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L-MT-17-075
10 CFR 50.71(e)
10 CFR 50.59(d)(2)

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Monticello Nuclear Generating Plant
Docket No. 50-263
Renewed Facility Operating License No. DPR-22

Submittal of Revision 35 to the Updated Safety Analysis Report

Pursuant to 10 CFR 50.71(e), Revision 35 to the Monticello Nuclear Generating Plant (MNGP) Updated Safety Analysis Report (USAR) is provided. This revision completes an update of the information in the USAR for the period from November 16, 2016, to October 16, 2017.

The changes in this revision reflect the incorporation of modifications, license amendments, and editorial corrections and clarifications. These changes are made in accordance with the guidance provided in Nuclear Energy Institute (NEI) Report NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports", Revision 1, dated June 1999 and Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in accordance with 10 CFR 50.71(e)", dated September 1999.

Enclosure 1, "Report of Changes, Tests and Experiments", indicates that two 10 CFR 50.59 evaluations were performed. This report is provided as required by 10 CFR 50.59(d)(2).

Enclosure 2, "Report of Changes to Licensee Docketed Commitments", indicates that in accordance with the guidance provided in NEI 99-04, "Guidelines for Managing NRC Commitment Changes", dated July 1999, for this period, there were two changes to commitments.

Enclosure 3, "Summary of Information Removed from the USAR", provides the information removed from the USAR for this revision cycle. This information is provided in accordance with Revision 1 of NEI 98-03 and Regulatory Guide 1.181.

A053
A006
NRR

Enclosure 4 contains Revision 35 of the MNGP USAR. The USAR is being submitted electronically, in its entirety, on CD-ROM according to the instructions in Regulatory Issue Summary (RIS) RIS 2001-005, "Guidance on Submitting Documents to the NRC by Electronic Information Exchange or on CD-ROM".

Enclosure 5, "Report of Changes to the Monticello Fire Protection Program", provides a summary of changes to the program. Changes to the Fire Protection Program are provided in accordance with the guidance contained in Generic Letter 86-10, "Implementation of Fire Protection Requirements", dated April 24, 1986.

Summary of Commitments

This letter contains no new commitments.

In accordance with 10 CFR 50.71(e)(2), I certify that the information presented herein, accurately presents changes made since the previous submittal prepared pursuant to Commission requirement and identifies changes made under the provisions of 10 CFR 50.59 not previously submitted to the Commission.

Christopher R. Church
Site Vice President
Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosures (5)

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC (without enclosure 4)

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES, TESTS AND EXPERIMENTS

The following includes a brief description and summary of the 10 CFR 50.59 evaluations performed between November 16, 2016 and October 16, 2017 for those changes, tests and experiments that were carried out without prior U.S. Nuclear Regulatory Commission (NRC) approval, pursuant to the requirements of 10 CFR 50.59(d)(2).

Evaluation No. SCR-15-0378 Revision 1: Replace SBGT (Standby Gas Treatment) Controllers to Improve Reliability

EC 26176, Replace SBGT Controllers to Improve Reliability

Activity Description:
Replaced both controllers for SBGT due to low reliability. The replacement controllers are digital. In order to address the potential for common cause failure; hardware and software diversity will be provided: FIC-2943 will be replaced with a Siemens Model 353 and FIC-2942 will be replaced with a Yokogawa YS1700. Activity has the same scope as SCR-15-0378, Rev. 0, previously submitted with exception of addition of evaluation of EMI/RFI testing.

Summary of Evaluation
The 10 CFR 50.59 evaluation's response to each of the eight 10 CFR 50.59(c)(2) criterion is "No." The evaluation concludes based on regulatory and industry guidance that the use of diverse designs between trains is sufficient to mitigate any common cause failures which are potentially introduced by the use of hardware and software in this application, and based on a comparison of failure mechanisms, the effect of failures of the new controllers are bounded by existing failure mechanisms. Therefore, the evaluation concluded that prior NRC approval is not required. The evaluation shows that the implementation of the digital aspects will not, increase the frequency of accidents, increase consequences of accidents, create new accidents, increase the occurrence of equipment malfunctions, increase consequences of equipment malfunctions, create malfunctions having different results, cause a design basis limit for a fission product barrier to be exceeded or altered, or result in departure of a method of evaluation. The bases for this statement are the same as provided in SCR-15-0378, Rev. 0 with the exception that the EMI/RFI testing ensures that the controllers do not introduce electromagnetic emissions either conducted or radiated that could have the potential to affect other equipment.

Evaluation No. SCR-17-034 Revision 0:
Increase GE14 Nuclear Fuel from 35 to 37 GWD/MTU for Core Average End of Cycle Exposure for Radiological Consequences Evaluations

Calculations 16-090 Revision 0, 04-038 Revision 3, 04-040 Revision 3, 04-041 Revision 3, 08-027 Revision 2 and USAR sections 14.7 and 14.11

Activity Description:
The primary scope of the proposed activity was to create and revise calculations to incorporate changes resulting from Monticello License Amendment 188 (AREVA Fuel Transition), as discussed further below, which were not addressed in the associated license amendment request or NRC safety evaluation.

Calculations determined the radiological consequences (doses) that would be incurred in the control room (CR), exclusion area boundary (EAB), and low population zone (LPZ) target areas as a result of a postulated Loss of Coolant Accident (LOCA), Fuel Handling Accident (FHA), and Control Rod Drop Accident (CRDA). The plant's emergency response Technical Support Center (TSC) dose impact following a LOCA was also evaluated. The radionuclide inventory for a GE14 core having a core average end of cycle exposure of 37 GWD/MTU was evaluated. This inventory determination was performed in accordance with section 3.1 (Fission Product Inventory) of Regulatory Guide (RG) 1.183 with no impact on other RG 1.183 sections.

Summary of Evaluation:

The evaluation was performed for new calculation 16-090 (Rev 0), revised calculations 04-038 (Rev 3), 04-040 (Rev 3), 04-041 (Rev 3), and 08-027 (Rev 2), and associated revisions to USAR Sections 14.07 and 14.11. The new calculation and calculation revisions computed updated design basis accident (DBA) target doses for Monticello based on an increase in the maximum core average end of cycle exposure value assumed in the current licensing basis (CLB) dose calculations from 35 GWD/MTU to 37 GWD/MTU for worse case bounding transition cores containing GE14 fuel. These calculations determined slightly larger doses than those which had been previously evaluated as part of applicable CLB Amendments 176 and 188 and reviewed by the NRC. The Updated Safety Analysis report (USAR) Section 14.07 and 14.11 revisions incorporate changes resulting from the new calculation and calculation revisions. The conclusion of this 50.59 evaluation is that the dose increases are not more than minimal.

The 50.59 evaluation's response to each of the eight 10 CFR 50.59(c)(2) criterion is "No". Therefore, the evaluation concluded that NRC review of the calculation activity and USAR is not required. The evaluations show that a higher assumed core average end of cycle exposure of 37 GWD/MTU for worse case transition cores containing GE14 fuel will not increase the frequency of accidents, increase consequences of accidents more than minimal, create new accidents, increase the occurrence of equipment malfunctions, increase consequences of equipment malfunctions, create malfunctions having different results, cause a design basis limit for a fission product barrier to be exceeded or altered, or result in departure of a method of evaluation.

The bases for this statement are that the calculation revisions and associated USAR changes only involve input to calculations for the evaluation of Alternate Source Term Methodology of Analyses. These analyses are used to determine bounding dose to the USAR described EAB, LPZ, and CR targets for USAR described FHA, CRDA, and LOCA design bases accidents. Dose acceptance criteria of being increased no more than minimal from NRC reviewed dose is the result. The method of evaluation used for the new and revised calculations is the same method used for the previous versions of these calculations which were approved by the NRC.

An increase in consequences from a proposed activity is defined to be no more than minimal if the increase (1) is less than or equal to 10 percent of the difference between the current calculated dose value and the regulatory guideline value (10CFR 100 or GDC 19, as applicable), and (2) the increased dose does not exceed the current Standard Review Plan (SRP) guideline value for the particular design basis event. The change in dose consequence for all results are <1%. Therefore, there is not more than a minimal increase in the consequences of an accident.

ENCLOSURE 2

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES TO LICENSEE DOCKETED COMMITMENTS

Commitments are identified and reported to the Commission in accordance with guidance provided in NEI Technical Report 99-04 Revision 0, "Guidelines for Managing NRC Commitment Changes."

This enclosure provides a brief description and a summary of changes to commitments established with the NRC by the Monticello Nuclear Generating Plant (MNGP) per NEI 99-04 guidelines.

For Revision 35:

Two commitments have been retired from the MNGP commitment tracking program.

1. Commitment Tracking Number:

   M87005A

   Commitment source document:

   Letter from C. E. Larson (NSPM) to H. J. Miller (NRC), "Response to Safety System Functional Inspection Report No. 50-263/87005," dated October 13, 1987 (ADAMS Accession No. ML113202195)

   Original Commitment Wording:

   Appropriate Administrative Controls Documents will be revised to:
   a. *Require verification of as built configuration whenever possible when performing engineering reviews or modifying circuits. As built verification will include comparing independent drawings and checking terminal strip configuration.*
   b. *Ensure that all personnel understand their responsibility to identify drawing discrepancies so that corrective actions can be taken.*
   c. *Require second level review of drawing changes.*

   Justification:

   This Commitment has been retired in accordance with NRC endorced guidance, NEI 99-04, Guidelines for Managing NRC Commitment Changes, because it does not meet the definition of an NRC commitment.

2. Commitment Tracking Number:

   M84118A

   Commitment source document:

   Technical Evaluation Report from C. Bomberger (Franklin Research Center) to NRC, TER-C55506-370, "Control of Heavy Loads: Monticello Nuclear Generating Station," dated January 30, 1984 (ADAMS Accession No. ML113330320)

   Original Commitment Wording:

   The carry height of the NFS-4 and NAC-1 casks, approved for use in NRC's letter of January 25, 1977, shall be limited to a maximum of 6 inches.

   Justification:

   The NFS-4 and NAC-1 casks were used at MNGP before the reactor building crane was upgraded to single failure proof. These casks are no longer available for use.

ENCLOSURE 3

MONTICELLO NUCLEAR GENERATING PLANT
SUMMARY OF INFORMATION REMOVED FROM THE USAR

Consistent with the guidance in Nuclear Energy Institute (NEI) Report NEI 98-03, "Guidelines for Updated Final Safety Analysis Reports," Revision 1 and Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance With 10 CFR 50.71(e)," Information removed from the Monticello Nuclear Generating Plant Updated Safety Analysis Report (USAR) is summarized below.

- <u>USAR Change 01511192</u>
  Affected sections:
  <u>Section USAR-03, Reactor</u>

  <u>3.1, General Summary</u>
  Editorial change to reflect that AREVA fuel is now in use and methods are described in USAR-03.02.

  <u>3.2.2 b., Fuel Damage Limits</u>
  Deletion of specific General Electric (GE) Linear Heat Generation Rate limits dues to implementation of AREVA Linear Heart Generation Rate (LHGR) methods which was approved in License Amendment 191.

  <u>3.2.6 Item 9, Monticello Operating Map</u>
  Deletion of Backup Stability Protection (BSP) boundary description was due to the implementation of license amendment 191, (Extended Flow Window).

  <u>3.3.4, Analytical Methods</u>
  Deletion of Detect and Suppress Solution-Confirmation Density discussion was due to implementation of license amendment 191 (Extended Flow Window)

  <u>3.4.2.3, Water Rods</u>
  Deletion of the description of how the water rod and spacer are assembled in the factory is greater level of detail than required to be consistent with other discussion of bundle component.

  <u>Section USAR-06, Plant Engineered Safeguards</u>

  <u>6.2.5.1, Design Basis</u>
  Deletion of the discussion of the Automatic Depressurization System bypass time setting for 1775 MWthermal (MWt) licensed thermal power operation due to implementation of license amendment 176 (Extended Power Uprate).

Section USAR-07, Plant Instrumentation and Control Systems

7.7.2.2, Description

Deletion of discussion of use of the maximum combined flow limiter position because the maximum combined flow limiter is not used in the AREVA method. The AREVA method was approved for use at Monticello in license amendment 188 (AREVA methods transition).

7.14, References

Deletion of the calculation describing use of the maximum combined flow limiter position because the maximum combined flow limiter is not used in the AREVA method, which was approved for use at Monticello in license amendment 188 (AREVA methods transition).

Section USAR-14, Plant Safety Analysis

14.3.1.2, Calculation of Operating Limit MCPR for Core Reload

Deletion of GE delta-CPR method discussion because AREVA method does not use the delta-CPR method. The new method was approved for use at Monticello in license amendment 188 (AREVA methods transition).

Deletion of power-biased thermal limit multipliers because these multipliers are not part of the approved method, as approved for use at Monticello in license amendment 188.

14.3.4, Power to Flow Operating Map

Deletion of the list of analysis performed for the MELLLA power-flow map expansion because the description is greater level of detail than required to be consistent with other power-flow map extensions described in the USAR.

14.4, Transient Events Analyzed for Core Reload

Deletion of GESTAR-II AOO results due to new results from AREVA methods implemented in license amendments 188 (AREVA methods transition) and 191 (Extended Flow Window).

14.7.6.3.1, Introduction

Deletion of calculation reference 27 because the computation previously performed in this calculation is now performed in Monticello calculation 04-041.

14.11, References

Editorial change, reference 174 was deleted as it was redundant to Reference 163.

- USAR Change 01518940
  Affected sections:
  Section USAR-06 – Plant Engineered Safeguards
  6.1.3.1, Gas Accumulation Management

  The Engineering Evaluations previously referenced were removed as the Gas Accumulation Management Program (GAMP) will reference the engineering documents that support GAMP implementation.

- USAR Change 01553835
  Affected sections:
  Section USAR-08 – Plant Electrical Systems
  8.5.2.3, Performance Analysis

  Deletion of "*two main*" removed from last sentence of section. EC 13284 replaced D40 (125 VDC swing charger) with the same make and model chargers as previously installed for D10 and D20. All three chargers have a high-voltage shutdown feature.

- USAR Change 01550896
  Affected sections:
  Section USAR-10 – Plant Auxiliary Systems
  10.2.2.3, Performance Analysis

  The specific year (1994) of the version ANSI/ANS 5.1 standard removed. The ANSI/ANS 5.1-2005 decay heat standard includes updated data and addresses several areas for future improvement identified in the 1994 standard. The 2005 standard supersedes the 1994 standard.

- USAR Change 604000000032
  Affected sections:
  Section USAR-10 – Plant Auxiliary Systems
  USAR-10.3.4.2.3, Alternate Nitrogen System

  Sentence stating "Train B also provides the sole pneumatic supply to the Primary Containment Hard Pipe Vent System" deleted. The Primary Containment Hard Pipe Vent pneumatic supply was changed from Alternate Nitrogen System Train B to a new pneumatic supply under modification EC 26083.

- USAR Change 604000000100
  Affected sections:
  Section USAR-14 – Plant Safety Analysis
    USAR-14.FIG

    Figures 14.7-1, 14.7-1a, 14.7-1b, 14.7-2, 14.7-3, 14.7-4, 14.7-5, and 14.7-6 have been deleted. Figures are no longer referenced in the USAR.

- USAR Change 604000000031
  Affected sections:
  Section USAR-I.05, Evaluation of High Energy Line Breaks Outside of Containment
    I.5.2.4.3, HPCI Room (Volume 8)
    Table I.5-1
    Table I.5-2

    Deletion of HPCI Room (Volume 8) description.
    The HPCI HELB in the HPCI room at MO-2036 has been deleted per provisions of USAR definition of a terminal end, in that a closed valve is not a terminal end if it is not supported as an anchor. A crack or break at this closed valve is not postulated since the pipe stress at this valve is less than that of USAR limits for which cracks and breaks must be postulated.

ENCLOSURE 5

MONTICELLO NUCLEAR GENERATING PLANT
REPORT OF CHANGES TO THE MONTICELLO FIRE PROTECTION PROGRAM

This enclosure contains a report of changes to the Monticello Fire Protection Program (FPP) in accordance with the provisions of Generic Letter (GL) 86-10.

In conformance with GL 86-10, the Updated Fire Hazards Analysis (UFHA) and the Safe Shutdown Analysis (SSDA) are incorporated directly into the Updated Safety Analysis Report (USAR) as Appendix J.5 and J.4 respectively. The following summarizes fire protection program document changes since the previous submittal.

1. USAR J.04, SAFE SHUTDOWN ANALYSIS

   Revision 35

   - USAR 01511192
     - Added additional references for the transition to AREVA fuel.
   - USAR 01544518
     - Tables J.4.5-1 and J.4.6-1 – Editorial change to correct safe shutdown division for DG 190-DG2
   - USAR 604000000075
     - Tables J.4.5-1, J.4.5-2, and J.4.6-1 – Delta SRV tailpipe pressure sensor was added for information purposes and consistency.
     - Tables J.4.5-1 and J.4.6-1 – Revised the fire zone identifier for RHR Air Compressors (K-10A and K-10B)

2. USAR J.05, FIRE HAZARDS ANALYSIS

   Revision 35

   - USAR 604000000071
     - Revision to assumption 3.9 to be consistent with USAR J.05 Section 5.3, VII Construction Language.
     - Table 2 Portable Extinguisher changed from NO to YES for Fire Zones 1D, 4C, 5C, 23B, and 40. It was changed from YES to NO in Fire Zone 42. These changes were then reflected in the corresponding Fire Hazards Analysis Matrix for the listed Fire Zones.
     - Table 2 Safe Shutdown Systems Affected changed from YES to NO for Fire Zone 11. This change was then reflected in the Fire Hazards Analysis Matrix for Fire Zone 11.

- Fire Hazards Analysis Matrix for Fire Zones 1F and 2C, Comments section was updated to include information regarding FPEE-17-001.
- Fire Hazards Analysis Matrix for Fire Zone 2B, Fire Area Barriers was updated to include Fire Zone 2H North Wall.
- Fire Hazards Analysis Matrix for Fire Zone 2H, Fire Area Barriers was updated to include Fire Zone 2B South Wall and Fire Zone 3B Ceiling.
- Fire Hazards Analysis Matrix for Fire Zone 3B, Fire Area Barriers was updated to include Fire Zone 2H Floor.
- Changed identified fire door for Fire Zone 2C and 30 to Door-210 from Door-67.
- The square footage in the Fire Hazards Analysis Matrix was changed for Fire Zones 7A, 7B, and 7C.
- Fire Hazards Analysis Matrix for Fire Zone 9, Fire Area Barriers was updated to include Fire Zone 19B North Wall
- Changed fire door for Fire Zones 12C and 16 from Door-18A to Door-18.
- Fire Hazards Analysis Matrix for Fire Zone 12C, Fire Area Barriers was updated to include Fire Zone 8 South Wall.
- Fire Hazards Analysis Matrix for Fire Zone 16, Fire Area Barriers ceiling was changed from 18 to 17.
- Fire Hazards Analysis Matrix for Fire Zone 19B, Fire Area Barriers was updated to include Fire Zone 9 South Wall and Fire Zone 20 East Wall
- Fire Hazards Analysis Matrix for Fire Zone 20, Fire Area Barriers was updated to include Fire Zone 19B West Wall
- Fire Hazards Analysis Matrix for Fire Zone 30, Fire Area Barriers was updated to change Fire Zone 37 to 39 (editorial)
- Adding Door-210 to Attachment B of the USAR and removing Door-67.
- Adding FPEE-17-001 to Attachment D of the USAR

3. <u>4 AWI-08.01.00, FIRE PROTECTION PROGRAM PLAN</u>

Revision 20 – PCR 01535369

- 3.0, 4.10.4.A, Figure 5.2 – Changed Nuclear Oversight Manager to NOS Fleet Oversight Manager.
- Section 4.0 – Revised OE from 4.15 to 4.14 and changed review requirements from once per quarter to annual.
- 4.10.4.A, Figure 5.2 – Changed inspection to audits.

- Section 4.15 (old) – Deleted. Described review requirements are already covered by other fire protection program reviews.
- Sections 4.5.3, 4.6.1.C – Deleted subparagraph (d) from 10CFR50.48 reference.

4. B.08.05-05, FIRE PROTECTION - SYSTEM OPERATION, Tables A.2-1, A.2-2, A.2-3 and A.2-4

   Revision 70 – PCR 01558926

   - Table A.2-1 – Bases Section was revised to remove FPEE-16-002 language.
   - Table A.2-3 – Revised to restore fire zone detector functionality criteria as the requirements of FPEE-16-002 was no longer necessary