L-PI-04-040
10 CFR 50.90

MAR 25 2004

U S Nuclear Regulatory Commission
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PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKETS 50-282 AND 50-306
LICENSE Nos. DPR-42 AND DPR-60

LICENSE AMENDMENT REQUEST (LAR) DATED MAR 25 2004
MAIN STEAM LINE BREAK MASS AND ENERGY RELEASE USING BAW-10169-A

Pursuant to 10 CFR 50.90, Nuclear Management Company, LLC (NMC), hereby requests the following amendment to the Operating Licenses for the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2. This LAR proposes changes to the PINGP licensing basis and does not include any material changes to the Facility Operating License, Technical Specifications (TS) or TS Bases. Upon approval, the licensing basis changes proposed in this LAR will be made to the Updated Safety Analysis Report (USAR).

The proposed amendment would allow the use of the methodology described in Framatome-ANP (FRA-ANP) Topical BAW-10169-A "RSG Plant Safety Analysis – B&W Safety Analysis Methodology for Recirculating Steam Generator Plants", dated October 1989 that utilizes the RELAP5/MOD2-B&W code described in Topical BAW-10164-A "RELAP5/MOD2-B&W – An Advanced Computer Program for Light-Water Reactor LOCA and Non-LOCA Transient Analysis", Rev 3, dated October, 1996 for the generation of mass and energy (M&E) release rates during a Main Steam Line Break (MSLB) accident for PINGP. As discussed in Exhibit A, the RELAP5/MOD2-B&W code is technically appropriate for generating mass and energy release rates during a MSLB and the MSLB methodology described in BAW-10169-A (with the appropriate conservative inputs) will generate conservative mass and energy release rates.

The NRC Safety Evaluation Report for topical BAW-10169-A, dated August 20, 1989 states the methodology is "...acceptable for calculating the reactor system response in

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performing safety analysis of non-LOCA transients and accidents." This acceptance clearly included analyses evaluating the core response during a MSLB, i.e. margin to critical heat flux. This methodology has been approved for use by other utilities for analyzing the core response to a MSLB as part of reload safety analyses, e.g. Safety Evaluation of Topical Report BAW-1173-P, Revision 2, Mark-BW Reload Safety Analysis for Catawba and McGuire, dated February 20, 1991.

In support of the PINGP Unit 1 Replacement Steam Generator project, FRA-ANP performed an analysis that generated M&E release rates during a MSLB using the methodology described in topical BAW-10169-A. While performing the 10 CFR 50.59 evaluation on the FRA-ANP generated MSLB M&E release rates, NMC was unable to conclude that the NRC acceptance of the MSLB methodology described in BAW-10169-A included the intended application of generating M&E releases to be used in subsequent analyses. To determine the NRC's position on this subject, a conference call was held with the NRC on February 12, 2004. During the call the NRC Staff indicated that the use of BAW-10169-A for generation of MSLB M&E would be appropriate since the core response is also dependant on maximizing the M&E release. However the NRC Staff was unsure what process should be used for adoption of BAW-10169-A as the methodology for generating of MSLB M&E release rates. The NRC Staff indicated that they would discuss the issue with their management and inform NMC if a 10 CFR 50.59 Evaluation would suffice or an LAR is required. On March 11, 2004 the NRC informed NMC that a LAR should be submitted. Thus this LAR is being submitted to the NRC for approval.

Based on the discussion in the attached Exhibit A, NMC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

NMC requests approval of the proposed amendment by August 26, 2004.

Exhibit A contains the licensee's evaluation of this proposed change. Exhibit B presents the proposed USAR mark-ups.

This LAR contains no new commitments and no revisions to existing commitments.

In accordance with 10 CFR 50.91, NMC is notifying the State of Minnesota of this LAR by transmitting a copy of this letter and attachments to the designated State Official.

NUCLEAR MANAGEMENT COMPANY, LLC

Please address any comments or questions regarding this LAR to Mr. H Oley Nelson at 1-651-388-1121.

I declare under penalty of perjury that the foregoing is true and accurate.
Executed on MAR 25 2004

Joseph M. Solymossy
Site Vice President, Prairie Island Nuclear Generating Plant

CC Regional Administrator, USNRC, Region III
Project Manager, Prairie Island Nuclear Generating Plant, USNRC, NRR
NRC Resident Inspector - Prairie Island Nuclear Generating Plant
Glenn Wilson, State of Minnesota

Attachments:

Exhibit A, Licensee Evaluation
Exhibit B, Proposed USAR Changes (mark up)

Exhibit A

L-PI-04-040

LICENSEE EVALUATION

Main Steam Line Break Mass and Energy Release Using BAW-10169-A

1.0 DESCRIPTION

This license amendment request (LAR) is a request to change the licensing basis of the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2 operating under Licenses DPR-42 and DPR-60.

The Nuclear Management Company (NMC) is requesting that the Nuclear Regulatory Commission (NRC) approve the use of the methodology described in Framatome-ANP Topical BAW-10169-A (reference 2) that utilizes the RELAP5/MOD2-B&W code described in Topical BAW-10164-A (reference 1) for the generation of predicted mass and energy (M&E) release rates during a Main Steam Line Break (MSLB) accident for PINGP.

2.0 PROPOSED CHANGE

A brief description of the proposed changes to the PINGP Update Safety Analysis Report (USAR) is provided below. The specific wording changes to the USAR are provided in Exhibit B.

USAR Section 14.3.1 "Calculation Methods and Input Parameters" contains a brief description of the computer codes and methodologies used in performing the transient and accident analysis for the PINGP. The proposed change adds a similar description of the RELAP5/MOD2-B&W (RELAP5) code and Framatome-ANP Topicals BAW-10164-A (reference 1) and BAW-10169-A (reference 2) along with referencing the approval of this LAR for the generation of the MSLB M&E releases.

USAR section 14.5.5.3.1 "Containment Response", describes the methodology, key input assumptions, and single active safety grade component failure assumptions used in analyzing the containment response during a MSLB. The information currently contained in this section of the USAR is based on the methodology that the NRC approved in reference 3. With the addition of the Framatome-ANP methodology for the generation of the M&E portion of the analysis, this section is revised to reflect the new methodology along with the changes in key inputs that result from the use of Framatome-ANP methodology. The changes to the inputs will be discussed in more detail in Section 4.0 below.

USAR Section 14.5.5.3.4 "Containment Temperature Profile", describes the methodology, key input assumptions, and single active safety grade component failure assumptions used in analyzing the containment temperature profile during a MSLB. The information currently contained in this section of the USAR is based on the methodology that the NRC approved in reference 3. With the addition of the Framatome-ANP methodology for the generation of the M&E portion of the analysis, this section is revised to reflect the new methodology along with the changes in key inputs that result from the use of Framatome-ANP methodology. The changes to the inputs will be discussed in more detail in Section 4.0 below.

USAR Section 14.11 "References", lists the references for USAR section 14. The Framatome-ANP topicals BAW-10169-A and BAW-10164-A are added to the list of references.

In summary, this change will allow the use of the RELAP5/MOD2-B&W code and methodology described in Framatome-ANP's Topical BAW-10169-A to generate mass and energy releases during a MSLB accident.

3.0 BACKGROUND

NMC has historically performed the safety analyses that generated the M&E releases during a MSLB accident for PINGP using the approved methodology in reference 3. However, due to limitations and restrictions the current methodology can not be used to generate the M&E releases for the Framatome-ANP model 59/16 replacement steam generators that are planned to be installed in Unit 1. Consequently Framatome-ANP (FRA-ANP) performed a MSLB analysis based on the methodology described in reference 2 to generate M&E releases. These M&E releases were then used in a containment analysis that demonstrated that the use of the replacement steam generators would not exceed or alter the design basis limits for fission product barriers.

The use of the methodology in reference 2 was based on the summary statements contained in the NRC Safety Evaluation Report (SER) (reference 5) which states:

> The staff has reviewed Topical Report BAW-10169 and finds that RELAP5/MOD2-B&W and the reactor system modelings of Figures 6.1 and 6.2 are acceptable for calculating the reactor system responses in performing safety analysis of Non-LOCA transients and accidents.

However NMC could not determine that NRC's approval of topical BAW-10169 (reference 5) included the intended application of generating M&E release rates during a MSLB for use in a containment analysis. Hence the need for this LAR.

4.0 TECHNICAL ANALYSIS

RELAP5/MOD2-B&W Computer Code

The RELAP5 code, which is modularized according to components and functions, has been designed to model the behavior of all major components in the reactor system during accidents ranging from large-break Loss Of Coolant Accidents (LOCAs) to anticipated operational transients involving the plant control and protection system. The primary system, secondary system, feedwater train, system controls, and core neutronics can be simulated. Special component models include pumps, valves, heat structures, turbines, and accumulators. The fundamental equations, constitutive models and correlations, and method of solution of RELAP5/MOD2 are described in NUREG/CR-4312 and NUREG/CR-5194. RELAP5/MOD2-B&W preserves the original models of RELAP5/MOD2 and adds features and models required for licensing analysis for both LOCA and Non-LOCA accidents and transients.

The RELAP5 code received NRC approval in reference 4 for calculating the reactor system responses in safety analysis of transients and accidents, including large-break and small-break LOCAs. The SER states that the NRC found that the RELAP5 code contains appropriate phenomenological models suitable for calculating both LOCA and Non-LOCA transients. The NRC also found that the code contains nothing that is plant-specific in nature or that would preclude the application of the code to any of the recirculating steam generator plants. The NRC concluded that the code could be applied to any of the proposed Westinghouse and Combustion Engineering plants. The restrictions and limitations placed on the use of the code by the NRC in the SERs contained in reference 4 are only applicable to once-through steam generators and LOCA analyses. Thus there are no code restrictions identified in the SERs related to performing a MSLB mass and energy analysis.

Based on the above discussion it is concluded that using the RELAP5/MOD2-B&W code is technically appropriate for generating mass and energy releases during a MSLB.

Topical BAW-10169-A, October 1989

Reference 2 describes the approach used by FRA-ANP to perform safety analyses for Non-LOCA transients. Rather than illustrating the complete safety analysis methodology, the topical describes how the RELAP5 code will be used for the modeling of the reactor primary and secondary systems, and contains comparisons of the RELAP5 results to the safety analyses for several plants for selected representative transients. One of the selected transients was a Steam System Piping Failure. Although the discussion in reference 2 focused on what will be referred to as the "core response" (i.e. determining the core heat flux, Reactor Coolant System (RCS) temperature, RCS pressure, and Departure from Nucleate Boiling (DNB)), the discussion did include comparisons of the calculated mass release rate from the secondary system, e.g. Figure 7.5.28 of reference 2. The concern in a core response to a MSLB is that the cooldown may lead to a return to power and thus potential fuel damage. This cooldown is caused by the heat transfer from the primary to secondary systems, which in turn is dependant on the mass and energy release from the faulted steam generator. Thus the key elements of a core response methodology such as heat transfer coefficients, secondary side modeling, and use of the Moody choked flow models are selected to maximize the cooldown by maximizing the primary to secondary heat transfer including maximizing the mass and energy release from the faulted SG. Therefore the key elements of a core response methodology would be the same as for a methodology intended to maximize M&E releases that would be used in other analyses such as a containment response analysis. The difference in the analyses then becomes the selection of inputs used to maximize the desired response, e.g. initial RCS flow or the inclusion/removal of passive heat sinks. Since the key elements of the methodology are the same and the evolution of a MSLB transient is the same, the MSLB analysis in reference 2 is representative of a MSLB M&E analysis other than for the selection of inputs.

The SER in reference 5 contained eight conditions and restrictions on the acceptance of Topical report BAW-10169. Seven of the restrictions are not relevant to the calculation of mass and energy releases during MSLB because they deal with DNB calculations, selection of transients to be analyzed for a full reload safety evaluation, mixed cores, and Non-MSLB transients. The remaining restriction, number 4, requires that operation of control systems be neglected unless they make the results of the analysis more severe. The inputs selected for the M&E analyses will not model control systems except for those systems that make the results more severe, e.g. Feedwater control system.

To ensure a conservative return to power (and thus additional available heat input) one of the methodology requirements in references 2 and 5 is a 50/50 faulted/unfaulted loop reactivity weighting scheme. However the representative analyses performed in the reference 2 modeled a 4 loop plant by combining the 3 intact loops into one unfaulted loop. The combination of modeling 3 intact loops as one and the 50/50 reactivity weighting scheme results in the flow from the faulted loop having three times the reactivity impact as the flow from any of the 3 intact loops. To maintain the same impact for a two loop plant such as PINGP, the required 50/50 reactivity weighting scheme corresponds to a input weighting of 75/25 faulted/unfaulted.

Based on the above discussion it is concluded that using the MSLB methodology described BAW-10169-A (with the appropriate conservative inputs discussed below) will generate conservative mass and energy releases.

<u>Inputs</u>

The following is a discussion of the key inputs that are used to ensure that conservatively large M&E release rates are generated during a MSLB.

The RELAP5 model used to perform the analysis will be generated consistent with that described in BAW-10169-A. This model includes sources of energy such as the sensible heat of the steam generator metal, steam generator tubing and reactor vessel metal.

For those breaks that are between the non-return check valves and the faulted steam generator, the analysis will include steam flow from the intact steam generator through the pressure balancing line and out the break until the non-return check valve closes on the faulted loop. The non-return check valve is assumed to close 1 second after forward flow ceases consistent with PINGP current licensing basis (see reference 6). In addition, steam located in the steam line between the non-return check valve and the break location is also included in the release.

To maximize the mass available for release, a conservatively high initial steam generator liquid inventory will be assumed.

Consistent with the current PINGP licensing basis (see reference 6), the inputs to the modeling of the Feedwater control system are such that Main Feedwater Pumps continue to run at full capacity until a Feedwater Isolation signal is received at which time the pumps are tripped and allowed to coast down. The Main Feedwater Regulator and Bypass valves are opened at a rate faster than the non-emergency opening rate and close at a rate slower than the emergency rate after receipt of the Feedwater Isolation signal. The unisolated portions of the feedwater line are allowed to empty into the faulted steam generator.

The inputs for the Auxiliary Feedwater system model are such that they maximize the amount of water injected into the faulted steam generator until they are tripped by the automatic run-out protection.

To maximize the heat transfer from the primary side to the faulted steam generator, a conservatively high RCS flow is assumed.

Sensitivity studies

To ensure that the limiting conditions are found and analyzed, a spectrum of cases will be run for various power levels, break sizes and single active safeguard failure assumptions. For the generation of the M&E release rates to be used in a containment response analysis, the following sensitivity studies will be performed:

The sensitivity studies will cover a range of power levels from 0% power up to and including 102% of licensed full power.

The sensitive studies will cover break sizes from 0.6 ft^2 to the largest possible double end rupture.

The single failure sensitivity studies include assuming the failure of a safeguards train of equipment, the failure of the Feedwater Regulator valve to the faulted steam generator to close, and the failure of the Main Feedwater pump to trip.

Conclusion

Performing a MSLB M&E analysis using the RELAP5 code and the methodology described in BAW-10169-A with inputs discussed above will result in a conservatively high calculation of the mass and energy releases during a MSLB at PINGP.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

The Nuclear Management Company has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No

   The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing the use of the methodology described in Framatome-ANP Topical BAW-10169-A "RSG Plant Safety Analysis – B&W Safety Analysis Methodology for Recirculating Steam Generator Plants" that utilizes the RELAP5/MOD2-B&W code described in Topical BAW-10164-A "RELAP5/MOD2-B&W – An Advanced Computer Program for Light-Water Reactor LOCA and Non-LOCA Transient Analysis" for the generation of predicted mass and energy releases during a Main Steam Line Break accident.

   The methodology used to perform an analysis of a main steam line break is not an accident initiator, thus changing the methodology does not increase the probability of an accident.

   The mass and energy releases generated by the proposed methodology will be utilized to demonstrate that the design basis limits for fission product barriers are not exceeded. The proposed methodology does not alter the nuclear reactor core, reactor coolant system, or equipment used directly in mitigation of a main steam line break, thus radioactive releases due to a main steam line break accident are not affected by the proposed change in analysis methodology. Therefore, this change does not increase the consequences of an accident previously evaluated.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing the use of the methodology described in Framatome-ANP Topical BAW-10169-A "RSG Plant Safety Analysis – B&W Safety Analysis Methodology for Recirculating Steam Generator Plants" that utilizes the RELAP5/MOD2-B&W code described in Topical BAW-10164-A "RELAP5/MOD2-B&W – An Advanced Computer Program for Light-Water Reactor LOCA and Non-LOCA Transient Analysis" for the generation of predicted mass and energy releases during a Main Steam Line Break accident.

The analysis of a main steam line break using the proposed methodology does not alter the nuclear reactor core, reactor coolant system, or equipment used directly in mitigation of a main steam line break.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing the use of the methodology described in Framatome-ANP Topical BAW-10169-A "RSG Plant Safety Analysis – B&W Safety Analysis Methodology for Recirculating Steam Generator Plants" that utilizes the RELAP5/MOD2-B&W code described in Topical BAW-10164-A "RELAP5/MOD2-B&W – An Advanced Computer Program for Light-Water Reactor LOCA and Non-LOCA Transient Analysis" for the generation of predicted mass and energy releases during a Main Steam Line Break accident.

The proposed licensing basis change will result in a conservative calculation of the mass and energy releases during a Main Steam Line Break accident. This will ensure that there is no reduction in the margin of safety for analyses that utilize the generated mass and energy releases as inputs. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, the Nuclear Management Company concludes that the proposed submittal presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

General Design Criteria

The construction of the Prairie Island Nuclear Generating Plant was significantly complete prior to issuance of 10 CFR 50 Appendix A General Design Criteria. The Prairie Island Nuclear Generating Plant was designed and constructed to comply with the Atomic Energy Commission General Design Criteria as proposed on July 10, 1967 as described in the Updated Safety Analysis Report. The proposed Atomic Energy Commission General Design Criteria 10 and 49 provide design guidance for containment.

Criterion 10 – Containment

> *Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.*

The proposed amendment does not physically impact the containment structure or the other engineered safety features which together assure that the containment functional capability is retained. The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing use of Topical BAW-10169-A "RSG Plant Safety Analysis – B&W Safety Analysis Methodology for Recirculating Steam Generator Plants" for the generation of mass and energy release rates during a main steam line break accident. These mass and energy release rates may then be used by a containment response code to demonstrate that the required containment integrity will not be lost as a result of a main steam line break.

Criterion 49 – Containment Design Basis

> *The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy*

release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reaction, that could occur as a consequence of failure of emergency core cooling systems.

The proposed amendment does not deal with the energy release following a loss-of-coolant accident. Therefore this criterion is not affected by the proposed license amendment request.

10 CFR 50.59

10 CFR 50.59(c)(2)(viii) requires a licensee to obtain a license amendment pursuant to 10CFR 50.90 prior to implementing a proposed change if the change would "Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses".

Part of the definition of a departure from a method of evaluation per 10 CR 50.59(a)(2)(ii) is:

> Changing from a method described in the FSAR to another method unless that method has been approved by the NRC for the intended application.

The NMC was unable to conclude that the NRC had approved Topical BAW-10169-A for the intended application of generating mass and energy release to be used in subsequent analyses. Hence the NMC is submitting this license amendment request.

NUREG-0800 Standard Review Plan Section 6.2.1.4 "Mass and Energy Release Analysis for Postulated Secondary system Pipe Ruptures"

The Prairie Island Nuclear Generating Plant is not licensed to the criteria listed in NUREG-0800, and nothing in the proposed amendment is intended to commit Prairie Island Nuclear Generating Plant to the criteria in NUREG-0800.

However section 6.2.1.4 of NUREG-0800 does provide guidance for identifying sources of energy that should be included in a mass and energy calculation along with guidance for performing the calculation. The proposed amendment is consistent with the intent of NUREG-0800 section 6.2.1.4 in that it includes the major sources of energy and uses inputs that maximize the mass and energy release rates.

Conclusion

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

6.0 ENVIRONMENTAL CONSIDERATION

The Nuclear Management Company has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCE

1. BAW-10164-A, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light-Water Reactor LOCA and Non-LOCA Transient Analysis, " Rev 3, October, 1996.
2. BAW-10169-A, "RSG Plant Safety Analysis – B&W Safety Analysis Methodology for Recirculating Steam Generator Plants", October 1989.
3. NSPNAD-97002-P Revision 1, "Northern Sates Power Company's Steam Line Break Methodology," October 1998.
4. Letter from Gary M. Holahan, NRC to J. H. Taylor, B&W Fuel Company; Subject: "Acceptance for referencing of Licensing Topical Report BAW-10164P, Revisions 2 and 3, An Advanced Computer Program for Light-Water Reactor LOCA and Non-LOCA Transient analysis" dated March 14 1995.
5. Letter from Ashok C. Thadani, NRC to J. H. Taylor, B&W Fuel Company; Subject: "Acceptance for referencing of Licensing Topical Report BAW-10169P, 'RSG Plant Safety Analysis'", dated August 20 1989.
6. Letter from Tae Kim, NRC to Roger O. Anderson, NSP; Subject: "Prairie Island Nuclear Generating Plant, Units 1 and 2 – Safety Evaluation on Topical Report, NSPNAD-97002P, 'Prairie Island Nuclear Generating Plant Main Steam Line Break Methodology'(TAC Nos. M99108and M99109)", Dated January 21, 2000.