UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 7, 2019

Mr. Scott Sharp
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: MODIFYING THE DESIGN BASIS FOR QUALITY CLASSIFICATION OF CERTAIN FUEL HANDLING EQUIPMENT (EPID L-2018-LLA-0261)

Dear Mr. Sharp:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 229 to Renewed Facility Operating License No. DPR-42 and Amendment No. 217 to Renewed Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2 (PINGP), respectively. The amendments consist of changes to the updated safety analysis report in response to your application dated October 2, 2018, as supplemented by letter dated December 4, 2018.

The amendments revise the licensing basis regarding the safety classification of certain fuel handling equipment (FHE) to allow identified FHE classified as safety-related to be reclassified as augmented quality or non-safety related, depending upon applicable requirements. The amendment supports the safety reclassification by modifying the definition of a substantial amount of radioactivity as it applies to safety classification of the FHE from 1% of 10 CFR Part 100 limits to 10% of 10 CFR 100 limits. The amount of radioactivity associated with the 10 CFR Part 100 limits is that fission product release that results in the more limiting of:

- a dose to an individual at the PINGP exclusion area boundary for two hours immediately following the onset of the release of 25 rem whole body or 300 rem to the thyroid from iodine exposure; or

- a dose to an individual at the PINGP low population zone boundary during the entire period of the radioactive cloud passage of 25 rem whole body or 300 rem to the thyroid from iodine exposure.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Robert F. Kuntz, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 229 to DPR-42
2. Amendment No. 217 to DPR-60
3. Safety Evaluation

cc w/encls: ListServ

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY - MINNESOTA

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 229
License No. DPR-42

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), dated October 2, 2018, as supplemented by letter dated December 4, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Updated Safety Analysis Report.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA Scott P. Wall for/

Nancy L. Salgado, Branch Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License

Date of Issuance: November 7, 2019.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY - MINNESOTA

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 217
License No. DPR-60

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), dated October 2, 2018, as supplemented by letter dated December 4, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Updated Safety Analysis Report.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA Scott P. Wall for/

Nancy L. Salgado, Branch Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License

Date of Issuance: November 7, 2019.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 229 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 217 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY - MINNESOTA

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated October 2, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML18275A370), as supplemented by letter dated December 4, 2018 ADAMS Accession No. ML18338A431), Northern States Power Company (the licensee) submitted a license amendment request (LAR) for the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2.

The amendments revise the licensing basis regarding the safety classification of certain fuel handling equipment (FHE) to allow identified FHE classified as safety-related to be reclassified as augmented quality or non-safety related, depending upon applicable requirements. The amendment supports the safety reclassification by modifying the definition of a substantial amount of radioactivity as it applies to safety classification of the FHE from 1% of 10 CFR Part 100 limits to 10% of 10 CFR 100 limits. The amount of radioactivity associated with the 10 CFR Part 100 limits is that fission product release that results in the more limiting of:

- a dose to an individual at the PINGP exclusion area boundary for two hours immediately following the onset of the release of 25 rem whole body or 300 rem to the thyroid from iodine exposure; or

- a dose to an individual at the PINGP low population zone boundary during the entire period of the radioactive cloud passage of 25 rem whole body or 300 rem to the thyroid from iodine exposure.

2.0 REGULATORY EVALUATION

2.1 System Description

2.1.1 Gaseous Radioactive Waste System

Section 10.2.1.2.2, “Major Equipment Required for Fuel Handling,” of the PINGP updated safety analysis report (USAR, ADAMS Accession No. ML18155A445) contains a description of the major fuel handling equipment used for refueling. The equipment includes the spent fuel pool (SFP) bridge crane, manipulator crane, the fuel transfer system, the rod cluster control (RCC) changing fixture, and spent fuel assembly handling tools. This equipment handles single fuel assemblies, and its failure could initiate a fuel handling accident involving damage to the fuel cladding and a release of radioactive material. Enclosure 1 to the LAR, Section 2.4, “Description of Equipment,” provided the following descriptions of this equipment:

> Spent Fuel Pool Bridge Crane
>
> The spent fuel pool bridge crane is a wheel-mounted walkway, spanning the spent fuel pool which carries electric monorail hoists on an overhead structure. The fuel assemblies are moved within the spent fuel pool by means of a long[-]handled tool suspended from the hoist. The hoist travel and tool length are designed to limit the maximum lift of a fuel assembly to a safe shielding depth. The West Hoist of the Spent Fuel Pool Bridge Crane has been upgraded to single failure proof.
>
> Manipulator Crane and Load Cell
>
> The manipulator crane is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water. The bridge spans the refueling cavity and runs on rails set into the floor along the edge of the refueling cavity. The bridge and trolley motions are used to position the vertical mast over a fuel assembly in the core. A long tube with a pneumatic gripper on the end is lowered out of the mast to grip the fuel assembly. The gripper tube is long enough so the upper end is still contained in the mast when the gripper end contacts the fuel. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. The fuel is transported while inside the mast tube to its new position. The manipulator can lift only one fuel assembly at a time.
>
> Fuel Transfer System
>
> The fuel transfer system is an underwater conveyor car that runs on tracks extending from the refueling canal through the transfer tube and into the fuel transfer canal. The conveyor car receives a fuel assembly in the vertical position from the manipulator crane. The fuel assembly is lowered to a horizontal position for passage through the tube and then is raised to a vertical position in the fuel transfer canal.
>
> During plant operation, the conveyor car is normally stored in the fuel transfer canal. A blind flange is bolted on the refueling canal end of transfer tube to seal

the reactor containment. The terminus of the tube outside the containment is closed by a gate valve.

Rod Cluster Control Changing Fixture

A fixture is mounted on the refueling cavity wall for removing rod cluster control (RCC) assemblies from spent fuel assemblies and inserting them into new fuel assemblies. The fixture consists of two main components; a guide tube mounted to the wall for containing and guiding the RCC assemblies, and a wheel-mounted carriage for holding the fuel assemblies and positioning fuel assemblies under the guide tube. The guide tube contains a pneumatic gripper on a winch that grips the RCC assembly and lifts it out of the fuel assembly. By repositioning the carriage, a new fuel assembly is brought under the guide tube and the gripper lowers the RCC element and releases it. The manipulator crane loads and removes the fuel assemblies into and out of the carriage.

Spent Fuel Assembly Handling Tools

The Spent Fuel Handling Tools are used with the Fuel Pool Bridge Crane to move spent fuel assemblies in the spent fuel pools. They are manually operated through a mechanical linkage and use four cam-actuated fingers. The shank of the tools is long enough to prevent raising spent fuel elements to a height in the pool where insufficient radiation shielding is available for personnel.

Section 2.4, “Description of Equipment,” of Enclosure 1 to the license amendment supplement provided additional design information for the SFP bridge crane and manipulator crane. The SFP bridge crane uses the single failure proof west hoist for two heavy loads occasionally moved over stored fuel, the pool divider gates, and the pool covers. The manipulator crane was designed to withstand an earthquake while retaining a fuel assembly in the gripper without derailing or losing any components.

The licensee classified the above fuel handling equipment as Quality Assurance (QA) Type I. In general, QA Type I is safety-related per the PINGP classification scheme. Section 1.5, “General Design Criteria,” of the PINGP USAR, Revision 35, provides the following definition of QA Type I under “Criterion 1 – Quality Standards”:

QA Type I - Those items for which the Quality Assurance Program must assure the highest feasible degree of quality standards consistent with the importance of the safety function to be performed. This category includes those items of the plant which are essential to the prevention of accidents which could affect the public health and safety by the release of quantities(1) of radioactivity or are required in the mitigation of the consequences of such accidents.

A footnote defines the quantities of radioactivity considered substantial as follows:

(1) A substantial amount of radioactivity is defined as that amount of radioactive material which would produce radiation levels at the site boundary in excess of 1% of 10 CFR 100.

As explained below, the proposed change essentially replaces the 1% criterion in the footnote with a 10 % criterion. The result would be that even if the analysis of the failure of any of the

affected equipment might result in the uncontrolled release of radioactivity in excess of 1% of limit, the equipment would not be classified as "QA Type I" so long as the release was below a new criterion of 10%.

2.2 Proposed Changes

The LAR, as modified by the supplement dated December 4, 2018, proposed a change to the licensing basis for fuel handling equipment affecting the quality and safety classification of those components. The proposed change would revise the PINGP USAR to apply the American National Standards Institute/American Nuclear Society (ANSI/ANS) 58.14, "Safety and Pressure Integrity Classification Criteria for Light Water Reactors," 1993 (withdrawn June 18, 2004[1]), definition of a threshold amount of radioactivity released during an accident as part of classifying structures, systems, and components (SSCs) as QA Type I rather than the lower plant-specific value currently used in the USAR. As described in Section 1.2 of ANSI/ANS 58.14-1993, the purposes of the standard are to provide criteria for (1) the safety classification of items in light water reactor nuclear power plants, and (2) the assignment of pressure integrity Classes to pressure-retaining items. The safety classification scheme in ANSI/ANS 58.14-1993 (W2004) for a fuel handling accident uses 10% of 10 CFR 100.11 dose as threshold for when an item is classified as safety-related. By applying this criterion, the licensee proposes to amend its QA Type I definition from "A substantial amount of radioactivity is defined as that amount of radioactive material which would produce radiation levels at the site boundary in excess of 1% of 10 CFR 100." by appending the following text defining a different threshold for fuel handling equipment:

> In accordance with Amendments ###/### the classification of the Manipulator Cranes including the load cells, the Spent Fuel Pool Bridge Crane, the Spent Fuel Transfer System (exclusive of the transfer tube/blind flange), the Rod Cluster Control Changing Fixtures, and Spent Fuel Assembly Handling Tool classification is based on a definition of substantial amount of radioactivity of: that amount of radioactive material which would produce radiation levels at the site boundary in excess of 10% of 10CFR100.

Thus, this revised definition would be used to classify the QA Type of the specified fuel handling equipment, which are primarily used to handle single fuel assemblies. This proposed change would allow the licensee to classify the specified fuel handling equipment, which is currently classified as QA Type I, as augmented quality or not safety-related, depending on applicable design information. The licensee proposed no change to the Design Class or augmented quality provisions applicable to single failure proof hoists and design measures to ensure seismic integrity.

2.3 Regulatory Criteria

In accordance with 10 CFR 50.34(a)(3), each application for a construction permit shall include a preliminary safety analysis report that includes the principal design criteria and a description of the relation of the design bases to the principal design criteria. Pursuant to 10 CFR 50.34(b), the final safety analysis report shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the SSCs and of the facility.

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[1] A "withdrawn" standard is one that may contain outdated material or may have been superseded by another standard. The current version of the standard is ANSI/ANS-58.14-2011 (R2017) (meaining "reaffirmed" Jan 17, 2017).

Section 1.2, "Principal Design Criteria," of the PINGP USAR, Revision 35, includes the following discussion related to the facility principal design criteria:

> The Prairie Island Nuclear Generating Plant was designed and constructed to comply with NSP's [Northern States Power Company] understanding of the intent of the AEC [Atomic Energy Commission] General Design Criteria for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967 [32 FR 10213]. Since the construction of the plant was significantly completed prior to the issuance of the February 20, 1971, Appendix A, General Design Criteria, the plant was not reanalyzed and the FSAR [final safety analysis report] was not revised to reflect these later criteria. However, the AEC Safety Evaluation Report acknowledged that the AEC staff assessed the plant, as described in the FSAR, against the Appendix A design criteria and "... are satisfied that the plant design generally conforms to the intent of these criteria." …

Section 1.5 of the PINGP USAR, Revision 35, included a discussion of conformance with the General Design Criteria of Appendix A to 10 CFR Part 50. This discussion included the following information related to quality standards:

> CRITERION 1 - QUALITY STANDARDS
>
> Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety functions, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.
>
> ANSWER
>
> The systems and components of the facility are classified according to their importance in the prevention and mitigation of accidents which could cause undue risk to the health and safety of the public.
>
> Prairie Island's original classification system utilized a combination of Design Classes and Quality Assurance Types. Design Classes are defined in Section 12 and Quality Assurance Types are defined below.
>
> The original, PINGP specific, classification system has been replaced by a system utilizing industry standard and regulatory documents as input. Thus, the current classification system is based on SSC functions and uses standard industry terminology, such as safety related.

> In general, QA Type I is associated with Safety Related, QA Type II is associated with Augmented Quality (a subset of Non-Safety Related), and QA Type III is associated with standard quality Non-Safety Related. Safety Related, Augmented Quality, and Non-Safety Related are defined in applicable fleet procedures.
>
> A discussion of the codes and standards, quality assurance programs, test provisions, etc., applying to each system is included in that portion of the USAR describing that system. A listing of the applicable sections is included in Section 1.2.
>
> The systems and components of the facility have been classified according to their importance in the prevention and mitigation of accidents which could cause undue risk to the health and safety of the public. The structure, system and component Design Classes are defined in Section 12 and the Quality Assurance Types are defined below. A discussion of the codes and standards, quality assurance programs, test provisions, etc., applying to each system is included in that portion of the USAR describing that system…

Safety-related SSCs are defined in 10 CFR 50.2, "Definitions," as:

> … those structures, systems, and components that are relied upon to remain functional during and following design basis events to assure:
>
> (1) The integrity of the reactor coolant pressure boundary;
> (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
> (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

Section 12.2.1.1, "Classification of Structures and Components," of the PINGP USAR, Revision 35 (ADAMS Accession No. ML18155A456), provides the following information related to the safety classification of structures and components:

> All structures (including the Reactor Building), systems (including instruments and controls), and components are classified as Design Class I, II or III according to their function and importance in relation to the safe operation of the reactor, with emphasis on the degree of integrity required to protect the public. These are listed in Table 12.2-1. The table provides overall design classification of structures and components. For detailed design classifications and boundaries separating different design classes within this overall classification refer to the controlled drawings.
>
> ...
>
> The definition of the Nuclear Safety Design Classifications is given in the following paragraphs:
>
> a. Design Class I

Those structures and components including instruments and controls whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of substantial[1] amounts of radioactivity, and those structures and components vital to safe shutdown and isolation of the reactor.

b. Design Class I*

Some items in Table 12.2-1 are designated as Design Class I* indicating that these items have been originally designed or have been subsequently analyzed or tested to Design Class I, Design Basis Earthquake loading (dynamic) only, and that these items are treated as Design Class III items in all other respects.

[1] A substantial amount of radioactivity is defined as that amount of radioactive material which would produce radiation levels at the site boundary in excess of 1.0% of 10[ ]CFR[ ]100 limits.

Table 12.2-1, "Classification of Structures, Systems and Components," of the PINGP USAR, Revision 35, classifies the SFP bridge crane as Design Class I for structural and load carrying elements only and the manipulator crane as Design Class I*. Other conventional equipment has been classified as Design Class III., which is designed to commercial standards and is not safety related.

Guidance for NRC staff review of fuel handling equipment design is contained in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light Water Reactor] Edition," Section 9.1.4, "Light Load Handling System and Refueling Cavity Design" (ADAMS Accession No. ML13318A923). Section 9.1.4 of NUREG-0800 addresses handling of fuel and spent fuel which, if dropped, mishandled, or damaged, could cause releases of radioactive materials or unacceptable personnel radiation exposures. This section indicates that ANSI/ANS 57.1-1992, "Design Requirements for LWR Fuel Handling Systems," provides guidance for safe handling of irradiated fuel. Section 6.2 of ANSI/ANS 57.1-1992 states that fuel handling equipment is generally not classified as safety-related.

3.0 TECHNICAL EVALUATION

3.1 Dose Threshold for Quality Assurance Type 1

The LAR, as amended, proposed a change to the design basis dose threshold for classification of SSCs used for handling of single fuel assemblies as QA Type I, which generally conforms to classification as safety-related. For this specific set of SSCs, the LAR, as amended, proposed to increase the dose threshold from 1 percent of 10 CFR Part 100 dose limits to 10 percent of 10 CFR Part 100 dose limits. The licensee cited conformance with ANSI/ANS 58.14-1993 as part of the basis for this change.

Although the NRC staff has not endorsed the use of ANSI/ANS 58.14-1993 (W2004) for designating the safety classification of facility SSCs, the standard provides a basis for using 10 percent of the guideline exposures of 10 CFR 100.11 as a threshold for classification as safety-related. The amount of radioactivity associated with the exposure limits specified in 10 CFR 100.11(a)(1) and 10 CFR 100.11(a)(2), respectively, are those fission product releases that result in:

- a dose to an individual at the PINGP exclusion area boundary for two hours immediately following the onset of the release of 25 rem whole body or 300 rem to the thyroid from iodine exposure
- a dose to an individual at the PINGP low population zone boundary during the entire period of the radioactive cloud passage of 25 rem whole body or 300 rem to the thyroid from iodine exposure

Section 14.5.1, "Fuel Handling," of the PINGP UFSAR, Rev. 35 (ADAMS Accession No. ML18166A204), describes the consequence analysis methods used to determine the dose to individuals at the exclusion area and the low population zone boundaries. This UFSAR Section describes that the accident source term (i.e., the quantity and isotopic and chemical composition of released fission products; see the definition in 10 CFR 50.2) comply with the requirements of 10 CFR 50.67, and the method used to calculate the dose conformed with the guidance of USNRC Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." This analysis provides a dose to individuals at the exclusion area and low population zone boundaries based on the radioactivity released by the design basis fuel handing accident. The licensee determined the limiting dose would occur at the exclusion area boundary and would be 2.28 rem total effective dose equivalent (TEDE), which is less than 10 percent of the 10 CFR 100.11(a)(1) dose limit of 25 rem whole body. In accordance with the definitions provided in 10 CFR 20.1003, *dose* is a generic term that includes reference to *total effective dose equivalent,* which is defined as the sum of the products of the dose to the organ or tissue and a weighting factor applicable to the irradiated organ or tissue for all exposed organs and tissues and for both external and committed internal radiation exposures. Therefore, TEDE includes whole body dose as a component.

Section 4.2.2 of ANSI/ANS 58.14-1993 (W2004) equates the term "comparable" and the phrase "greater than or equal to 10%." The dose limit alone is not a risk-informed metric because the dose limit reflects consequences only. Risk, as commonly used, reflects the product of the probability of an outcome and its consequences. A footnote associated with 10 CFR 100.11 (and a separate footnote associated with 10 CFR 50.67) indicate that the fission product releases used in the calculations for comparison with the dose values should be based on a major accident that bounds all other accidents considered credible for the facility, which are very low probability events. In RG 1.183, the staff prescribes lower dose limits for specific accidents, based in part on the likelihood of occurrence. The prescribed dose limit for the fuel handling accident is 6.3 rem TEDE at the exclusion area boundary, which is approximately 25 percent of the dose limit of 25 rem specified in 10 CFR 50.67. This lower dose limit reflects both the much greater likelihood of a fuel handling accident than a bounding reactor accident and the low probability that a fuel handling accident would result in an actual release comparable to the release assumed for the accident using the conservative fuel handling accident assumptions specified in Appendix B, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident," to RG 1.183. The greater likelihood of a fuel handling accident results from a combination of factors that include the frequency of fuel handling, the classification of fuel handling equipment as generally not safety-related per the guidelines of ANSI/ANS 57.1, and the significant contribution of human factors to fuel handling accidents. Thus, while a direct comparison between a fraction of a dose limit and a criterion for specifying safety-related equipment is not consistent with risk-informed principles, the request to establish a threshold dose value for designating fuel handling equipment as safety-related at 10 percent of Part 100 dose limits is conservative relative to the dose criteria specified in RG 1.183 for the fuel handling accident.

The definition of safety-related SSCs from 10 CFR 50.2 includes the criterion of having the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in Section 100.11 of 10 CFR Part 100. The SSCs that contribute significantly to preventing or mitigating the consequences of design basis accidents should be subject to higher standards in their design, fabrication, and procurement of replacement parts. However, the specific fuel handling equipment specified in this amendment request has not been credited with preventing a fuel handling accident because the accident analysis assumes the accident occurs. This specified fuel handling equipment also does nothing to mitigate the consequences of a fuel handling accident. Therefore, the change in the design basis accident dose threshold for designating certain fuel handling equipment as QA Type 1 (safety-related) is consistent with the intent of the definition of safety-related in 10 CFR 50.2 and is acceptable.

3.2 Change in Classification of Fuel Handling Equipment

The LAR stated that the effect of the change in the threshold is to allow a change in the QA Type classification of select fuel handling equipment that normally handles a single fuel assembly at a time. The PINGP QA type controls application of the full spectrum of quality assurance measures to the design, manufacture, and procurement of safety-related SSCs. However, Section 12.2.1.1 of the PINGP updated final safety analysis report provided separate design safety classifications for specific structures and components. Specifically, this Section provides for Design Class I and Design Class I*, which allow for designation of augmented quality provisions to certain features or functions for structures and components. As noted above, Table 12.2-1 of the PINGP USAR classifies the SFP bridge crane as Design Class I for structural and load carrying elements only and the manipulator crane as Design Class I*. Other conventional fuel handling equipment has been classified as Design Class III., which is designed to commercial standards and is not safety related.

The classification of the structural and load carrying elements of the spent fuel bridge as Design Class I provides reasonable assurance that the quality of those elements used to prevent damage to greater than one fuel assembly would be subject to augmented quality controls. These augmented quality controls reasonably assure that the single-failure-proof hoist at the west end of the spent fuel bridge will be able to stop and hold loads greater than the weight of a single fuel assembly and that the integrity of the bridge structure itself would not be challenged by design-basis events. These design attributes are necessary to protect against equipment failures that could result in consequences greater than those associated with the design-basis fuel handling accident. Therefore, the proposed change in the threshold dose for classification of fuel handling equipment as Quality Type I is acceptable because it would not affect the quality assurance applied to the important-to-safety functions of the spent fuel bridge crane structure and its single-failure-proof hoist.

Similarly, the classification of the manipulator crane as Design Class I* provides reasonable assurance that the quality of those elements used to prevent damage to greater than one fuel assembly would be subject to augmented quality controls. These augmented quality controls reasonably assure that the integrity of the manipulator crane structure would not be challenged by design-basis events. This design attribute is necessary to protect against structural failure that could result in consequences greater than those associated with the design-basis fuel handling accident. Therefore, the proposed change in the threshold dose for classification of fuel handling equipment as Quality Type I is acceptable for the manipulator crane because it would not affect the quality assurance applied to the important-to-safety function of the manipulator crane structure.

All other attributes of the spent fuel bridge crane and manipulator crane, and all other equipment used to handle only one fuel assembly at a time are classified as Design Class III and, as result of the proposed change to the definition applied to Quality Type I, would no longer be classified as safety-related. The Design Class is unchanged from the existing USAR description. The change in classification to not safety-related is consistent with the definition of safety-related from 10 CFR 50.2 because the specified equipment does not prevent or mitigate the consequences of the design-basis fuel handling accident. This change is also consistent with the guidance of Section 9.1.4 of NUREG-0800 for quality classification of fuel handling equipment. Therefore, the proposed change to criteria for classification of certain fuel handling structures and components as Quality Type I is acceptable.

4.0 STATE CONSULTATION

In accordance with the U.S. Nuclear Regulatory Commission (NRC or Commission) regulations, the Minnesota State official was notified of the proposed issuance of the amendments on August 20, 2019. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding January 31, 2019 (84 FR 812). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Steve Jones, NRR

Date of issuance: November 7, 2019.

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: MODIFYING THE DESIGN BASIS FOR QUALITY CLASSIFICATION OF CERTAIN FUEL HANDLING EQUIPMENT (EPID L-2018-LLA-0261) DATED NOVEMBER 7, 2019

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NAME	DRoth	NSalgado ***(SWall for)***	RKuntz
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