to the metric of interest.[29]

For the Aurora shutdown PRA, the metric of interest is the SFF, because the Aurora has been designed to reduce or eliminate the probability and consequences of radioactive material release. Using dose as the metric of interest for the Aurora would result in a large set of event sequences with no dose outcome, and a small set of events where small releases occur at exceptionally low frequencies.[30] As a result, the SFF is a more useful metric and is used for the shutdown PRA.

24.3 Shutdown PRA methodology

The shutdown PRA considers all challenge conditions initiated through the failure of a structure, system, or component (SSC) due to internal causes. By modeling the event sequences that could occur due to the failure of an internal SSC, the shutdown PRA provides insight into which SSCs are most important for ensuring that the shutdown rods are inserted into the core, and that no radiological release occurs.

The methodology used to ensure all relevant initiating events are included in the PRA is as follows:

6. Identify all potential internal hazards historically considered for nuclear reactors, as well as those unique to the Aurora design, over all operating states.
7. Screen internal events that are not applicable to the Aurora out of the PRA.
8. Group the internal events into event families based on a common challenge condition.
9. Determine the failure modes for each relevant SSC in the Aurora.
10. Define a set of quantitative criteria for the frequencies and consequences associated with the end states of the internal event sequences.
11. Quantify the event sequences using event trees with conservative event frequencies that take uncertainties into consideration.
12. Evaluate whether any end state meets the pre-determined screening criteria.

---

[29] For example, Level 1 PRAs developed for light water reactors focus on describing events that contribute to the core damage frequency (CDF) of those systems.

[30] Recent efforts to provide guidance for non-LWR advanced reactor PRAs have focused on dose as the metric of interest (which has historically been labeled a Level 3 PRA). However, for the design reasons mentioned above, this metric is not useful for the Aurora.

24.4 Shutdown PRA results

The shutdown PRA identified three event families, each based on a shared challenge condition, related to the operation of the Aurora. The identified event families for the shutdown PRA include the following:

1. Internal events that challenge core cooling
2. Internal events that challenge the heat sink
3. Internal events that are driven by a positive reactivity insertion

The events identified as part of these event families cover the full range of plant operating states and internal failures that create a challenge condition for the reactor. Examples of initiating events that fall into each of these event categories are briefly described in the following paragraphs.

Internal events that challenge core cooling revolve around failures of the heat pipes to transfer heat from the fuel in the Aurora core to the heat exchanger system. Due to the many-fold redundancy of heat pipes in the Aurora core, events in this family are localized events.

Internal events that challenge the heat sink are those that impact the ability of the Aurora secondary system to remove heat from the heat pipes via the heat exchanger system. Example events include a trip of the secondary system pump, a controller failure on the ultimate heat sink heat exchanger, or a break in the secondary system piping.

Internal events that are driven by a positive reactivity insertion almost entirely consist of malfunctions associated with the rotation of the control drums as they compensate for reactivity letdown with burnup. If the control drums rotate too quickly relative to fuel depletion, a positive reactivity insertion results. This positive reactivity insertion may not always pose a challenge condition to the reactor (e.g., if the excess drum rotation rate is very small). The only other source of a positive reactivity insertion challenge condition is the removal of the shutdown rods in too large of a step during reactor startup.

Given that the shutdown PRA revolves around evaluating whether a single shutdown rod inserts successfully, the event trees that result for each of the initiating events over all event families have a consistent form: an initiating event occurs, and either at least one shutdown rod successfully fully inserts, or no rods successfully fully insert. The frequency of the initiating event for the event tree is unique to each initiating event, as is the conditional probability that a single rod successfully fully inserts. As such, while a large set of event sequences were evaluated as part of the shutdown PRA, the event trees associated with these event sequences can be represented by the structure shown in Figure 24-1.

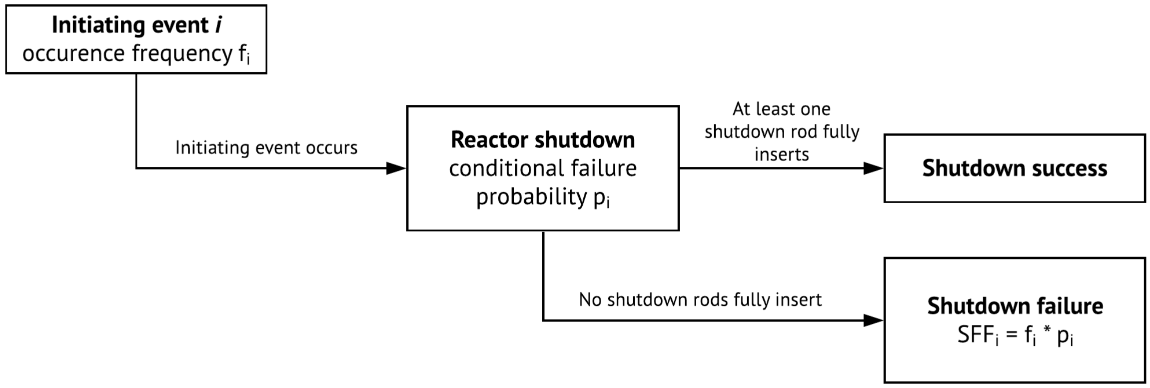

Figure 24-1: Basic event tree structure for the shutdown PRA

Table 24-1 shows the cumulative shutdown failure frequency of each event family. The highest cumulative shutdown failure frequency for any event family is 1.74×10^{-11}, for the "internal events that challenge core cooling" event family. Since the other two event families have estimated shutdown failure frequencies that are orders of magnitude lower, the cumulative shutdown failure frequency over all event families is still 1.74×10^{-11}. The quantitative results of the shutdown PRA demonstrate the robustness of the Aurora design.

The failure of a single shutdown rod insertion was found to have a probability on the order of 10^{-6}, and the probability of failure of two shutdown rods has a probability on the order of 10^{-12}. Probabilities less likely than 10^{-7} are not historically considered (per NUREG 1860), however, since the probability for one shutdown rod insertion failure is within that probability, this insight was used to add the failure of one shutdown rod to the safety analysis as described in Section 5.6.

Table 24-1: SFF per reactor-year for the identified event families

Event family	SFF (1/reactor-year)
Internal events that challenge core cooling	1.74E-11
Internal events that challenge the heat sink	2.02E-18
Internal events driven by a positive reactivity insertion	2.36E-13

24.5 Conclusion

As the Aurora is not challenged by decay heat generation and removal following shutdown, the mature PRA developed for the Aurora seeks to evaluate the challenges that exist to achieving successful shutdown, using the metric of SFF to quantify risk. The shutdown PRA found that the total shutdown failure frequency summed over all event families is 1.74×10^{-11}. As a result, no credible events were identified that resulted in a radiological release, supporting the deterministic safety analysis presented in Chapter 5, "Transient analysis."

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