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Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

July 27, 2005

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10 CFR 54

U. S. Nuclear Regulatory Commission
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Monticello Nuclear Generating Plant
Docket No. 50-263
License No. DPR-22

Response to Request for Additional Information Regarding Severe Accident Mitigation Alternatives for the Monticello Nuclear Generating Plant (TAC NO. MC6441)

References:
1. Letter from NMC to the U. S. Nuclear Regulatory Commission, "Application for Renewal of Operating License," (L-MT-05-014) dated March 16, 2005
2. Letter from NRC to NMC, "Request for Additional Information (RAI) Regarding Severe Accident Mitigation Alternatives (SAMA) for Monticello Nuclear Generating Plant," dated May 27, 2005 (ADAMS Ascension No. ML051470339)

On March 16, 2005, Nuclear Management Company, LLC (NMC) requested the renewal of the operating license for the Monticello Nuclear Generating Plant (MNGP), to extend the terms of their operating license an additional 20 years beyond the current expiration dates.

On May 27, 2005, the NRC issued a request for additional information (RAI) concerning the analysis of Severe Accident Mitigation Alternatives performed in support of the MNGP License Renewal Application. The response to this RAI is enclosed.

I declare, under penalty of perjury, that the foregoing is true and correct.

Executed on 7/27/05.

John T. Conway

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ENCLOSURE 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING THE ANALYSIS OF SAMAs FOR THE MONTICELLO NUCLEAR GENERATING PLANT

Pursuant to 10 CFR 54, the Nuclear Management Company, LLC, (NMC) submitted an application to renew the operating licenses for the Monticello Nuclear Generating Plant (MNGP), dated March 16, 2005. As a result of the NRC's Request for Additional Information (RAI) regarding Severe Accident Mitigation Alternatives (SAMAs) for the MNGP, NMC is hereby providing a response to the NRC's RAI.

<u>NRC Question</u>

1. The SAMA analysis is said to be based on a slight modification (referred to as the SAMA model) of the most recent version of the Monticello probabilistic safety assessment (PSA) (the 2003 model). Provide the following information regarding these PSA models:

    a. Explain why the core damage frequency (CDF) from station blackout (SBO) in Table F.2-1 is only 1.5E-6 per year, when the fire protection system line break in TB-931W, which appears to result in SBO, has a much higher CDF.

    <u>NMC Response</u>

    For the purpose of this analysis, SBO is defined as failure of offsite power and failure of both Emergency Diesel Generators (EDGs). Scenarios that lead to the same effects (power unavailable from offsite or the EDGs) but do not involve EDG failures are not included. Therefore flood induced SBO scenarios are not counted in the value of CDF due to SBO. (In those scenarios, the EDGs operate fine, but power is unavailable to equipment because the electrical buses fed by the EDGs fail because they are under water.)

    The method of evaluating SBO contribution to CDF is creating a gate called CDF-SBO, which is an AND gate of CDF and SBO. SBO is an AND gate of EDG-11 failure to provide power to emergency Bus 15, EDG-12 failure to provide power to emergency Bus 16, and offsite power is unavailable. Failure of Buses 15 and 16 due to flooding or other causes are not captured within this logic.

<u>NRC Question</u>

1. The SAMA analysis is said to be based on a slight modification (referred to as the SAMA model) of the most recent version of the Monticello probabilistic safety assessment (PSA) (the 2003 model). Provide the following information regarding these PSA models:

    b. An April 2002 NMC request for an integrated leak rate test interval extension gives the CDF as 1.57E-05 per year. This is somewhat different from the value quoted for the 1999 update of 1.44E-05 per year. Please explain.

NMC Response

The CDF value reported in the integrated leak rate test interval extension request (1.57×10^{-5} per year) is the sum of frequencies for all accident release categories. Summing the release category frequencies, results in counting scenarios that are nonminimal with respect to core damage. For example "loss of all high pressure injection" and "failure to depressurize" results in core damage. If that same scenario also involves "failure to establish containment heat removal," and "drywell failure," it results in a different consequence category than if the "failure to establish containment heat removal" did not occur. The frequency of this nonminimal scenario is included in the sum of all accident consequence frequencies, but it is not counted when core damage frequency is calculated by evaluating a gate representing core damage frequency without regard to the accident consequence category. As a result, the frequency is reported as a higher value when reported as the sum of all accident consequence bin frequencies than when only core damage frequency is reported.

NRC Question

1. The SAMA analysis is said to be based on a slight modification (referred to as the SAMA model) of the most recent version of the Monticello probabilistic safety assessment (PSA) (the 2003 model). Provide the following information regarding these PSA models:

    c. Clarify whether the importance analyses results given in Tables F.5-1 and F.5-2 are based on the 2003 model or the SAMA model. If the former, confirm that none of the changes made (between the 2003 model and the SAMA model) significantly affect the risk profile or importance ranking. Describe whether any additional SAMAs were suggested by the dose-risk importance list review.

NMC Response

The results from the importance analyses are based on the results from quantification of the SAMA model.

No additional SAMAs were suggested by the review of dose-risk importance rankings associated with the SAMA model MNGP probabilistic risk assessment (PRA).

NRC Question

1. The SAMA analysis is said to be based on a slight modification (referred to as the SAMA model) of the most recent version of the Monticello probabilistic safety assessment (PSA) (the 2003 model). Provide the following information regarding these PSA models:

   d. Describe the evolution of the current Level 2 PSA (including the supporting Modular Accident Analysis Program (MAAP) calculations) relative to that described in the Monticello individual plant examination (IPE). Clarify whether the Level 2 model changes identified in Section F.2.4.3 are the only changes to the Level 2 model.

NMC Response

Calculation files documenting the Level 2 model were not approved at the time the IPE was issued. The model was revisited a few years ago, and those calculation files were reviewed. Resolution of comments generated during the review caused minor changes to the model, although results were not re-quantified and re-characterized to determine the effects. In 2004 MNGP transferred the Level 2 model into the same software used for the Level 1 model (changed from Set Equation Transformation System to Computer Aided Fault Tree Analysis). During the software transfer, the risk model functional decision blocks were unchanged, but the method of evaluation was changed as described below.

Containment event trees still ask the same questions, such as "injection in the reactor pressure vessel (RPV) after core damage but before RPV failure," "containment failure due to hydrogen explosion, steam explosion, liner melt," "containment flooding," etc. However, instead of having containment event trees specific to the accident class, there are trees for: 1) *containment failed*, 2) *containment intact*, 3) *SBO-containment failed*, and 4) *SBO-containment intact*. SBO trees are separated from non-SBO trees to allow credit for recovery of power at various times of interest. Flood induced SBO trees are not processed by the SBO trees because flooded components are assumed to be unrecoverable in the ~2-day time frame modeled in this analysis.

The Level 1 and Level 2 models are now one integrated fault tree model, assembled with 'risk and reliability' software from event trees and fault trees documented in approved calculation files. Detailed studies have not been performed to identify changes resulting from this software transfer. This is largely because modifications to the Level 1 model dramatically impacted results, masking any changes due to those resulting from the software transfer, which are expected to be small.

In preparation for the license renewal application SAMA analysis, ERIN Engineering reviewed MNGP risk analysis models (Level 1 and Level 2). Comments from that review were incorporated into the SAMA model, as summarized in Section F.2.4.3 of the SAMA report.[#]

MAAP cases were regenerated to identify success criteria and characterize radiological source terms. The updated MAAP analyses were performed using an updated version of the software (MAAP 4.0.5 instead of MAAP 3B). The parameter file used by the updated MAAP model is based on the old version but includes updated fuel data, new parameters that were not included in the old version of the model, and corrected values where errors were found. New source term calculations were generated to support accident binning, in support of the SAMA analysis.

NRC Question

1. The SAMA analysis is said to be based on a slight modification (referred to as the SAMA model) of the most recent version of the Monticello probabilistic safety assessment (PSA) (the 2003 model). Provide the following information regarding these PSA models:

    e. Provide a matrix or other documentation that relates the accident classes of Table F.2.1 to the release frequencies of Table F.2.3, and a characterization of the accident sequences that are the dominant contributor to the various release categories. Clarify how the MAAP cases were selected to represent each release category.

NMC Response

Following is a summary of dominant risk scenarios for each accident consequence bin:

Negligible: The most likely *negligible* release scenarios involve high pressure injection failure and operator failure to demand depressurization. This precludes injection, resulting in core melt. After corium penetrates the reactor vessel, it depressurizes, and low pressure injection cools the corium. Containment heat removal is established, and the accident does not produce a significant radiological release to the environment. The most likely scenarios involve a service water break in the reactor building and failure to depressurize before core melt occurs. Service water leaks in the reactor building dominate because they cause failure of all high pressure injection sources (feedwater and control rod drive hydraulic (CRDH) because they depend on service water; and high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) because they fail when a

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[#] The SAMA report is Attachment F of Appendix E – Environmental Report to the MNGP license renewal application.

few inches of water accumulates in the basement of the reactor building). The next most likely scenarios involve SBO where power is recovered some time after the core melts.

Small-Late: The most likely scenarios are the two floods that result in imminent core damage. One is a service water break in the stator cooling room. It fails offsite power and both divisions of emergency power because it accumulates in the Division II 4kV room, causing failure of offsite and Division II emergency power, then flows downstairs and fails Division I emergency power because it floods the 4kV buses. HPCI and RCIC remain available until their batteries deplete. The reactor cannot be depressurized because safety relief valves (SRVs) cannot be manually opened without direct current (DC) power, so the reactor remains pressurized, precluding low pressure injection until after corium penetrates the vessel. Injection from an alternate injection source (fire protection system) then floods containment, reducing consequences. Containment eventually overpressurizes.

The second leak that leads to imminent core melt is a service water leak in the east corridor of the Turbine Building on the 931' elevation. It flows into the plant administrative building and accumulates in the basement, where it causes failure of both divisions of 125V and Division II 250V batteries. This fails feedwater and CRDH because of their dependence on service water, and it fails HPCI and RCIC because they depend on 125V batteries. It also prevents depressurization because manual operation of SRVs depends on the flooded batteries, so the reactor remains pressurized, precluding low pressure injection until after corium penetrates the vessel. Injection from an alternate injection source (fire protection system) then floods containment, reducing consequences. Containment eventually overpressurizes.

Small-Early: This consequence bin is dominated by two types of scenarios. One group of scenarios are those of the negligible consequence bin, where the accident was recovered in vessel, but successful injection after core melt fails to prevent corium from penetrating the reactor vessel. These scenarios lead to core-concrete interactions that generate noncondensable gases and containment flooding, thereby requiring containment venting. The other group of scenarios is anticipated transient without scram (ATWS) scenarios (where containment is assumed to fail) leading to core melt, but all mitigative actions are successful thereafter to effectively mitigate consequences.

Medium-Late: This accident class is dominated by scenario's involving a service water leak in the basement of the turbine building and failure to depressurize the reactor. Containment fails due to lack of containment heat removal, but since failure is in the wetwell airspace, releases are scrubbed, mitigating the consequences.

Medium-Early: Dominant sequences involve service water leaks in the reactor building (fails feedwater and CRDH due to dependence on service water and HPCI and RCIC due to flooding) and failure to depressurize (as in the *negligible* cases above), but the containment liner melts allowing early, significant radiological release to the environment.

Large-Late: Dominant sequences are initiated by a fire protection system leak in the stator cooling room, which fails both divisions of 4kV alternating current (AC) power (prolonged SBO) and alternate injection, which prevents containment flooding.

Large-Early: Dominant sequences are the most likely core melt scenarios (small, late release scenarios, including single event cutsets) with hydrogen explosion or liner melt, causing early containment failure.

Extreme: This consequence bin is defined as loss of coolant accidents (LOCAs) outside containment that lead to core melt. The most likely scenarios involve unisolated line breaks in the turbine building (causing failure of Division I essential motor control centers (MCCs) and the low pressure coolant injection (LPCI) swing bus in addition to feedwater, HPCI, RCIC, and CRDH), coupled with failure of alternate injection. It is noted that the difference between frequencies reported for this accident "Class 5" in Table F.2-1 (8.97×10^{-10} per year) and Release Bin "Extreme" in Table F.2-3 (2.64×10^{-9} per year) is purely due to a change in the level of truncation.

A MAAP case was generated for each group of similar accident sequence (release model event tree end state) to characterize the radiological source term. A review of the resulting source terms was performed, and accident sequences with similar source terms (timing and magnitude) were grouped together, or binned. The bounding source term was used for each accident consequence bin. A calculation file documents the process.

NRC Question

1. The SAMA analysis is said to be based on a slight modification (referred to as the SAMA model) of the most recent version of the Monticello probabilistic safety assessment (PSA) (the 2003 model). Provide the following information regarding these PSA models:

   f. Although it is stated that the SAMA evaluation is based on a modification to the 2003 model (i.e., the SAMA model), the Level 3 results provided in Table F.3-4 are based on the frequency results given in Table F.2-3, which are from the 2003 model. If the SAMA benefit calculations are based on the SAMA model and the baseline frequencies used in the Phase II evaluations are based on the 2003 model, there would appear to be an unaccounted-for difference that could bias the results. Discuss how the results of the SAMA model were incorporated into the SAMA evaluation, and the impact of any unaccounted-for differences.

NMC Response

SAMA tables F.2-1 and F.2-3 are labeled incorrectly. Values in these tables are based on results from the SAMA model instead of the 2003 model that is referred to in the parenthesizes in each of the table titles.

NRC Question

2. Provide the following with regard to the treatment of external events in the SAMA analysis:

   a. The contribution to fire risk for the cable spreading room is given in the environmental report as 11.5 percent. This corresponds to a CDF of 8.98E-07 per year using the total fire CDF of 7.81E-06 per year as quoted in the U.S. Nuclear Regulatory Commission's IPE for external events (IPEEE) safety evaluation report (SER). This is different from the cable spreading room fire CDF of 1.45E-06 per year given in the SER. Provide a brief explanation for this apparent discrepancy.

NMC Response

The frequency of core damage initiated by a fire in the cable spreading room is estimated in the IPEEE to be 1.45×10^{-6} per year (in Table B.2.11.1), and the frequency of core damage initiated by any fire is estimated to be 7.81×10^{-6} per year. This corresponds to 18.6% of fire initiated CDF. However, Figure 1.4-2 shows the contribution of fire to be 11.5% instead of 18.6%. Summing the frequencies from IPEEE Table B.2.11.1 gives 8.36×10^{-6} per year instead of 7.81×10^{-6} per year. This very well could be from incorrectly assessing CDF due to cable spreading room fires as 1.45×10^{-6} per year instead of 8.98×10^{-7} per year (corresponding to 11.5%). If the value in Table B.2.11.1 is changed from 1.45×10^{-6} per year to 8.98×10^{-7} per year, the total changes from 8.36×10^{-6} per year to 7.81×10^{-6} per year, which is consistent with the rest of the data presented. Therefore, it appears that IPEEE Figure 1.4-2 and the Environmental Report (ER) are correct, and the SER and IPEEE Table B.2.11.1 have this value incorrectly reported.

NRC Question

2. Provide the following with regard to the treatment of external events in the SAMA analysis:

   b. The Monticello IPEEE submittal for internal fire identifies three insights or improvements that were to be considered further: (1) two out of three success criteria for the service water pumps, (2) credit for control rod drive injection after bypassing the load shed logic, and (3) elimination of the AC power dependency on the safety/relief valves. It is stated that the last has been implemented. Confirm this and provide the status of the other two items.

NMC Response

Two of the three recommendations from Revision 0 of the IPEEE are not actually plant modifications meant to reduce actual risk imposed by plant operations but rather PRA modeling changes recommended so that the PRA model better reflects actual risk. The first recommendation is to determine if service water is functional with only one pump operating, and if so, change the success criteria from two to one of three pumps (PRA model change that does not change actual risk). Those studies were performed, and a single service water pump was found to be sufficient for the system to be functional. The current SAMA model reflects this.

The second recommendation is to revise emergency operating procedures to direct operators to bypass load shed logic, allowing control rod drive injection if offsite power is lost and emergency diesel generators function properly. The emergency operating procedures identify control rod drive (CRD) system as a primary source of injection and direct operators to maximize flow to the reactor if needed. Abnormal operating procedures address bypassing load shed logic if necessary. The current model credits these actions.

The PRA model that Revision 0 of the IPEEE is based on modeled SRVs with dependence on AC power for depressurization. The third recommendation is to revise the model to reflect recent modifications implemented to eliminate this AC power dependence (PRA model change that does not change actual risk). The SRVs depend on power from batteries, so they are available for depressurization during SBO unless the batteries deplete. The PRA model reflects this configuration. Note that SAMA 2 (one of the modifications in the recommended set of SAMAs) enhances battery charger reliability by providing alternate power to battery chargers, improving reliability of SRVs for prolonged SBO scenarios.

Revision 1 of the IPEEE also recommends two PRA model changes that do not affect actual risk. The first recommendation is to credit manual fire suppression in areas other than the main control room. A draft fire risk analysis model developed by NMC for the MNGP uses industry data for estimating the frequency of fires that propagate to more than one component, so it inherently credits manual suppression by assuming that manual suppression reliability at MNGP is similar to the industry.

The second recommendation in Revision 1 of the IPEEE is to credit CRD and the main condenser for fires that do not cause their failure. The current draft fire risk analysis model implements this recommendation.

NRC Question

3. Provide the following information concerning the MACCS (MELCOR Accident Consequences Code System Version 2) analyses:

   a. The MACCS2 analysis for Monticello is based on a reference boiling-water reactor (BWR) core inventory at end-of-cycle, scaled by the power level for Monticello. The calculations were based on a 3-year fuel cycle (12-month reload) with an average power density for the assembly groups ranging from 24 to 30 MW/MTU. Current BWR fuel management practices use longer fuel cycles (time between refueling) and result in significantly higher fuel burnup. The use of the reference BWR core instead of a plant-specific cycle could significantly underestimate the inventory of long-lived radionuclides important to population dose (such as Sr-90, Cs-134, and Cs-137), and thus impact the SAMA evaluation. Justify the adequacy of the SAMA identification and screening given the fuel enrichment and burnup expected at Monticello during the renewal period.

NMC Response

Constellation Energy has previously responded to a similar RAI for the Nine Mile Point SAMA evaluation. In the response, plant-specific fission product inventories, produced by General Electric, were obtained for evaluating the impact on the overall SAMA conclusions. The data obtained was for end-of-cycle activity levels for a bounding case of 1,400 effective full power days with an average 4.1 percent enrichment. The Nine Mile Point evaluation noted that the activity levels for Sr-90, Cs-134 and Cs-137 increased in the range of 60 to 73 percent over the reference BWR inventories.

An additional MNGP-specific MACCS2 sensitivity calculation was performed assuming an increase in the inventories for Sr-90, Cs-134 and Cs-137 equal to 65%. This was based on the average increase for these same radionuclides as evaluated in the Nine Mile Point RAI response. The revised MACCS2 results show that the average dose increased by 16.4% and that the economic cost risk increased by 29.0%. Using the updated release category specific MACCS2 results for this sensitivity, each of the Phase II SAMAs was recalculated to obtain the revised benefit cost. The following table provides a comparison between the base case and sensitivity case averted cost risk for each of the Phase II SAMAs. Based on the revised radionuclide inventories, SAMA 39 is marginally cost beneficial. It should be noted that this SAMA was previously identified as cost beneficial in both the 3% discount rate and 95% sensitivity studies.

SAMA ID	Averted Cost-Risk (Base Case)	Averted Cost-Risk (Radionuclide Sensitivity)	Cost of Implementation	Change in Cost Beneficial Status from Base Case?
2	$79,191	$97,530	$75,000	No
4	$8,457,131	$10,276,933	$2,000,000	No
6	$102,790	$121,718	$100,000	No
8	$211,458	$257,299	$2,000,000	No
10	$271,594	$343,036	$760,000	No
11	$687,044	$860,845	$50,000	No
12	$2,611,782	$3,274,572	$50,000	No
13	$3,505	$3,522	$50,000	No
16	$279,480	$344,479	$200,000	No
28	$1,332	$1,568	$50,000	No
36	$1,899,615	$2,349,399	$100,000	No
37	-$5,581,445	-$7,149,942	$100,000	No
38	$330,557	$401,549	$10,000,000	No
39	$752,567	$914,192	$786,991	Yes
40	$178,243	$216,524	$230,000	No
Recommended	$6,988,166	$8,465,024	$425,000	No

NRC Question

3. Provide the following information concerning the MACCS (MELCOR Accident Consequences Code System Version 2) analyses:

   b. The MACCS analysis assumes all releases occur at ground level and have a thermal content the same as ambient. These assumptions could be non-conservative when estimating offsite consequences. Provide an assessment of the impact that alternative assumptions might have on the estimated offsite consequences (doses to the population within 50 miles) and the conclusions of the SAMA evaluation.

NMC Response

The MACCS2 analysis for the MNGP assumes all releases occur at ground level and have a thermal content same as ambient. A sensitivity study was performed to assess the impact of alternative assumptions where the release height is conservatively taken as the top of the reactor building (42.7 meters) and the plume thermal content is assumed to be 1.0E+7 watts. The alternative assumptions resulted in an increase in population dose of 7.6% and an increase in offsite economic cost of 9.3% over the base case results. The results of this sensitivity are bounded by the case presented in Response 3a.

NRC Question

3. Provide the following information concerning the MACCS (MELCOR Accident Consequences Code System Version 2) analyses:

   c. Annual meteorology data from the year 2000 were used in the MACCS2 analyses. Provide a brief statement regarding the acceptability of use of this year's data rather than a different year's data.

NMC Response

The MACCS2 analysis for the MNGP used meteorological data from the year 2000. Year 2000 data is representative of 5 year meteorological data previously tabulated for the alternate source term project. Previous MACCS2 sensitivity analyses for SAMA reports have shown that typical variations in the meteorological data can result in a +/- 5% change in population dose and offsite economic cost. This minimum change is judged acceptable and is encompassed in the 3% discount rate and 95% CDF sensitivity analyses that are included in the SAMA evaluation.

NRC Question

3. Provide the following information concerning the MACCS (MELCOR Accident Consequences Code System Version 2) analyses:

   d. The off-site economic cost risk at Monticello is larger than estimated at other sites having similar CDF and population dose. Identify and briefly discuss the key MACCS2 input assumptions or other factors that may contribute to this larger value at Monticello, e.g., per diem cost for relocated individuals, the costs to relocate an individual, and the value of farm and non-farm wealth.

NMC Response

The MACCS2 analysis for the MNGP utilized monetary cost values based on NUREG-1150, updated by the consumer price index, and SECPOP2000 (NUREG/CR-6525) economic data. The information taken from these sources includes the per diem cost for relocated individuals, the cost to relocate an individual, and the value of farm and non-farm wealth. The parameters for the MNGP, as demonstrated in the table below, are consistent with the example MACCS2 input provided in Table 5 of the draft Nuclear Energy Institute (NEI) SAMA Guidance document.

MACCS2 ECONOMIC PARAMETER COMPARISON

Variable	Description	NEI Value	MNGP Value
DPRATE	Property depreciation rate (per yr)	0.2	0.2
DSRATE	Investment rate of return (per yr)	0.12	0.12
EVACST	Daily cost for a person who has been evacuated ($/person-day)	43.05	41.58
POPCST	Population relocation cost ($/person)	7967.12	7700.00
RELCST	Daily cost for a person who is relocated ($/person-day)	43.05	41.58
CDFRM0	Cost of farm decontamination for various levels of decontamination ($/hectare)	896.59 1992.49	866.25 1925.00
CDNFRM	Cost of non-farm decontamination per resident person for various levels of decontamination ($/person)	4781.42 12754.28	4630.00 12320.00
DLBCST	Average cost of decontamination labor ($/man-year)	55792.80	53900.00
VALWF0	Value of farm wealth ($/hectare)	4547.23	4702.00
VALWNF	Value of non-farm wealth ($/person)	126107.80	90551.00

The only major difference between the MNGP input and the NEI example input is in the non-farm wealth value. The MNGP value is about 28 percent less than the example data, which would trend the MNGP economic cost lower than the NEI example. Based on a review of this data and other input, the estimated economic cost for MNGP is considered to be consistent with the characteristics of the 50-mile population for the site.

A review of the MNGP population dose calculation has also shown the results to be consistent with the site's characteristics. On first inspection, it would appear that the population dose calculated for MNGP is low compared with similar calculations for other sites. However, further investigation shows that the lower population dose is governed by the characteristics of the surrounding watershed and the corresponding mechanisms for ingesting radionuclides.

Based on the methodology used by MACCS2 to analyze water ingestion from fallout and runoff into water, large bodies of fresh water that are used as drinking supplies are assigned higher ingestion factors than river systems used for the same purpose. All of the bodies of water that provide a drinking water pathway in the 50 mile area surrounding the MNGP are classified as river systems based on NUREG/CR-4551. As demonstrated in the table below, the MACCS2 analysis for the MNGP resulted in a water ingestion dose that was only 2.8% of the total population dose.

MONTICELLO NUCLEAR GENERATING PLANT WATER INGESTION DOSE

Source Term	Pop Dose (Sv)	Water Ingestion Dose	Total Costs ($)	Frequency (per yr)	Wtd Dose Risk (per-rem/yr)	Wtd Water Ingestion Dose	Wtd Cost Risk ($/yr)	% Dose due to Water Ingestion
E-E	5.25E+04	2.70E+03	2.29E+10	2.64E-09	1.4E-02	7.1E-04	6.05E+01	5.1%
L-E	4.11E+04	1.23E+03	3.13E+10	4.20E-06	1.7E+01	5.2E-01	1.31E+05	3.0%
L-L	2.20E+04	7.45E+02	1.63E+10	7.19E-06	1.6E+01	5.4E-01	1.17E+05	3.4%
M-E	2.86E+04	6.75E+02	1.99E+10	8.99E-08	2.6E-01	6.1E-03	1.79E+03	2.4%
M-L	4.06E+03	2.25E+01	4.25E+08	1.09E-06	4.4E-01	2.5E-03	4.63E+02	0.6%
S-E	3.87E+03	1.42E+01	3.23E+08	1.81E-07	7.0E-02	2.6E-04	5.85E+01	0.4%
S-L	1.04E+03	3.41E+00	8.17E+07	3.97E-05	4.1E+00	1.4E-02	3.24E+03	0.3%
Frequency Weighted Totals (p-rem and $)					3.8E+01	1.1E+00	2.54E+05	2.8%

Sites that classify a large body of water (e.g., one of the great lakes) in the 50-mile surrounding area as a lake system and utilize recommended water ingestion factors from NUREG/CR-4551 may have a significantly larger population dose from water ingestion.

While the methods used to develop these results for the MNGP are consistent with NUREG/CR-4551, they are not necessarily consistent with all industry guidance. For example, the water body classification system applied in the NUREG-1437 analysis for the determination of water ingestion dose is more detailed than what is used in NUREG/CR-4551. The water ingestion doses factors suggested for all river systems in NUREG/CR-4551 are based on the dynamics of the Mississippi river while NUREG-1437 distinguishes between rivers based on their size. NUREG/CR-4551's use of a large river as the water ingestion model for all river systems may result in under predicting the water ingestion dose for small river sites.

The more detailed river classification system used in NUREG-1437 resulted in the identification of the MNGP as a site where the water ingestion doses could be higher than for other sites located near larger, more dynamic bodies of water. While the water ingestion dose for most U.S. nuclear sites were considered as bounded by the results for the Fermi site, it was determined that thirteen sites, including the MNGP, could be outliers for the following reasons:

(1) low on-site average annual flow rates

(2) comparatively long residence times

(3) comparatively large surface-area-to-volume ratios.

However, NUREG-1437 identifies some mitigating factors related to these 13 sites that limit the impact of the small river characteristics. For example, the combined residence time and surface-area-to-volume ratios for the 13 small river sites exceed values at the Fermi site by less than a factor of 3 while these sites have populations lower than Fermi by at least a factor of 2. The increase in dose expected from the larger residence times and surface-area-to-volume ratios are counterbalanced by the lower population for those 13 sites. NUREG-1437 concluded that the population dose at the 13 potential outlier sites was expected to remain a small fraction of the value estimated for the atmospheric pathway. As a point of reference, the uninterdicted dose from fallout onto open bodies of water for the Fermi site was estimated to be less than 2% of that from the atmospheric pathway total. This is consistent with the results from the MNGP SAMA analysis.

In summary, the economic value parameters used in the MNGP MACCS2 model are consistent with the industry guidance and produce results that are considered to be appropriate for the MNGP site. The water ingestion dose model, while consistent with the NUREG/CR-4551 guidance that is typically used in industry SAMA analyses, is simplified compared with what has been developed in NUREG-1437. The NUREG/CR-4551 methodology could under predict water ingestion doses for small river sites; however, mitigating factors exist such that the MNGP results are consistent with what is predicted by NUREG-1437 for a small river site.

NRC Question

4. Provide the following with regard to the SAMA identification and screening processes:

   a. The IPE identified a number of plant, procedure, and training modifications that have either been implemented or were under consideration. It is stated in Section F.5.1.4 that all of these have been considered in the Phase 1 SAMA list. Please indicate which SAMAs include:

      i. modification to assure faster operation of condensate demineralizer, bypass valve on loss of air (stated as completed in the IPE)

NMC Response

The IPE states that the valve was modified to ensure that the condensate demineralizer bypass valve opens more rapidly on a loss of instrument air. Because the modification has already been implemented, the SAMAs do not include this modification. However, additional reviews indicate that there is little consensus that the modified system would prevent condensate / feedwater failure upon loss of instrument air. Therefore, the risk analysis model never credited the modification (still assumes dependency on instrument air), and the recommendation should have been considered as a SAMA.

Quantification of the current risk analysis model (which incorporates the recommended modifications) with and without the dependency on instrument air shows an extremely small risk reduction. CDF is reduced by about 4×10^{-8} per year, and the reduction occurs almost entirely in negligible and small, late release scenarios. The value of the modification is estimated to provide a 25-year risk reduction worth of less than $2,000.

NRC Question

4. Provide the following with regard to the SAMA identification and screening processes:

   a. The IPE identified a number of plant, procedure, and training modifications that have either been implemented or were under consideration. It is stated in Section F.5.1.4 that all of these have been considered in the Phase 1 SAMA list. Please indicate which SAMAs include:

      ii. operator training on recovery of the failed residual heat removal (RHR) (stated as being considered in the IPE), and

NMC Response

Operators have been trained to manually operate equipment in the field (when feasible) if control from the control room is unavailable. Risk dominant failure of RHR involves valve failures due to loss of control power. The risk analysis model incorporated this recovery because it has been incorporated into operator training and is significant.

NRC Question

4. Provide the following with regard to the SAMA identification and screening processes:

   a. The IPE identified a number of plant, procedure, and training modifications that have either been implemented or were under consideration. It is stated in Section F.5.1.4 that all of these have been considered in the Phase 1 SAMA list. Please indicate which SAMAs include:

      iii. testing of the boron injection hose (stated as under consideration in the IPE).

NMC Response

The IPE stated that alternate boron injection hoses should be periodically tested, but it also noted that alternate boron injection has a very small impact on CDF. This recommendation was subsumed by SAMA 13, which was identified based on the current Level 2 model importance list. This SAMA includes what were considered to be more effective means of addressing alternate boron injection reliability and replaced the fire hose testing enhancement identified in the IPE. A review of the standby liquid control system shows that crediting alternate boron injection using CRDH pumps with perfect reliability has no distinguishable impact on core damage frequency. Likewise, eliminating all credit for this function has a negligible increase CDF (undetectable at 1E-10 per year truncation). Therefore, improving the reliability of a backup to this system certainly is not risk significant.

NRC Question

4. Provide the following with regard to the SAMA identification and screening processes:

   b. SAMA 36, Divert Water from Turbine Building 931-foot Elevation East, is indicated in Table F.5-1 to be applicable for the largest CDF contributor, IEF-FS-TB931W, which involves a flood in the turbine building 931-foot elevation east area. Clarify whether this SAMA will mitigate a flood in the west area or the east area of the 931-foot elevation. Discuss whether there could be similar SAMAs for the other floods (e.g., IEF-SW-TB911, IEF-SW-RHR1, IEF-SW-RHR2, IEF-SW-RB896).

NMC Response

The table summarizes the flood mitigating modifications recommended for implementation. SAMA 36 is really only applicable to floods in the Turbine Building 931' elevation east corridor, but is included in the list because it is part of the recommended flood mitigation package. Diversion of flood sources to safe destinations was considered for each flood area, but the only area where that proved to be reasonable was the Turbine Building 931' elevation east corridor.

NRC Question

4. Provide the following with regard to the SAMA identification and screening processes:

   c. SAMA 7, Rupture Disk Bypass Line, is indicated in Table F.5-3 to be subsumed by SAMA 16. Based on Table F.5-1, SAMA 7 addresses event MVR4543XXN, which is failure of the rupture disk to open. SAMA 16 appears to involve a change in vent valves so that they "fail open" on loss of support, while maintaining the rupture disk. Explain how SAMA 16 reduces the importance of the rupture disk failing to open, and why it is a reasonable alternative to SAMA 7.

NMC Response

It is correct that SAMA 16 does not directly address the rupture disk failure, but SAMA 16 has been chosen as the best method for treating containment vent reliability, given that it is a low cost alternative which impacts the largest failure mode for venting. (The vast majority of hardpipe vent unavailability is from the fact that the vast majority of containment heat removal scenarios (scenarios where containment venting is needed) involve prolonged SBO, so solenoid valves are unable to change to the energized position to apply pneumatic pressure and open the valves.)

SAMA 16 intends to capture the fact that improving hardpipe vent performance would have a significant safety benefit, so it is being maintained as a possible issue to pursue after the recommended modifications (SAMAs 2, 11, 12, 28, 36, & 37) are implemented. Since the time when the SAMA analysis was performed, these recommended modifications have been implemented, and the value of modifying the hardpipe vent design has been re-evaluated and found to still be significant. Modifications were made to allow credit for manual, local operation of the hardpipe vent valves, but further improvement of hardpipe vent system reliability would still provide significant safety benefit. The improvement is being pursued to determine if cost effective modifications can be implemented. Because this improvement may be applicable for all BWRs, a white paper has been submitted for the BWR Owners Group review.

NRC Question

4. Provide the following with regard to the SAMA identification and screening processes:

   d. In Table F.5-1, events ASMY83XXXL and ASMY85XXXL each have risk reduction worth of 1.005. Explain why no SAMAs have been identified for these events.

   NMC Response

   The components are a manual bypass switch and a manual disconnect switch that support operation of instrument AC panel Y80. Y80 has a very high importance because its failure precludes containment venting (all three vent paths), fails Division II containment heat removal (residual heat removal (RHR) / residual heat removal service water (RHRSW) control valve CV-1729), trips the mechanical vacuum pump, and fails HPCI.

   If either Y83 or Y85 trips open, Y80 fails, and this makes these switches very important. However, manual switches have extremely high reliability (it is exceedingly rare that a manual switch spuriously opens). Because the consequences of failure are extremely high, but the probability of failure is very low, the risk reduction importance is marginally high (1.005 is the threshold at which components are considered for SAMA). The marginal risk reduction importance means that it can only prove to be cost effective if an inexpensive modification can mitigate the issue.

   No cost effective means was identified for mitigating this issue. However, implementation of recommended modifications (SAMAs 2, 11, 12, 28, 36, & 37) have dramatically decreased the importance of these switches. Risk reduction worth is now less than 1.005 for each of these switches, so the issue is not being pursued.

NRC Question

5. Provide the following with regard to the Phase II cost-benefit evaluations:

   a. The discussion of SAMA 12 in Section F.6.7 indicates that the benefit of the SAMA (which involves implementation of a procedure to direct the pressurization of the fire service water system using a fire truck) was estimated by assuming an improved diesel fire pump in addition to the use of a fire pumper truck. It is not clear why an improved fire pump was credited, and this credit appears inconsistent with the cost estimate. Please explain.

NMC Response

The reported probability for diesel fire pump failing to run (event YPDP105XXR) was more than 10%, based on industry data available. However, the MNGP diesel fire pump is operated weekly and has experienced very few failures in its history, so this value seems to be exceedingly high. No physical changes to the pump were expected, but an improved reliability was used as part of the SAMA to better reflect reality. The effect of this change is to somewhat overstate the value of the SAMA, which is conservative because it may cause the plant to implement a safety beneficial modification that would otherwise have been less beneficial than stated.

Improvement of the fire protection system reliability during SBO has a dramatic effect on safety (if done in concert with other recommended SAMAs that enable the reactor to remain depressurized during prolonged SBO scenarios), and the cost of improving system reliability is very low. Therefore this SAMA would be very cost beneficial, regardless of whether the diesel fire pump reliability were left at its old value or changed to a more realistic value. SAMA 12 was implemented in 2005.

NRC Question

5. Provide the following with regard to the Phase II cost-benefit evaluations:

   b. Discuss in more detail the reasons why SAMA 37 results in an 81.7 percent increase in dose risk if implemented individually, whereas when implemented along with the other five recommended SAMAs, there is a 79.6 percent net reduction in dose risk. Also discuss the rationale for including SAMA 28 within the set of recommended SAMAs given the negative net value for this SAMA if implemented individually, and the rationale for excluding SAMA 16 from the set of recommended SAMAs given the positive net value for this SAMA if implemented individually.

NMC Response

Manual operation of RCIC alone does not represent a success path for dominant risk scenarios (prolonged SBO). In fact, it only holds off the inevitable because containment heat removal and containment venting are both unavailable in prolonged SBO scenarios. This eventually results in containment rupture due to overpressure, precluding operators from occupying the RCIC room to support manual operation of the system. In addition, RCIC would eventually deplete condensate storage tank (CST) inventory and would overheat if torus water (saturated conditions) were used as the suction source.

For these reasons, RCIC only shifts risk dominant sequences from leading to core melt in 6 hours (2 hours after batteries deplete) to 20 hours (a few hours after containment ruptures or CST inventory is depleted). This has competing effects on risk. The delay allows an increased probability of recovering offsite power or power from an EDG, which has the effect of reducing risk. At the same time, it also increases risk because the additional time allows containment to heat up and pressurize, so that it ruptures when corium penetrates the reactor pressure vessel, allowing a direct release of unscrubbed fission products to be released to the environment.

The risk reduction gained by allowing an increased probability of power recovery is negligibly small because prolonged SBO scenarios are dominated by flood induced causes (both divisions of AC power fail due to flooding), and these would not be recovered within this time frame (less than a day). The increase in risk experienced by implementing this modification alone reflects the significant increase in consequences of failing a pressurized reactor vessel into a pressurized containment at saturated conditions.

SAMA 28 is essential to include in the recommended group of SAMAs because it is necessary for RCIC to remain a viable long-term injection source. SAMA 16 is excluded from the recommended SAMAs because it causes issues associated with containment isolation. This issue is being investigated outside of the license renewal process. See response to RAI question 6 for additional details on this subject.

NRC Question

5. Provide the following with regard to the Phase II cost-benefit evaluations:

   c. While SAMA 40 was specifically identified from the results of the fire IPEEE and the analysis of the benefit for reducing fire risk appears conservative, it would appear that this SAMA would also impact the risk from internal events. It is noted, however, that the lack of hot well inventory is not in the importance lists for Tables F.5-1 and F.5-2. This may be due to assumptions of the PSA model rather than an indication of the low importance. Provide an assessment of the potential impact of this SAMA on internal events, and if appropriate, a revised cost-benefit analysis for this SAMA that includes consideration of the additional risk reduction and averted cost risk for internal events.

NMC Response

The MNGP hotwell has automatic makeup capability from the CSTs and is credited as such in the risk analysis model (see fault trees gates F002 and F003). Hotwell makeup is shown to have very little importance, so requantification is not necessary. However, operator action to manually align makeup to the hotwell during fire scenarios was identified as important due to the unavailability of that system to perform its function. While details are not available related to all of the important contributors to failure of the makeup system, a fire in the Turbine Building 931' area (fire zones XII/17, 19A, and 19B) results in loss of the control of the makeup valves (11% of the Class II CDF for fire events). Overall, the operator failure to provide hotwell makeup contributed to 23% of the Class II fire CDF, which is accounted for in the ER evaluation. SAMA 40 was intended to provide a diverse, emergency backup to the normal makeup system.

NRC Question

6. In Section F.6.16.2, NMC provides the net values for the remaining SAMAs given implementation of the six recommended SAMAs (see table on page F-68). These values are based on a 7 percent real discount rate. Provide the net value results based on a 3 percent real discount rate. Discuss how inclusion of SAMA 16 within the set of recommended SAMAs would impact these values.

NMC Response

The assumption of 7% real discount rate only has a modest effect on the estimated value of implementing a modification. The majority of SAMAs are grossly more expensive than the benefit they offer, given that the proposed modifications are implemented.

The benefit is significantly less than the cost for all of the SAMAs except SAMA 16, which shows a much higher benefit than implementation cost. Therefore, re-assessing all cases with a 3% real discount rate would produce minimal benefits. The SAMA closest to being cost beneficial is SAMA 4. If a 3% real discount rate is assumed, the benefit increases, but is still less than the $2 M estimated cost of implementation.

Implementation of SAMA 16 would clearly reduce the benefit of each of the remaining SAMAs. It reduces overall plant risk by reducing the likelihood of core melt caused by containment pressurization and reduces the consequences of core melt scenarios by improving the likelihood of release scrubbing. Furthermore, it reduces unquantified consequences because it enables simple, effective containment recovery, whereas it may take weeks to effectively terminate the release from a ruptured containment. Since there are no synergistic effects between this SAMA and the other remaining SAMAs, the effect of its implementation only reduces their cost effectiveness.

The primary goals of the MNGP risk analysis group include identification of cost effective means of minimizing risk posed by plant operations. The changes recommended in the SAMA analysis were recommended for implementation prior to initiation of the SAMA analysis. The risk analysis group is currently working with the industry to pursue SAMA 16 and will continue to actively search for additional cost effective means of mitigating risk.