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Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

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Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

50.59 EVALUATION SUMMARY REPORT

With this letter, the Nuclear Management Company, LLC, (NMC) submits two enclosures. Enclosure 1 contains descriptions and summaries of safety evaluations for changes, tests, and experiments made under the provisions of 10 CFR 50.59 during the period since the last update.

Enclosure 2 contains discussion of changes to regulatory commitments made within our Regulatory Commitment Change Process during the period since the last update.

Summary of Commitments

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

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Enclosures (2)

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ENCLOSURE 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
REPORT OF CHANGES, TESTS, AND EXPERIMENTS – DECEMBER 2007

Below are a brief description and a summary of the safety evaluation for each of those changes, tests, and experiments which were carried out at the Prairie Island Nuclear Generating Plant by Nuclear Management Company, LLC (NMC) without prior Nuclear Regulatory Commission (NRC) approval, pursuant to the requirements of 10 CFR 50.59.

50.59 Evaluation No. 1038 – Use of Ultimate Strength Design (USD) Methodology to Evaluate Vertical Seismic Loads on Floors

Description of Change

This non-design change pertains to the use of different methods for evaluating seismic loads than those pre-scribed in the Prairie Island Updated Safety Analysis Report (USAR). The methods of evaluation were used in the following four calculations:

1. Calculation S-B01-VS-001, Evaluation of Auxiliary Building Floors for Vertical Seismic Loads
2. Calculation S-B01-VS-002, Evaluation of Unit 1 Reactor Building Floors for Vertical Seismic Loads
3. Calculation S-B01-VS-003, Evaluation of Unit 1 Reactor Building Floors for Vertical Seismic Loads
4. Calculation S-B01-VS-004, Evaluation of Screenhouse Floors for Vertical Seismic Loads

In each of the above calculations, concrete beams and slabs were analyzed using Ultimate Strength Design (USD) methodology versus Working Stress Design (WSD) methodology prescribed in USAR Table 12.2-6, "Allowable Stresses – Reinforced Concrete," and elsewhere.

Summary of 50.59 Evaluation

Because the activity described in this evaluation is four structural analyses versus a physical change, Questions #1 thru #7 in the Basis of Determination part of this evaluation were determined not to be applicable. Specifically, additional accidents or malfunctions, different accidents or malfunctions, more severe accidents or malfunctions, etc., due to one or more physical changes are not possible.

Question #8 was determined to be applicable. The response to this question concluded that, though the four structural calculations evaluated herein used methodologies different from that described in USAR Section 12, the USD methodology used in the four analyses was in accordance with Sections 3.8.3 & 3.8.4 of NUREG-0800, Standard Review Plan, and was formally endorsed by the NRC in an April 1992 Safety Evaluation Report applicable to the Prairie Island D5/D6 Building. This 50.59 Evaluation determined that it would be appropriate to use this same methodology to evaluate the structural adequacy of existing concrete floors in four other Design Class 1 structures.

50.59 Evaluation No. 1039 – Revised Auxiliary Building High Energy Line Break Analysis

Description of Change

The change being evaluated is a new Auxiliary Building High Energy Line Break Analysis utilizing mass and energy releases generated by Westinghouse and new initial compartment temperatures. This analysis encompasses mass and energy release from both the Framatome Replacement Steam Generators as well as the original Westinghouse Steam Generators. The analysis determined the peak compartment pressures as well as the temperature profile. The pressure results were utilized to ensure the structural integrity of the concrete block walls and steam exclusion doors. The temperature profile was used to ensure the Environmental Qualifications for equipment needed to mitigate the High Energy Line Break was satisfactory.

Summary of 50.59 Evaluation

There are no physical or operational changes being made to any plant equipment. Thus there is no impact on the frequency of occurrence of an accident, nor for an accident of a different type. The resultant Auxiliary Building pressures following a High Energy Line Break satisfy all applicable acceptance criteria and the resultant temperatures are bounded by the equipment qualification records. Thus there is no impact on the likelihood of a malfunction, nor for a malfunction with a different result, nor an increase in consequences of an accident or malfunction. The activity does not involve a design basis limit for a fission product barrier. The methodology used in the analyses has been reviewed against methods described in the USAR and methods approved for other facilities. Based on this review it was concluded that the methodology is not a departure from that described in the Updated Safety Analysis Report.

50.59 Evaluation No. 1050 – Revised Small Break LOCA Analysis Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model (WCAP-10054-P-A Add. 2 Rev. 1)

Description of Change

This 50.59 evaluation involves a change in methodology only. It proposes to officially recognize the revision to the Small Break Loss of Coolant Accident (SBLOCA) methodology described in WCAP 10054-P-A, Addendum 2 Revision 1. The Nuclear Regulatory Commission (NRC) has expressed concerns about a portion of this methodology regarding removing the loop seal clearing restriction for the intact loop. As a result, Prairie Island will adopt all portions of this methodology with the exception that the loop seal clearing restriction will be kept for the intact loop (only the broken loop loop seal will be allowed to fully clear).

The revised methodology, as documented in WCAP 10054-P-A, Addendum 2 Revision 1, was approved by the NRC for use at other Westinghouse PWRs (Pressurized Water Reactors) and has been reviewed for applicability to Prairie Island. All the input assumptions of this revised methodology as well as conditions and limitations bound Prairie Island Units 1 and 2. Therefore, this new methodology is acceptable for use for Prairie Island Units 1 and 2.

Summary of 50.59 Evaluation

This proposed change is only a change to a methodology, so only question 8, "Does the activity result in a departure from a method of evaluation described in the USAR, or any pending submittal, used in establishing the design bases or in the safety analysis?" of the Basis Of Determination applies.

The proposed change modifies the SBLOCA methodology. The revised methodology has been approved for other PWRs (Kewaunee) and the NRC acknowledged that the new methodology is applicable to Westinghouse PWR designs including those with UPI (Upper Plenum Injection). Prairie Island Units 1 and 2 meet all the input assumptions of the new methodology as well as conditions and limitations stated in WCAP 10054-P-A, Addendum 2, Revision 2. Therefore, the change in methodology does not result in a departure from a method of evaluation as described in USAR Section 14.7 used in establishing the design bases or the safety analyses.

50.59 Evaluation No. 1054 – Unit 1 Cycle 24 Core Reload, Revisions 0 through 3

Description of Change

This design change is required to allow for continued power operation of Prairie Island Unit 1 for approximately 18 months. The fuel in the current core will be burned to a state that no longer allows for significant full power operation. This reload will replace burned fuel from Unit 1 Cycle 23 with 48 fresh fuel assemblies. This will allow the Unit 1 reactor to produce power at its rated capacity.

Summary of 50.59 Evaluation

The USAR Chapter 14 evaluations performed by NMC's Nuclear Analysis Department (NAD) and Westinghouse demonstrate that the Prairie Island Unit 1 Cycle 24 reload design and associated Core Operating Limits Report (COLR) do not result in the accepted safety limits for any accident being exceeded. The Cycle 24 design is consistent with the description of the core in the USAR. The core contains 121 fuel assemblies using a 14 x 14 fuel rod array, with 29 control rods in the same locations as described in the USAR. The only change from Cycle 23 is the distribution of new and used assemblies. This results in a redistribution of the isotopic distribution of the core that changes the core physics parameters of the reactor. The effect of these changes in the cycle physics parameters on cycle operation and accident analyses have been evaluated using NRC approved methods.

The accident analyses show that no design limits are exceeded during any analyzed transient for the cycle as designed. The cycle does not exceed any fuel burnup limits. Therefore the reload modification for Unit 1 Cycle 24 is safe and consistent with Prairie Island's current Licensing Basis.

50.59 Evaluation No. 1055 – Unit 2 Cycle 24 Core Reload, Revisions 0 and 1

Description of Change

This design change is required to allow for continued power operation of Prairie Island Unit 2 for approximately 21 months. The fuel in the current core will be burned to a state that no longer allows for significant full power operation. This reload will replace burned fuel from Unit 2 Cycle 23 with 56 fresh fuel assemblies. This will allow the Unit 2 reactor to produce power at its rated capacity. Revision 0 is valid only for Modes 5 and 6.

Summary of 50.59 Evaluation

The USAR Chapter 14 evaluations performed by NAD and Westinghouse demonstrate that the Prairie Island Unit 2 Cycle 24 reload design and associated COLR do not result in the accepted safety limits for any accident being exceeded. The Cycle 24 design is consistent with the description of the core in the USAR. The core contains 121 fuel assemblies using 14 x 14 fuel rod array, with 29 control rods in the same locations as described in the USAR. The only change from Cycle 23 is the distribution of new and used assemblies. This results in a redistribution of the isotopic distribution of the core that changes the core physics parameters of the reactor. The effect of these changes in the cycle physics parameters on cycle operation and accident analyses have been evaluated using NRC approved methods. No analysis needed to be re-run for this core design.

The accident analyses show that no design limits are exceeded during any analyzed transient for the cycle as designed. The cycle does not exceed any fuel burnup limits. Therefore the reload modification for Unit 2 Cycle 24 is safe and consistent with Prairie Island's current licensing basis.

50.59 Evaluation No. 1056 – Mode 4 LOCA Technical Specification Bases and Procedural Changes

Description of Change

Activity Description:

- Revise Emergency Procedures 1E4 [2E4] "Core Cooling Following Loss of RHR (Residual Heat Removal) Flow" to rely on the SI subsystem as the primary source of injection during a LOCA in Mode 4 with RHR system as a backup.

- Revise TS Bases B 3.5.3 to clarify that an operable train of Emergency Core Cooling System consists of the SI subsystem for injection flow and the RHR subsystem for recirculation.

Summary of 50.59 Evaluation

The changes to the emergency response procedures and TS Bases do not affect how the equipment is operated, its reliability, or performance; thus there is not impact on the frequency of an accident, likelihood of a malfunction, possibility of a new accident, or possibility of a malfunction with a different result. The evaluation demonstrates that the SI subsystem is capable of removing decay heat and thus the consequences of a LOCA in Mode 1 continues to bound those of a LOCA in Mode 4. The changes do not involve a design basis limit for a fission product barrier nor a method of evaluation. Therefore, the changes meet the design and license basis requirements.

ATTACHMENT 2

CHANGES TO REGULATORY COMMITMENTS

Regulatory Commitment Change 05-03 - Change Frequency of Residual Heat Removal (RHR) Pump Venting in Reduced Inventory

Change the RHR suction venting time from every one hour to every six hours. Original commitment was made to ensure air did not build up in the RHR pump suction. Experience has shown that very little air builds up in an hour and that extending the venting frequency is acceptable.

Regulatory Commitment Change 06-01 - Test Frequency Flexibility for Unit 1 Containment Fan Coil Unit Motor-Operated Valves (MOVs)

Nuclear Management Company, LLC (NMC) Committed to the Joint Owners Group (JOG) MPR-18-07 guidance for MOV testing. JOG guidance would have MOVs: MV-32133, MV-32139,MV-32142, MV-32379, and MV-32380 tested on a frequency not to exceed ten years. The ten-year frequency expires on 2/12 and 2/13/06 for these MOVs. Currently, testing at power is not desirable due to the risk of pressure locking. This change would be a one time extension of the ten-year test frequency of these MOVs - testing completed in May 2006 Unit 1 refueling outage.

Regulatory Commitment Change 06-04 - Change Due Date of Limited Exam Relief Requests

Original commitment was to submit Limited Exam Relief Requests with the outage summary report for each limited examination. The change was that Limited Exam Relief Requests will be submitted no more than 12 months after each applicable outage for each limited examination. The reason for the change is that 4th Interval Inservice Inspection is risk-based, so external analysis will be required for all components for which only a limited examination was performed and no prior examination history exists. The 12 month due date will not go beyond the NRC requirement (which is 12 months after the interval).

Regulatory Commitment Change 06-23 - Cancel Commitment on Work Control Process

Original commitment (reconstituted from a June 30, 1975 letter to the NRC) was to revise administrative procedures for the work control process to require consideration of potential effects of work on nearby equipment. This commitment was cancelled because it was deemed unnecessary (hot work process controls have been in place nearly 30 years and will remain regardless of whether there is an associated commitment).

Regulatory Commitment Change 06-24 - Cancel Commitment on Work Control Process and Fire Protection

Original commitment (reconstituted from a June 30, 1975 letter to the NRC) was to revise administrative procedures for the work management process to assure consideration is given to the need for fire prevention or suppression or equipment. This commitment was cancelled because it was deemed unnecessary (hot work process controls have been in place nearly 30 years and will remain regardless of whether there is an associated commitment).

Regulatory Commitment Change 06-25 - Cancel Commitment on Modification Process and Fire Protection

Original commitment (reconstituted from a June 30, 1975 letter to the NRC) was to revise administrative procedures for the modification process to issue a work instruction for modifications designed by offsite organizations, including responsibilities to control installation work. This commitment was cancelled because current modification process controls are better than those of 1975; thus, it is reasonable to assert that such controls will remain to meet the intent of this commitment, even if the commitment no longer exists. That is, this commitment is to do something so fundamental to the modification process in the current day that a commitment is not necessary.

Regulatory Commitment Change 06-26 - Cancel Commitment on Fire Protection Training

Original commitment (reconstituted from a June 30, 1975 letter to the NRC) was to establish a method for indoctrination of offsite personnel on permanent fire protection administrative policies and procedures. This commitment was cancelled because the current badge training process is far beyond what the indoctrination training of 1975 would have been. Current training includes the necessary fire protection training (including hot work permits, combustible source use permits, fire alarms, and extinguisher types) for workers badged for unescorted access (any other "offsite" worker would have to be escorted by a badged/trained worker). The elimination of this commitment will not result in the elimination of fire protection training.

Regulatory Commitment Change 06-27 - Cancel Commitment on Fire Protection and Work Control

Original commitment (reconstituted from a June 30, 1975 letter to the NRC) was to revise the work control process to give specific consideration to Combustible Materials, Ignition Sources, Safety Monitoring Personnel, Training of Personnel, and Work Deferral. This commitment was cancelled because the training piece of this commitment is not needed (current training practices are adequate without a commitment and will remain so) and because the remainder of this commitment is addressed in later commitments.

Regulatory Commitment Change 06-28 - Cancel Commitment on Fire Protection and Indoctrination Training

Original commitment (reconstituted from a December 30, 1975 letter to the NRC) was to establish a method for indoctrination of offsite personnel on permanent fire protection administrative policies and procedures. This commitment was cancelled to the new process – current badge training process is far beyond what the indoctrination training of 1975 would have been. Current training includes the necessary fire protection training (including hot work permits, combustible source use permits, fire alarms, and extinguisher types) for workers badged for unescorted access (any other "offsite" worker would have to be escorted by a badged/trained worker). The elimination of this commitment will not result in the elimination of fire protection training

Regulatory Commitment Change 06-29 - Cancel Commitment on Fire Brigade Training

Original commitment (reconstituted from a June 30, 1975 letter to the NRC) was to provide training for fire teams on the use of fire fighting equipment, including the use of water on electrical fires. This commitment was cancelled because fire brigade training is required, whether or not this commitment exists. In the 30 years since the original commitment was made, the means for fighting electrical fires have been well incorporated into fire brigade training.

Regulatory Commitment Change 06-32 - Change Frequency of Undervessel Bare Metal Visual (BMV) Inspection

Original commitment (from response to NRC Bulletin 2003-02) was to perform a 100% bare-metal visual exam of the lower reactor pressure vessel (RPV) dome up to and including each bottom-mounted instrumentation (BMI) penetration to RPV junction. This examination will be completed on each unit during refueling outages subsequent to the current Unit 2 refueling outage. This commitment was revised to change the frequency to every other refueling outage. The change in frequency is warranted based on site and industry experience during BMV inspections and reduction in radiation dose.

Regulatory Commitment Change 07-01 - Change Commitment on Fire Damper Installation

Original commitment described in the NRC safety evaluation as, "The Licensee has committed to the following modifications: (1) A concrete fire barrier will be placed in the pipe trench that passes through the auxiliary feedwater pump rooms at the boundary between the two rooms. The existing grating will be notched and a 1/4-inch thick checkered floor plate will be tack welded in place to provide resistance to buckling. (2) Fire-rated dampers (3-hour or equivalent) will be installed in all return ventilation ducts that penetrate the boundaries of the rooms." The revised commitment deleted the second item. Fire modeling demonstrated that the physical layout of the ductwork and rooms prevents damage in the auxiliary feedwater pump room from a fire originating in the 480V normal switchgear rooms.