Appendix F

GEIS Environmental Issues Not Applicable to Monticello Nuclear Generating Plant

Appendix F: GEIS Environmental Issues Not Applicable to Monticello Nuclear Generating Plant

Table F-1 lists those environmental issues listed in the *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* (GEIS), NUREG-1437, Volumes 1 and 2 (NRC 1996, 1999)[a] and 10 CFR Part 51, Subpart A, Appendix B, Table B-1, that are not applicable to Monticello because of plant or site characteristics.

Table F-1. GEIS Environmental Issues Not Applicable to Monticello

ISSUE—10 CFR Part 51, Subpart A, Appendix B, Table B-1	Category	GEIS Sections	Comment
SURFACE WATER QUALITY, HYDROLOGY, AND USE (FOR ALL PLANTS)			
Altered salinity gradients	1	4.2.1.2.2; 4.4.2.2	Monticello cooling system does not discharge to an estuary.
Altered thermal stratification of lakes	1	4.2.1.2.2; 4.4.2.2	Monticello cooling system does not discharge into a lake.
GROUNDWATER USE AND QUALITY			
Groundwater use conflicts (potable and service water, and dewatering; plants that use >100 gpm)	2	4.8.1.1; 4.8.1.2	Monticello does not use more than 100 gpm groundwater.
Groundwater-use conflicts (Ranney wells)	2	4.8.1.4	Monticello does not have or use Ranney wells.
Groundwater quality degradation (Ranney wells)	1	4.8.2.2	Monticello does not have or use Ranney wells.
Groundwater quality degradation (saltwater intrusion)	1	4.8.2.1	Monticello does not discharge to saltwater.
Groundwater quality degradation (cooling ponds in salt marshes)	1	4.8.3	Monticello does not have or use cooling ponds.
Groundwater quality degradation (cooling ponds at inland sites)	2	4.4.4	Monticello does not have or use cooling ponds.
TERRESTRIAL RESOURCES			
Cooling pond impacts on terrestrial resources	1	4.4.4	Monticello does not have or use cooling ponds.

[a] The GEIS was originally issued in 1996. Addendum 1 to the GEIS was issued in 1999. Hereafter, all references to the "GEIS" include the GEIS and its Addendum 1.

References

10 CFR Part 51. Code of Federal Regulations, Title 10, *Energy*, Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions."

U.S. Nuclear Regulatory Commission (NRC). 1996. *Generic Environmental Impact Statement for License Renewal of Nuclear Plants.* NUREG-1437, Volumes 1 and 2, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1999. *Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Main Report,* Section 6.3 – Transportation, Table 9.1, Summary of findings on NEPA issues for license renewal of nuclear power plants, Final Report." NUREG-1437, Volume 1, Addendum 1, Washington, D.C.

Appendix G

NRC Staff Evaluation of Severe Accident Mitigation Alternatives (SAMAs) for Monticello Nuclear Generating Plant

Appendix G: NRC Staff Evaluation of Severe Accident Mitigation Alternatives (SAMAs) for Monticello Nuclear Generating Plant

G.1 Introduction

Nuclear Management Company, LLC (NMC) submitted an assessment of severe accident mitigation alternatives (SAMAs) for Monticello Nuclear Generating Plant (Monticello) as part of the environmental report (ER) (NMC 2005a). This assessment was based on the most recent Monticello probabilistic safety assessment (PSA) available at that time, a plant-specific offsite consequence analysis performed using the MELCOR Accident Consequence Code System 2 (MACCS2) computer code, and insights from the Monticello individual plant examination (IPE) (NSP 1992) and individual plant examination of external events (IPEEE) (NSP 1995a,b). In identifying and evaluating potential SAMAs, NMC considered SAMAs that addressed the major contributors to core damage frequency (CDF) and population dose at Monticello, as well as SAMA candidates for other operating plants which have submitted license renewal applications. NMC identified 40 potential SAMA candidates. This list was reduced to 16 unique SAMA candidates by eliminating SAMAs that: are not applicable to Monticello due to design differences, are of low benefit in boiling water reactors, have already been implemented at Monticello or whose benefit has been achieved at Monticello using other means, or have estimated costs that would exceed the dollar value associated with completely eliminating all severe accident risk at Monticello. NMC assessed the costs and benefits associated with each of the potential SAMAs and concluded in the ER that several of the candidate SAMAs evaluated may be cost-beneficial and warrant further review for potential implementation.

Based on a review of the SAMA assessment, the U.S. Nuclear Regulatory Commission (NRC) issued a request for additional information (RAI) to NMC by letter dated May 27, 2005 (NRC 2005). Key questions concerned: changes to the Level 2 PSA model and source terms since the IPE; MACCS2 input data (core inventory, releases, meteorology data, and offsite economic costs); further information on several specific candidate SAMAs; additional information/clarification regarding SAMAs related to external events; and the rationale used by NMC to arrive at a set of "recommended" SAMAs for further evaluation. NMC submitted additional information by letter dated July 27, 2005 (NMC 2005b). In the response, NMC provided: a description of the current Level 2 model and dominant risk scenarios for each accident consequence bin; results of sensitivity studies related to radionuclide inventories, release heights and thermal content of the plume; rationale for seemingly larger offsite economic cost risk at Monticello; specific requested information for SAMAs related to external events; and the rationale used to arrive at the set of "recommended" SAMAs. NMC's responses addressed the staff's concerns.

An assessment of SAMAs for Monticello is presented below.

G.2 Estimate of Risk for Monticello Nuclear Generating Plant

NMC's estimates of offsite risk at the Monticello are summarized in Section G.2.1. The summary is followed by the staff's review of NMC's risk estimates in Section G.2.2.

G.2.1 Nuclear Management Company, LLC's Risk Estimates

Two distinct analyses are combined to form the basis for the risk estimates used in the SAMA analysis: (1) the Monticello Level 1 and 2 PSA model, which is an updated version of the IPE (NSP 1992), and (2) a supplemental analysis of offsite consequences and economic impacts (essentially a Level 3 PSA model) developed specifically for the SAMA analysis. The SAMA analysis is based on a slight modification of the 2003 Monticello Level 1 and 2 PSA model, referred to as the SAMA model. The scope of the Monticello PSA does not include external events.

The baseline CDF for the purpose of the SAMA evaluation is approximately 4.5×10^{-5} per year. The CDF is based on the risk assessment for internally-initiated events at extended power uprate conditions. NMC did not include the contribution from external events within the Monticello risk estimates; however, it did account for the potential risk reduction benefits associated with external events by doubling the estimated benefits for internal events. This is discussed further in Section G.6.2.

The breakdown of CDF by initiating event is provided in Table G-1. As shown in this table, events initiated by internal floods are the dominant contributors to CDF. Station blackout (SBO) sequences contribute 1.52×10^{-6} per year (about 3 percent of the total internal events CDF), while anticipated transient without scram sequences are insignificant contributors to CDF (8.24×10^{-8} per year). NMC defined SBO as loss of offsite power and both emergency diesel generators. This definition excludes the SBO-like conditions resulting from flooding-induced loss of electrical buses which are large contributors to the internal flooding CDF. When the contribution from flooding events is also included, events resulting in SBO-like conditions account for the majority of the CDF.

Table G-1. Monticello Core Damage Frequency

Initiating Event	CDF (per year)	% Contribution to CDF
Fire protection system line break in turbine building (TB) 931-ft elevation west	3.2×10^{-5}	71
Service water (SW) line break in TB 931-ft elevation east	5.8×10^{-6}	13
SW line break in TB 911-ft elevation	1.8×10^{-6}	4
Loss of offsite power	1.8×10^{-6}	4
SW line break in residual heat removal (RHR) A room	8.9×10^{-7}	2
SW line break in residual heat removal (RHR) B room	8.9×10^{-7}	2
SW line break reactor building 896-ft elevation	4.5×10^{-7}	1
Turbine trip	4.5×10^{-7}	1
Loss of feedwater	4.5×10^{-7}	1
Other	4.5×10^{-7}	1
Total CDF (from internal events)	**4.5×10^{-5}**	**100**

The Level 2 Monticello PSA model that forms the basis for the SAMA evaluation represents an adaptation and updating of the IPE Level 2 model. The IPE Level 2 model involved the development of containment event trees that describe the response of the containment to the severe accident phenomena for each of the Level 1 accident classes. The current Level 2 model retains the IPE containment event tree logic and is directly linked with the Level 1 model via the linked fault tree process. In addition, the SAMA model incorporates several modeling changes to better reflect an improved understanding of Level 2 PSA issues as suggested by an independent peer review, most notably, drywell shell failure due to contact with core debris, several items related to radionuclide release states, and net positive suction head (NPSH) limits following containment venting.

The result of the Level 2 PSA is a set of release categories with their respective frequency and release characteristics. The results of this analysis for Monticello are provided in Table F.2-4 of the ER. The frequency of each release category was obtained from the quantification of the linked Level 1 - Level 2 models. The release characteristics were obtained from the results of modular accident analysis program (MAAP) analyses that were determined to bound the release fraction for the sequences in each release category.

The offsite consequences and economic impact analyses use the MACCS2 code to determine the offsite risk impacts on the surrounding environment and public. Inputs for these analyses include plant-specific and site-specific input values for core radionuclide inventory, source term and release characteristics, site meteorological data, projected population distribution (within an 80-kilometer (50-mile) radius) for the year 2030, emergency response evacuation modeling, and economic data. The core radionuclide inventory is based on the generic boiling water reactor (BWR) inventory provided in the MACCS2 manual, adjusted to represent the Monticello uprated power level of 1,775 megawatt thermals [(MW(t)]. The magnitude of the onsite impacts (in terms of clean-up and decontamination costs and occupational dose) is based on information provided in NUREG/BR-0184 (NRC 1997b).

In the ER, NMC estimated the dose to the population within 80 kilometers (50 miles) of the Monticello site to be approximately 0.38 person-sievert (Sv) (38 person-rem) per year. The breakdown of the total population dose by containment release mode is summarized in Table G-2. Containment failures within the late time frame (greater than 6 hours following declaration of a general emergency) and within the early time frame (less than 6 hours following declaration of a general emergency) provide similar contributions to the population dose risk at Monticello.

Table G-2. Breakdown of Population Dose by Containment Release Mode

Containment Release Mode	Population Dose (person-rem[a] per year)	% Contribution
Late containment failure	20.4	54
Early containment failure	17.6	46
Intact containment	Negligible	Negligible
Total	**38.0**	**100**

[a] 1 person-rem per year = 0.01 person-Sv per year

G.2.2 Review of Nuclear Management Company, LLC's Risk Estimates

NMC's determination of offsite risk at Monticello is based on the following three major elements of analysis:

- The Level 1 and 2 risk models of the 1992 IPE submittals (NSP 1992) and the external events analyses of the 1995 IPEEE submittals (NSP 1995a,b),
- The major modifications to the IPE models that have been incorporated in the Monticello PSA models used to support the SAMA analyses, and

- The MACCS2 analyses performed to translate fission product release frequencies from the Level 2 PSA model into offsite consequence measures.

Each of these analyses was reviewed to determine the acceptability of NMC's risk estimates for the SAMA analysis, as summarized below.

The staff's review of the Monticello IPE is described in an NRC report dated May 26, 1994 (NRC 1994). Based on a review of the original IPE submittal and subsequent supplements and revisions, the staff concluded that the IPE submittal met the intent of Generic Letter (GL) 88-20; that is, the IPE was of adequate quality to be used to look for design or operational vulnerabilities. The IPE did not identify any severe accident vulnerabilities associated with either core damage or poor containment performance.

Although no vulnerabilities were identified, a number of modifications to the plant, procedures and training were identified that had either been implemented, were to be implemented, or were being considered at the time of the completion of the IPE process. The outstanding items have subsequently been implemented, further evaluated and found not to be sufficiently beneficial to be considered further, or have been included as a SAMA in the current evaluation (NMC 2005a,b).

There have been numerous revisions to the IPE model since its submittal. A comparison of internal events CDF between the IPE and the SAMA PSA models indicates an increase of approximately 1.9×10^{-5} per year in the total CDF (from 2.6×10^{-5} per year to 4.47×10^{-5} per year). The increase is mainly attributed to modeling and hardware changes that have been implemented since the IPE was submitted. There has been a significant increase in internal flooding CDF from 6.8×10^{-6} per year to 4.15×10^{-5} per year and a sizeable reduction in the loss of offsite power contribution from 1.3×10^{-5} per year to 1.8×10^{-6} per year due to hardware and modeling changes. A summary listing of those changes that resulted in the greatest impact on the internal events CDF was provided in the ER (NMC 2005a) and are summarized in Table G-3.

The IPE CDF value for Monticello is close to the average of the CDF values reported in the IPEs for BWR 3/4 plants. Figure 11.2 of NUREG-1560 shows that the IPE-based total internal events CDF for BWR 3/4 plants ranges from 9×10^{-8} to 8×10^{-5} per year, with an average CDF for the group of 2×10^{-5} per year (NRC 1997a). It is recognized that other plants have updated the values for CDF subsequent to the IPE submittals to reflect modeling and hardware changes. The current internal events CDF results for Monticello are comparable to other plants of similar vintage and characteristics.

Table G-3. Monticello PSA Historical Summary

PSA Version	Summary of Changes from Prior Model	CDF
1992	IPE Submittal	2.6×10^{-5}
1995	• Added non-safety diesel generator to supply battery chargers	1.37×10^{-5}
	• Added hard pipe containment vent	
	• Improved safety/relief valve (SRV) pneumatics	
	• Added cross-tie for diesel fire pump for low-pressure makeup	
	• Replaced instrument air compressor with one not dependent on service water	
	• Established more realistic success criteria for service water pumps	
1999	• Incorporated effects of extended power uprate	1.44×10^{-5}
2003	• Updated failure rate data	4.43×10^{-5}
	• Revised operator error structure to explicitly model dependencies	
	• Credited manual alignment of low pressure safety systems when control power not available	
	• Incorporated new findings on two significant flood scenarios	
	• Modified recovery modeling for offsite power and emergency diesel generators	
	• Credited control rod drive injection under certain conditions	
SAMA	• Small number of event failure probability changes resulting from data update tasks	4.47×10^{-5}
	• Lowered truncation limit to 1×10^{-11}	

The staff considered the peer reviews performed for the Monticello PSA, and the potential impact of the review findings on the SAMA evaluation. In the ER, NMC described the previous peer reviews, the most significant of which was the Boiling Water Reactor Owners Group (BWROG) Peer Review of the 1995 PSA model conducted in 1997. The BWROG review concluded that the Monticello PSA can be effectively used to support applications involving relative risk significance. NMC stated that all peer review comments, or the evolutions of those peer review comments, are captured by the 2003 model, and that no outstanding model issues exist outside the normal PSA maintenance program and that none of the PSA maintenance tasks are known to have the potential to impact the SAMA conclusions.

Given that the Monticello internal events PSA model has been peer-reviewed and the peer review findings were either addressed or judged to have no adverse impact on the SAMA evaluation, and that NMC satisfactorily addressed staff questions regarding the PSA, the staff *concludes that the internal events Level 1 PSA model is of sufficient quality to support the* SAMA evaluation. Further consideration of the Level 2 PSA model is provided below.

As indicated above, the current Monticello PSA does not include external events. In the absence of such an analysis, NMC used the Monticello IPEEE to identify the highest risk accident sequences and the potential means of reducing the risk of posed by those sequences, as discussed below.

The Monticello IPEEE was submitted in March 1995 (NSP 1995a), in response to Supplement 4 of Generic Letter 88-20. A revision to the IPEEE was submitted in November 1995 (NSP 1995b). Northern States Power did not identify any fundamental weaknesses or vulnerabilities to severe accident risk in regard to the external events related to seismic, fire, or other external events. In a letter dated April 14, 2000, the staff concluded that the submittals met the intent of Supplement 4 to Generic Letter 88-20, and that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities (NRC 2000).

The Monticello IPEEE uses a focused scope Electric Power Research Institute (EPRI) seismic margins analysis. This method is qualitative and does not provide numerical estimates of the CDF contributions from seismic initiators. The seismic IPEEE identified a number of outliers of items within the scope of the Unresolved Safety Issue (USI) A-46 program. Resolution of these outliers was accomplished in the context of USI A-46. Given the satisfactory resolution of these outliers, Monticello found that none of the plant's high confidence in low probability of failure values were less than the 0.3g review level earthquake used in the IPEEE. The NRC review and closure of USI A-46 for Monticello are documented in a letter dated November 12, 1998 (NRC 1998).

Notwithstanding the conclusions of the IPEEE, as part of the SAMA evaluation NMC reviewed the seismic analysis results and history to determine whether there were any unresolved issues that could impact the seismic risk at Monticello, particularly, any unfinished plant enhancements that were needed to ensure equipment on the safe shutdown list would be capable of withstanding the review level earthquake, or any additional plant enhancements that were identified as means of reducing seismic risk but were discarded due to cost considerations. Based on their review, NMC concluded that there were no outstanding issues that could impact the SAMA results.

Based on the licensee's IPEEE efforts to identify and address seismic outliers and the expected cost associated with further seismic risk analysis and potential plant modifications, the staff concludes that the opportunity for seismic-related SAMAs has been adequately explored and that there are no cost-beneficial, seismic-related SAMA candidates.

The Monticello IPEEE fire analysis employed a combination of a probabilistic risk analysis (PRA) and EPRI's fire-induced vulnerability evaluation (FIVE) methodology. An initial screening phase was performed for fire areas outside of the main reactor/turbine building complex. Fire boundaries were then developed considering spread of fire across area boundaries using the FIVE methodology. PRA techniques were then utilized to progressively analyze the various fire accident sequences that could lead to core damage. This involved using the IPE model of internal events to quantify the CDF resulting from a fire-initiating event. The CDF for each zone was obtained by multiplying the frequency of a fire in a given fire zone by the conditional core damage probability associated with that fire zone including, where appropriate, the impact of fire suppression and fire propagation. The potential impact on containment performance and isolation was evaluated following the core damage evaluation.

The total fire CDF was estimated as 7.81×10^{-6} per year (NSP 1995b). The following seven fire areas (room/burn sequences) are considered to be the dominant contributors and comprise more than 80 percent of the total fire CDF:

Fire Area	Area Description	CDF
VIII/9	Control room	1.5×10^{-6}
XII/BS5	Turbine building 931-ft elevation	1.3×10^{-6}
IX/BS4	Feedwater pump area	1.2×10^{-6}
VI/8	Cable spreading room	9.0×10^{-7}
II/BS2	Reactor building 935/962-ft elevation west	5.6×10^{-7}
IX/12A	Lower 4kV switchgear room	5.1×10^{-7}
XXII/BS6	Div. II area of the emergency filtration train building	4.1×10^{-7}

The fire CDF is approximately 17 percent of the current internal events CDF. In the ER, NMC described each of the fire areas listed above and identified candidate SAMAs to potentially reduce the associated fire risk. As a result, NMC identified potential enhancements which it further considered as SAMAs. These include:

- Permanently posting an operator at the alternate shutdown system (ASDS) panel
- Modifying the ASDS panel to include additional system controls, and
- Adding an emergency level control system to the hotwell.

The staff inquired about the status of several insights/potential improvements that were identified by NMC in the IPEEE and Revision 1 to the IPEEE. NMC indicated that two of three improvements identified in the original IPEEE submittal were actually PSA modeling changes that would better reflect actual risk (NMC 2005b). These improvements involve a revision to the service water pump success criteria and the elimination of the SRV dependence on alternating current (ac) power for depressurization to be consistent with a previous plant change. Both enhancements have been incorporated into the current SAMA PSA model. The third improvement, taking credit for control rod drive (CRD) injection after bypassing the load shed logic, has been incorporated into emergency operating procedures and credited in the current PSA model.

Revision 1 of the IPEEE identified two additional modeling improvements (NSP 1995b). These two improvements involve crediting manual fire suppression in areas other than the control room, and crediting CRD injection and the main condenser as a heat sink for fires that do not cause their failure. These improvements have not been credited in the Monticello fire analysis but would tend to reduce the analyzed risk and the potential for cost-beneficial SAMAs; their omission is therefore conservative.

The IPEEE analysis of high winds, floods and other external events followed the screening specified in Supplement 4 to GL 88-20 (NRC 1991) and did not identify any significant sequences or vulnerabilities (NSP 1995a). The Monticello ER qualitatively discusses the risks from high winds, external flooding and probable maximum precipitation, and transportation and nearby facility accidents. NMC considered the potential for SAMAs to reduce these risks, but concluded that no further modifications would be cost-beneficial. It is noted that the risks from aircraft were explicitly excluded since this was being considered in other forums along with other sources of sabotage.

Due to the relatively low contribution to CDF from fire and other external events, NMC doubled the benefit which was derived from the internal events model to account for the contribution from external events. This doubling was not applied to those SAMAs that specifically addressed fire risk (i.e., SAMAs 38 through 40), since these SAMAs are specific to fire risks and would not have a corresponding risk reduction in internal events. The fire risk analysis is described in the IPEEE and in the environmental report as producing conservative CDF results. While conservative assumptions were used for the majority of fire areas, other aspects of the analysis were considered to be optimistic (NRC 2000). Thus, the degree of conservatism in the result is not clear. Notwithstanding the above, the staff agrees with the applicant's conclusion that the risks posed by external events is less than that due to internal events. Therefore, the staff concludes that the applicant's use of a multiplier of two to account for external events is reasonable for the purposes of the SAMA evaluation.

The staff reviewed the general process used by NMC to translate the results of the Level 1 PSA into containment releases, as well as the results of this Level 2 analysis. NMC characterized the releases for the spectrum of possible radionuclide release scenarios using a set of seven release categories, defined based on the timing and magnitude of the release. The frequency of each release category was obtained from the quantification of a linked Level 1 - Level 2 model which effectively evaluates a containment event tree for each Level 1 accident sequence. Four containment event trees were utilized that differentiated between intact or failed containment at the time of reactor vessel failure and events with and without SBO. The release characteristics for each release category were obtained from the results of MAAP 4.0.5 analyses of conservatively determined representative sequences for each category. The process for assigning accident sequences to the various release categories and selecting a representative accident sequence for each release category is described in the ER and in response to RAIs (NMC 2005a,b). The release categories and their frequencies are presented in Tables F.2-2, F.2-3, and F.2-4 of the ER (NMC 2005a). All releases were modeled as occurring at ground level and with a thermal content the same as ambient. The staff concludes that the process used for determining the release category frequencies and source terms is reasonable and appropriate for the purposes of the SAMA analysis.

The total frequency of releases resulting from the Level 2 analysis is slightly greater than the CDF. In the ER and in response to an RAI, NMC stated that the difference is due to the inclusion of some non-minimal cutsets for scenarios that have higher releases than the corresponding minimal cutsets for the scenarios assessed for the CDF (NMC 2005a,b). The frequency of these non-minimal cutsets should have been subtracted from the frequency of the lower release categories. Therefore, this introduces a slight conservatism in the SAMA analysis.

The staff's review of the Level 2 IPE (NRC 1994) concluded that it addressed the most important severe accident phenomena normally associated with the Mark I containment type, and identified no significant problems or errors.

The Level 2 PSA model was independently reviewed in 2004 by an NMC contractor who concluded that the model was adequate to support the SAMA analysis subject to the resolution of three issues. These issues are:

- Updating the drywell shell failure probabilities due to debris contact

- Addressing items related to radionuclide release states (including shell failure timing and application of drywell spray for the prevention of shell failure, matching order of events in accident sequences to emergency procedure instructions, and how accident scenarios are represented in MAAP)

- Including established NPSH limits for low pressure coolant injection/containment spray operation following containment venting in the MAAP analysis.

These items were resolved in the Level 2 model used for the SAMA analysis. The staff notes that the above issues could be important to accident progression. Therefore, the decision to incorporate updates in these areas appears reasonable.

Based on the staff's review of the Level 2 methodology, and the fact that the Level 2 model was reviewed in more detail as part of the BWROG peer review and a more recent independent contractor review and updated to address the review findings, the staff concludes that the Level 2 PSA provides an acceptable basis for evaluating the benefits associated with various SAMAs.

As mentioned previously, the reactor core radionuclide inventory used in the consequence analysis is based on the generic BWR inventory provided in the MACCS2 manual, adjusted to represent the Monticello power level of 1775 MWt. In response to an RAI concerning the impact of current and future fuel management practices, NMC performed an additional Monticello-specific MACCS2 sensitivity calculation assuming a 65 percent increase in the inventories for Sr-90, Cs-134, and Cs-137. This level of increase was based on a prior calculation for Nine Mile Point in which the end-of-cycle activity levels for a bounding case of 1400 effective full power days were compared to the reference BWR inventories. Use of this increased inventory results in an approximately 29 percent increase in the total costs associated with a severe accident at Monticello. Using realistic mid-life or average conditions would result in a smaller increase. NMC assessed the impact that this change might have on the SAMA screening process and determined that one SAMA (SAMA 39) could become marginally cost-beneficial. However, this SAMA was already identified as potentially cost-beneficial when using a 3 percent real discount rate and when 95th percentile values are used, as discussed in Section G.6.2. Based on this limited impact, the staff concludes that the scaling based on the plant-specific power level yields sufficiently accurate and reasonable results for the dose assessment.

The staff reviewed the process used by NMC to extend the containment performance (Level 2) portion of the PSA to an assessment of offsite consequences (essentially a Level 3 PRA). This included consideration of the source terms used to characterize fission product releases for the applicable containment release category and the major input assumptions used in the offsite consequence analyses. The MACCS2 code was utilized to estimate offsite consequences. Plant-specific input to the code includes the source terms for each release category and the reactor core radionuclide inventory (both discussed above), site-specific meteorological data, projected population distribution within an 80-kilometer (50-mile) radius for the year 2030, emergency evacuation modeling, and economic data. This information is provided in Appendix F of the ER (NMC 2005a).

NMC used site-specific meteorological data for the 2000 calendar year as input to the MACCS2 code. The data were collected from the onsite meteorological tower. In response to an RAI, NMC stated that it considered the year 2000 data to be representative of 5-year meteorological data previously tabulated for the alternate source term project. Small data voids were filled using interpolation between data points. Larger data voids were filled using data from the previous, or following, week for the same time of day. The staff notes that previous SAMA analyses results have shown little sensitivity to year-to-year differences in meteorological data and concludes that the use of the 2000 meteorological data in the SAMA analysis is reasonable.

The population distribution the applicant used as input to the MACCS2 analysis was estimated for the year 2030, using SECPOP2000 (NRC 2003), U.S. Census block-group level population data, and population growth rate estimates (USCB 2000a). The 1990 and 2000 census data were used to estimate a regional annual average population growth rate (USCB 2000b). This annual average population growth rate was applied uniformly to all sectors to calculate the year 2030 population distribution, which NMC has determined is conservative relative to the population projections based on the county-specific growth rates. The staff concludes that the methods and assumptions for estimating population are reasonable and acceptable for purposes of the SAMA evaluation.

The emergency evacuation model was modeled as a single evacuation zone extending out 10 miles from the plant. It was assumed that 95 percent of the population would move at an average speed of approximately 2.5 miles per hour with a delayed start time of 30 minutes (NMC 2005a). This assumption is conservative relative to the NUREG-1150 study (NRC 1990), which assumed evacuation of 99.5 percent of the population within the emergency planning zone. The staff concludes that the evacuation assumptions and analysis are reasonable and acceptable for the purposes of the SAMA evaluation.

Much of the site-specific economic data was provided from SECPOP2000 (NRC 2003) by specifying the data for each of the counties surrounding the plant, to a distance of 50 miles. SECPOP2000 utilizes economic data from the 1997 Census of Agriculture (USDA 1998). In addition, generic economic data that applied to the region as a whole were revised from the MACCS2 sample problem input when better information was available. These included the value of farm and non-farm wealth and the fraction of farm wealth from improvements (e.g., buildings, equipment). Information on the duration of growing seasons for the remaining crops (pasture, green leafy vegetables, roots/tubers and other food crops) were the same as those used in all five NUREG-1150 sites (NRC 1990). NMC compared these data against the information that was available for Minnesota and judged them to be reasonable.

The staff noted that the offsite economic cost risk at Monticello is larger than that estimated at other sites having similar core damage frequency and population doses. In response to the staff's RAI, NMC stated that the economic value parameters used as input to the Monticello MACCS2 analyses are consistent with industry guidance, and produced results that are considered to be appropriate for the Monticello site. Upon further review by the staff, the differences in offsite economic cost risk between sites appear to be due to the differences in the site-specific 50-mile population distributions.

The staff concludes that the methodology used by NMC to estimate the offsite consequences for Monticello provides an acceptable basis from which to proceed with an assessment of risk reduction potential for candidate SAMAs. Accordingly, the staff based its assessment of offsite risk on the CDF and offsite doses reported by NMC.

G.3 Potential Plant Improvements

The process for identifying potential plant improvements, an evaluation of that process, and the improvements evaluated in detail by NMC are discussed in this section.

G.3.1 Process for Identifying Potential Plant Improvements

NMC's process for identifying potential plant improvements (SAMAs) consisted of the following elements:

- Review of the most significant basic events from the Levels 1 and 2 PSA,
- Review of Phase II SAMAs from license renewal applications for seven other U.S. nuclear sites,
- Review of potential plant improvements identified in the Monticello IPE and IPEEE, and
- Review of seven dominant room/burn areas, and SAMAs that could potentially reduce the associated fire risk.

Based on this process, an initial set of 40 candidate SAMAs, referred to as Phase I SAMAs, was identified. In Phase I of the evaluation, NMC performed a qualitative screening of the initial list of SAMAs and eliminated SAMAs from further consideration using the following criteria:

- The SAMA is not applicable at Monticello due to design differences,
- The SAMA is of low benefit in boiling water reactors,

- The SAMA has already been implemented at Monticello or its benefit has been achieved at Monticello using other means, or
- The SAMA costs more than $8.6 million to implement (the modified maximum averted cost-risk (MMACR), which represents the dollar value associated with completely eliminating all internal and external event severe accident risk at Monticello).

Based on this screening, 24 SAMAs were eliminated leaving 16 for further evaluation. The remaining SAMAs, referred to as Phase II SAMAs, are listed in Table F.5-4 of the ER (NMC 2005a).

During Phase II of the evaluation, NMC screened out one additional SAMA because its benefit was small compared to its relevant importance ranking. A detailed evaluation was performed for each of the 15 remaining SAMA candidates, as described in Sections G.4 and G.6 below. To account for the potential impact of external events, the estimated benefits based on internal events were multiplied by a factor of two (except for those SAMAs specific to fire risks, since those SAMAs would not have a corresponding benefit on the risk from internal events.)

NMC also assessed the impact on the initial screening if the MMACR were based on a 3 percent discount rate rather than 7 percent, or if the MMACR were increased by a factor of 2.5 to reflect the potential impact of uncertainties. As a result, three additional SAMAs would have been retained for the Phase II analysis. Theses SAMAs are discussed further in Section G.6.2.

G.3.2 Review of Nuclear Management Company, LLC's Process

NMC's efforts to identify potential SAMAs focused primarily on areas associated with internal initiating events and internal fires. The initial list of SAMAs generally addressed the accident sequences considered to be important to CDF from functional, initiating event, and risk reduction worth perspectives at Monticello, and included selected SAMAs from prior SAMA analyses for other plants.

The preliminary review of NMC's SAMA identification process raised some concerns regarding the completeness of the set of SAMAs identified. The staff requested information on certain improvements that were identified during the IPE but that did not appear to be addressed by a candidate SAMA (NRC 2005). In response to the RAI, NMC explained that one of the improvements (assure faster operation of the condensate demineralizer bypass valve on loss of air) should have been considered as a SAMA but was not. Upon further review by NMC, the potential benefit for the modification would be less than $2000, which is significantly less than the cost, and therefore, would not be justified (NMC 2005b). NMC stated that a second modification (operator training on recovery of the failed RHR) is addressed by SAMA 26, operator training on a failed main condenser. For the remaining modification in question (testing of the boron injection hose), NMC stated that the alternate boron injection has a very small impact on CDF, and that the associated recommendation was subsumed by SAMA 13 (NMC 2005b).

The staff also questioned the ability of two SAMAs to accomplish their intended objectives, i.e., SAMA 7, rupture disk bypass line, and SAMA 36, divert water from TB931 East. SAMA 7 was subsumed by SAMA 16, passive overpressure relief. The staff noted that SAMA 16 does not address the same failure modes that are relevant to SAMA 7 (NMC 2005a). NMC stated that SAMA 16 does not directly address rupture disk failure; however, SAMA 16 was chosen as the best method to address containment vent reliability (NMC 2005b). With regard to SAMA 36, the staff noted that in Table F.5-1 of the ER, basic event IEF_FS-TB931W, which is a flood in the turbine building at the 931-foot elevation west, is indicated to be addressed by SAMA 36 (NMC 2005a). NMC clarified that this SAMA is only applicable to the east region of the turbine building but was included in the importance list as part of the recommended flood mitigation package (NMC 2005b).

Lastly, the staff questioned the applicant about two basic events that have a risk reduction worth of 1.005, but for which no candidate SAMAs were considered. In response to the staff's question, NMC stated that the components involved are a manual bypass switch and a manual disconnect switch that support operation of an instrument ac panel. This particular instrument panel is important because its failure precludes operation of all three containment vent paths, fails division II containment heat removal, trips the mechanical vacuum pump, and fails high pressure coolant injection. NMC argued that manual switches are extremely reliable (i.e., have a very low failure probability); therefore, only an inexpensive modification that could mitigate the consequence would be cost-beneficial (NMC 2005b). The staff agrees that there would be no cost-beneficial SAMAs to address these basic events.

Based on this additional information as described above, the staff concludes that the set of SAMAs evaluated in the ER addresses the major contributors to CDF and offsite dose.

NMC identified Monticello-specific candidate SAMAs for fire events using a combination of the Monticello PSA model and the IPEEE. The fire risk at Monticello is dominated by seven room/burn sequences, the largest contributor being the control room. As a result, three fire-related SAMAs were identified and retained for the Phase II evaluation. Potential plant enhancements for other external events (high winds, external floods, and transportation and nearby facility accidents) were determined to be too costly or bounded by existing scenarios. The staff concludes that the applicant's rationale for eliminating these enhancements from further consideration is reasonable.

The staff notes that the set of SAMAs submitted is not all inclusive, since additional, possibly even less expensive, design alternatives can always be postulated. However, the staff concludes that the benefits of any additional modifications are unlikely to exceed the benefits of the modifications evaluated and that the alternative improvements would not likely cost less than the least expensive alternatives evaluated, when the subsidiary costs associated with maintenance, procedures, and training are considered.

The staff concludes that NMC used a systematic and comprehensive process for identifying potential plant improvements for Monticello, and that the set of potential plant improvements identified by NMC is reasonably comprehensive and therefore acceptable. This search included reviewing insights from the plant-specific risk studies, reviewing plant improvements considered in previous SAMA analyses, and using the knowledge and experience of its PSA personnel.

G.4 Risk Reduction Potential of Plant Improvements

NMC evaluated the risk-reduction potential of the 15 remaining SAMAs that were applicable to Monticello. The changes made to the model to quantify the impact of the SAMAs are detailed in Section F.6 of Appendix F to the ER (NMC 2005a). The SAMA evaluations were performed using realistic assumptions with some conservatism.

NMC used model re-quantification to determine the potential benefits. The CDF and population dose reductions were estimated using the SAMA model version of the Monticello PSA. Table G-4 lists the assumptions considered to estimate the risk reduction for each of the evaluated SAMAs, the estimated risk reduction in terms of percent reduction in CDF and population dose, and the estimated total benefit (present value) of the averted risk. The determination of the benefits for the various SAMAs is further discussed in Section G.6.

For those SAMAs that specifically address fire events (i.e., SAMAs 38 through 40), the reduction in CDF and population dose was not directly calculated. For these SAMAs, a bounding estimate of the impact of the SAMA was made based on general assumptions regarding the approximate contribution to total risk from external events (relative to that from internal events), the fraction of the external event risk attributable to fire events, and the fraction of the fire risk affected by the SAMA and associated with each fire compartment (based on information from the IPEEE.) For example, it is assumed that the contribution to risk from external events is approximately equal to that from internal events, and that internal fires contribute 85 percent of the external events risk. The IPEEE fire analysis was then used to identify the fraction of the fire risk that could be eliminated by potential enhancements in various fire rooms/burn sequences. A similar process was applied to the proposed fire enhancements for each fire room/burn sequence considered.

The staff has reviewed NMC's bases for calculating the risk reduction for the various plant improvements and concludes that the rationale and assumptions for estimating risk reduction are reasonable and somewhat conservative (i.e., the estimated risk reduction is similar to or somewhat higher than what would actually be realized). Accordingly, the staff based its estimates of averted risk for the various SAMAs on NMC's risk reduction estimates.

G.5 Cost Impacts of Candidate Plant Improvements

NMC estimated the costs of implementing the 15 candidate SAMAs through the application of engineering judgement, use of other licensees' estimates for similar improvements, and development of site-specific cost estimates. The cost estimates conservatively did not include the cost of replacement power during extended outages required to implement the modifications, nor did they include contingency costs associated with unforeseen implementation obstacles. The cost estimates provided in the ER did not generally account for inflation. When using costs estimates prior to 1995, NMC applied a 2.75 percent per year inflation rate to arrive at year 2004 estimated costs (e.g., SAMA 39).

The staff reviewed the bases for the applicant's cost estimates (presented in Section F.6 of Appendix F to the ER). For certain improvements, the staff also compared the cost estimates to estimates developed elsewhere for similar improvements, including estimates developed as part of other licensees' analyses of SAMAs for operating reactors and advanced light-water reactors. The staff reviewed the costs and found them to be consistent with estimates provided in support of other plants' analyses.

The staff concludes that the cost estimates provided by NMC are sufficient and appropriate for use in the SAMA evaluation.

Table G-4. SAMA Cost-Benefit Screening Analysis for Monticello

SAMA	Assumptions	% Risk Reduction: CDF	% Risk Reduction: Population Dose	Total Benefit Using 7% Discount Rate ($)	Total Benefit Using 3% Discount Rate ($)	Cost ($)
2—Enhance direct current (dc) power availability by providing a direct connection from DG-13, the security diesel, or another source to the 250-volt (V) battery chargers or other required loads.	Additional credit given for alignment and operation of direct feed line to battery chargers from DG-13. Failure probability of modification is 1E-02	<1	1	79,000	109,000	75,000
4—Install a direct drive diesel injection pump as additional high pressure injection system.	Failure probability of 1E-02 for alignment and operation of this system with no dependencies on other plant systems and not subject to flooding failures. Added to all high-pressure injection failure gates.	98	8	460,000	11,520,000	2,000,000
6—Install additional fan and louver pair for emergency diesel generator (EDG) heating, ventilation, and air conditioning.	Additional credit given for modification in case of failure of both trains of existing EDG room cooling. Failure probability of modification is 1E-02.	2	1	103,000	137,000	100,000
8—Improve EDG-emergency service water (ESW) pumping capability by utilizing the fire service water (FSW) system as a back up for EDG cooling.	Failure probability of 1E-02 for actuation and operation of system. Added to all gates for failure of both divisions of EDG-ESW.	1.8	2.4	211,000	290,000	2,000,000
10—Install drywell igniters or passive hydrogen ignition system.	Additional credit given to new system to prevent hydrogen deflagration with failure probability of 1E-02 for new system.	0	3.5	272,000	380,000	760,000
11—Enhance alternate injection reliability by including the residual heat removal service water and FSW cross-tie valves in the maintenance program.	Assumed valve testing every 5 years leading to a factor of 10 reduction in valve failure probability.	<1.	9	687,000	959,999	50,000

SAMA	Assumptions	% Risk Reduction CDF	% Risk Reduction Population Dose	Total Benefit Using 7% Discount Rate ($)	Total Benefit Using 3% Discount Rate ($)	Cost ($)
12—Proceduralize use of fire pumper truck to pressurize the FSW system.	Replaced existing failure of fire pump that credits a fire pumper truck, with a failure probability of 1E-02, and reduced diesel fire pump failure probability by a factor of 10 based on plant experience.	<1	34	2,12,000	3,684,000	50,000
13—Enhance, test and train on alternate boron injection with the control rod drive system.	Additional credit given by revising the failure probability for boron injection via the reactor water clean up system to 1E-03 with complete dependence on operator action to inject boron via the standby liquid control.	<1	0	3500	4200	50,000
16—Provide passive overpressure relief by changing the containment vent valves to fail open and improving the strength of the rupture disk.	All hard pipe vent failures replaced with effective rupture disk failure probability of 1E-03.	2.5	3.5	279,000	383,000	200,000
28—Develop procedure to refill condensate storage tank (CST) with FSW system.	Insufficient CST volume failure ANDed with 1E-02 failure probability to refill CST.	0	~0	1300	1900	50,000
36—Install interlock to open door to hot machine shop and change swing direction of door to plant administration building to divert water from turbine building 931-foot elevation east.	Failure probability of door to open and divert water to "safe zone" set to 1E-03.	13	23	1,900,000	2,614,000	100,000
37—Develop guidance to allow local, manual control for reactor core isolation cooling (RCIC) operation.	Failure probability of manual operation of RCIC is set to 1E-02 and credit for RCIC injection given following heat removal failure and containment vent success. Dependence on electric power removed for operator success in late injection.	16	-82[(a)]	-5,581,000	-7,850,000	100,000

SAMA	Assumptions	% Risk Reduction CDF	% Risk Reduction Population Dose	Total Benefit Using 7% Discount Rate ($)	Total Benefit Using 3% Discount Rate ($)	Cost ($)
38—Post an operator at the ASDS panel full time.	Eliminate all risk for Class 1A sequences due to fires that require control room evacuation.	Not estimated		331,000	450,000	10,000,000
39—Enhance the ASDS panel to include additional system controls for opposite division.	Eliminate all risk for Class 1D sequences due to fires that require control room evacuation.	Not estimated		753,000	1,025,000	787,000
40—Add an emergency level control system to the hotwell.	Eliminate all risk for Class 2 sequences due to fires that require operator-based hotwell makeup.	Not estimated		178,000	243,000	230,000

[a]Implementation of this SAMA is estimated to result in an 82-percent increase in dose-risk due to the timing of core damage relative to containment failure.

G.6 Cost-Benefit Comparison

NMC's cost-benefit analysis and the staff's review are described in the following sections.

G.6.1 Nuclear Management Company, LLC's Evaluation

The methodology used by NMC was based primarily on NRC's guidance for performing cost-benefit analysis, i.e., NUREG/BR-0184, *Regulatory Analysis Technical Evaluation Handbook* (NRC 1997b). The guidance involves determining the net value for each SAMA according to the following formula:

Net Value = (APE + AOC + AOE + AOSC) - COE where, APE = present value of averted public exposure ($)

AOC = present value of averted offsite property damage costs ($)

AOE = present value of averted occupational exposure costs ($)

AOSC = present value of averted onsite costs ($)

COE = cost of enhancement ($).

If the net value of a SAMA is negative, the cost of implementing the SAMA is larger than the benefit associated with the SAMA and it is not considered cost-beneficial. NMC's derivation of each of the associated costs is summarized below.

NUREG/BR-0058 has recently been revised to reflect the agency's policy on discount rates. Revision 4 of NUREG/BR-0058 states that two sets of estimates should be developed one at 3 percent and one at 7 percent (NRC 2004). NMC provided both sets of estimates (NMC 2005a).

- **Averted Public Exposure (APE) Costs**

The APE costs were calculated using the following formula:

APE = Annual reduction in public exposure (person-rem/year)

x monetary equivalent of unit dose ($2000 per person-rem)

x present value conversion factor (10.76 based on a 20-year period with a 7 percent discount rate).

As stated in NUREG/BR-0184 (NRC 1997b), it is important to note that the monetary value of the public health risk after discounting does not represent the expected reduction in public health risk due to a single accident. Rather, it is the present value of a stream of potential losses extending over the remaining lifetime (in this case, the renewal period) of the facility. Thus, it reflects the expected annual loss due to a single accident, the possibility that such an accident could occur at any time over the renewal period, and the effect of discounting these potential future losses to present value. For the purposes of initial screening, which assumes elimination of all severe accidents, NMC calculated an APE of approximately $817,000 for the 20-year license renewal period.

- **Averted Offsite Property Damage Costs (AOC)**

The AOCs were calculated using the following formula:

AOC = Annual CDF reduction

x offsite economic costs associated with a severe accident (on a per-event basis)

x present value conversion factor.

For the purposes of initial screening which assumes all severe accidents are eliminated, NMC calculated an annual offsite economic risk of about $253,600 based on the Level 3 risk analysis. This results in a discounted value of approximately $2,730,000 for the 20-year license renewal period.

- ***Averted Occupational Exposure (AOE) Costs***

The AOE costs were calculated using the following formula:

AOE = Annual CDF reduction

x occupational exposure per core damage event

x monetary equivalent of unit dose

x present value conversion factor.

NMC derived the values for averted occupational exposure from information provided in Section 5.7.3 of the regulatory analysis handbook (NRC 1997b). Best estimate values provided for immediate occupational dose (3300 person-rem) and long-term occupational dose (20,000 person-rem over a 10-year cleanup period) were used. The present value of these doses was calculated using the equations provided in the handbook in conjunction with a monetary equivalent of unit dose of \$2000 per person-rem, a real discount rate of 7 percent, and a time period of 20 years to represent the license renewal period. For the purposes of initial screening, which assumes all severe accidents are eliminated, NMC calculated an AOE of approximately \$17,000 for the 20-year license renewal period.

- **Averted Onsite Costs**

Averted onsite costs (AOSC) include averted cleanup and decontamination costs and averted power replacement costs. Repair and refurbishment costs are considered for recoverable accidents only and not for severe accidents. NMC derived the values for AOSC based on information provided in Section 5.7.6 of NUREG/BR-0184, the regulatory analysis handbook (NRC 1997b).

NMC divided this cost element into two parts the onsite cleanup and decontamination cost, also commonly referred to as averted cleanup and decontamination costs, and the replacement power cost.

Averted cleanup and decontamination costs (ACC) were calculated using the following formula:

ACC = Annual CDF reduction

x present value of cleanup costs per core damage event

x present value conversion factor.

The total cost of cleanup and decontamination subsequent to a severe accident is estimated in NUREG/BR-0184 to be \$1.1 x 10^9 (discounted over a 10-year cleanup period). This value integrated over the term of the proposed license extension. For the purposes of initial screening, which assumes all severe accidents are eliminated, NMC calculated an ACC of approximately \$529,000 for the 20-year license renewal period.

Long-term replacement power costs (RPC) were calculated using the following formula:

RPC = Annual CDF reduction

x present value of replacement power for a single event

x factor to account for remaining service years for which replacement power is required

x reactor power scaling factor

NMC based its calculations on the value of 587 megawatt electrics (MWe), which is the current electrical output for Monticello (after the extended power uprate). Therefore, NMC applied power scaling factors of 587 MWe/910 MWe to determine the replacement power costs. For the purposes of initial screening, which assumes all severe accidents are eliminated, NMC calculated the AOSC to be approximately $227,500 for the 20-year license renewal period.

Using the above equations, NMC estimated the total present dollar value equivalent associated with completely eliminating severe accidents at Monticello to be about $4,320,000. To account for additional risk reduction in external events, NMC doubled this value (to $8,642,000), to provide the MMACR, which represents the dollar value associated with completely eliminating all internal and external event severe accident risk at Monticello.

For each of the SAMAs remaining after the initial screening, the averted costs associated with the SAMA were estimated using the above equations in conjunction with the CDF and population dose reductions for the SAMA.

- **NMC's Results**

If the implementation costs for a candidate SAMA were greater than the MMACR of $8,642,000, then the SAMA was screened from further consideration. A more refined look at the costs and benefits was performed for the remaining SAMAs. If the implementation costs for a candidate SAMA exceeded the calculated benefit, the SAMA was considered not to be cost-beneficial. In the baseline analysis contained in the ER (using a 7-percent discount rate), NMC identified seven potentially cost-beneficial SAMAs. Based on an analysis using a 3-percent real discount rate, as recommended in NUREG/BR-0058 (NRC 2004), two additional SAMA candidates were determined to be potentially cost-beneficial. The potentially cost-beneficial SAMAs are:

- SAMA 2—enhance dc power availability by providing a direct connection from diesel generator 13, the security diesel, or another source to the 250 V battery chargers or other required loads.

- SAMA 4—install a direct drive diesel injection pump as additional high pressure injection system.

- SAMA 6—install additional fan and louver pair for EDG heating, ventilation, and air conditioning.

- SAMA 11—enhance alternate injection reliability by including the residual heat removal service water and FSW cross-tie valves in the maintenance program.

- SAMA 12—proceduralize the use of a fire pumper truck to pressurize the FSW system.

- SAMA 16— provide passive overpressure relief by changing the containment vent valves to fail open and improving the strength of the rupture disk.

- SAMA 36—install an interlock to open the door to hot machine shop and change swing direction of door to plant administration building to divert water from turbine building 931-foot elevation east.

- SAMA 39—upgrade the ASDS panel to include additional system controls for opposite division (cost-beneficial at 3 percent discount rate).

- SAMA 40—add emergency level control sensor and control valve to the hotwell (cost-beneficial at 3 percent discount rate).

NMC performed additional analyses to evaluate the impact of parameter choices and uncertainties on the results of the SAMA assessment (NMC 2005a). If the benefits are increased by a factor of 2.5 to account for uncertainties, one additional SAMA candidate (beyond those identified in the 3 percent discount rate case) was determined to be potentially cost-beneficial. The potentially cost-beneficial SAMAs, and NMC's plans for further evaluation of these SAMAs are discussed in more detail in Section G.6.2.

G.6.2 Review of Nuclear Management Company, LLC's Cost-Benefit Evaluation

The cost-benefit analysis performed by NMC was based primarily on NUREG/BR-0184 (NRC 1997b) and was executed consistent with this guidance.

In order to account for external events, NMC multiplied the internal event benefits by a factor of two for each SAMA, except those SAMAs that specifically address fire risk (SAMAs 38 through 40). Doubling the benefit for these SAMAs is not appropriate since these SAMAs are specific to fire risks and would not have a corresponding risk reduction in internal events. Given that the CDF from internal fires and other external events as reported by NMC is less than the CDF for internal events, the staff agrees that the factor of two multiplier for external events is reasonable.

NMC considered the impact that possible increases in benefits from analysis uncertainties would have on the results of the SAMA assessment. Currently, an uncertainty distribution is not available for the SAMA PSA model. Therefore, NMC reviewed the point estimate and 95th percentile CDFs for several SAMA submittals. The factor by which the 95th percentile CDFs exceed the point estimate CDFs ranged from 2.35 to 2.45 (NMC 2005a). NMC re-examined the initial set of SAMAs to determine if any additional Phase I SAMAs would be retained for further analysis if the benefits (and MMACR) were increased by a factor of 2.5. Three such SAMAs were identified, specifically, SAMAs 1, 9, and 14. However, based on further consideration of costs and limited effectiveness, NMC concluded that SAMAs 1 and 14 could not be cost-beneficial even if the systems were 100 percent reliable. NMC also considered the impact on the Phase II screening if the estimated benefits were increased by a factor of 2.5 (in addition to the factor of two multiplier already included in the baseline benefit estimates to account for external events). One additional SAMA (SAMA 9 dedicated alternate low-pressure injection/drywell spray system) became potentially cost-beneficial in NMC's analysis.

NMC recognized that a combination of low-cost SAMAs can provide much of the risk reduction associated with higher-cost SAMAs, and may act synergistically to yield a combined risk reduction greater than the sum of the benefits for each SAMA if implemented individually. NMC identified the following six low-cost SAMAs as a "recommended" combination of SAMAs that substantially reduces risk at Monticello for a relatively low cost of implementation. They are:

- SAMA 2—enhance dc power availability by providing a direct connection from diesel generator 13, the security diesel, or another source to the 250 V battery chargers or other required loads.
- SAMA 11—enhance alternate injection reliability by including the residual heat removal service water and FSW cross-tie valves in the maintenance program.
- SAMA 12—proceduralize the use of a fire pumper truck to pressurize the FSW system.
- SAMA 28—develop a procedure to refill the CST with FSW system.
- SAMA 36—install an interlock to open the door to hot machine shop and change swing direction of door to plant administration building to divert water from turbine building 931-foot elevation east.
- SAMA 37—develop guidance to allow local, manual control for RCIC operation.

To assess the impact of the implementation of the recommended SAMAs, NMC made the same modeling changes to the PSA as used previously to represent to the implementation of the SAMAs individually. The combined implementation cost of the set was assumed to be the sum of the individual SAMA implementation costs, without consideration given to the timing or manner in which the SAMAs are implemented. Implementation of the recommended SAMAs was estimated to result in an 86-percent reduction in CDF and an 80 percent reduction in dose, yielding a total benefit of almost $7 million. The combined implementation cost for the set is estimated to be $425,000.

The staff noted that two of the SAMAs in the set (SAMAs 28 and 37) were not previously identified as cost-beneficial, and that one of the two SAMAs (SAMA 37, manual RCIC operation) was actually estimated to result in an 82-percent increase in dose risk if implemented individually. The staff questioned how the implementation of SAMA 37, in combination with several other SAMAs, results in a large (approximately 80-percent) net decrease in risk (NRC 2005). In response to the staff's question, NMC stated that for a prolonged SBO, RCIC alone does not represent a success path, i.e., SAMA 37 would not create a success path (NMC 2005b). During SBO, containment heat removal and containment venting are both unavailable. Containment failure due to overpressure would preclude the operators from occupying the RCIC room to support manual operation of the RCIC. Even with manual operation, the RCIC will eventually fail due to lack of water in the CST or overheating if taking suction from the suppression pool. Manual operation of RCIC (i.e., SAMA 37), therefore, delays the core damage while the containment pressurizes due to lack of heat removal. Core damage and vessel melt through in a pressurized containment results in a greater chance of containment failure than if they occur in an unpressurized containment.

When SAMA 37 is implemented in combination with other SAMAs, particularly SAMAs 12 and 28, a new success path is created. SAMA 12 provides a source of containment spray and CST makeup independent of electric power, while SAMA 28 provides for refilling the CST so that RCIC can continue to function; i.e., SAMA 37 (manual operation of RCIC) is successful due to the refill of the CST. SAMA 2 provides power to solenoid valves allowing containment venting for prolonged SBO scenarios, and SAMA 36 delays or prevents loss of dc so that time is available for manual operation of RCIC. The net result is a significantly reduced CDF and risk when these SAMAs are implemented as a group.

Since the ER was submitted, NMC has implemented the six "recommended" SAMAs (SAMAs 2, 11, 12, 28, 36, and 37), and has reassessed the value of the remaining SAMAs. Implementation of the recommended SAMAs reduces the benefit of the remaining Phase II SAMAs (including SAMA 9, which was identified as a result of the uncertainty analysis) such that only one SAMA remains potentially cost-beneficial. SAMA 16 (passive overpressure relief for containment) becomes even more cost-beneficial (to approximately $1 million) because the set of SAMAs implemented by NMC shifts the risk to categories influenced by containment venting, which could be mitigated by SAMA 16. NMC did not identify SAMA 16 as a modification planned for further consideration in the ER. However, in response to an RAI (NRC 2005), NMC stated that after re-evaluating SAMA 16, the value of modifying the hard pipe vent design was found to still be significant, and that the improvement is being pursued to determine if cost-effective modifications can be implemented (NMC 2005b).

The staff concludes that, with the exception of the one potentially cost-beneficial SAMA discussed above, the costs of the SAMAs evaluated would be higher than the associated benefits.

G.7 Conclusions

NMC compiled a list of 40 SAMAs based on a review of: the most significant basic events from the plant-specific PSA, Phase II SAMAs from license renewal applications for other plants, and insights from the plant-specific IPE and IPEEE. A qualitative screening removed SAMA candidates that (1) were not applicable at Monticello due to design differences, (2) were of low benefit in BWRs, (3) had already been implemented at Monticello, (4) had been achieved at Monticello using other means, or (5) exceeded $8.6 million to implement (the modified maximum averted cost-risk). Twenty-four SAMAs were eliminated leaving 16 for evaluation. Another screening removed one additional SAMA leaving 15 SAMAs for further evaluation.

For the remaining SAMA candidates, a more detailed design and cost estimate were developed as shown in Table G-4. The cost-benefit analyses showed that seven of the SAMA candidates were potentially cost-beneficial in the baseline analysis. NMC performed additional analyses to evaluate the impact of parameter choices and uncertainties on the results of the SAMA assessment. As a result, several additional SAMAs were identified as potentially cost-beneficial. NMC evaluated the impact of implementing a selected set of six "recommended" low-cost SAMAs. The evaluation indicated that the remaining SAMAs, with the exception of one SAMA, would no longer be cost-beneficial. Since the ER was submitted, NMC stated that it has implemented all six of the "recommended" SAMAs (SAMAs 2, 11, 12, 28, 36, and 37). NMC is in the process of further evaluating the one remaining cost-beneficial SAMA (SAMA 16).

The staff reviewed the NMC analysis and concludes that the methods used and the implementation of those methods was sound. The treatment of SAMA benefits and costs support the general conclusion that the SAMA evaluations performed by NMC are reasonable and sufficient for the license renewal submittal. Although the treatment of SAMAs for external events was somewhat limited by the unavailability of an external event PSA, the likelihood of there being cost-beneficial enhancements in this area was minimized by: inclusion of several candidate SAMAs related to dominant fire events, improvements that have been realized as a result of the IPEEE process, and inclusion of a multiplier to account for external events.

The staff concurs with NMC's identification of areas in which risk can be further reduced in a cost-beneficial manner through the implementation of all or a subset of the identified, potentially cost-beneficial SAMAs. The staff agrees that the implementation of the "recommended" SAMAs by NMC is beneficial, and that after implementing the recommended SAMAs, only one additional SAMA remains potentially cost-beneficial. However, this SAMA does not relate to adequately managing the effects of aging during the period of extended operation. Therefore, it need not be implemented as part of license renewal pursuant to Title 10 of the *Code of Federal Regulations*, Part 54.

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10. SUPPLEMENTARY NOTES

Docket Number 50-263

11. ABSTRACT *(200 words or less)*

This supplemental environmental impact statement (SEIS) has been prepared in response to an application submitted to the Nuclear Regulatory Commission (NRC) by Nuclear Management Company, LLC (NMC), to renew the operating license for the Monticello Nuclear Generating Plant (Monticello), for an additional 20 years under 10 CFR Part 54. This SEIS includes the staff's analysis that considers and weighs the environmental effects of the proposed action, the environmental impacts of alternatives to the proposed action, and mitigation measures available for reducing or avoiding adverse impacts. It also includes the staff's recommendation regarding the proposed action and responses to the draft SEIS.

The recommendation of the NRC staff is that the Commission determine that the adverse environmental impacts of license renewal for Monticello are not so great that preserving the option of license renewal for energy-planning decisionmarkers would be unreasonable. This recommendation is based on the following: (1) the analysis and findings in the Generic Environmental Impact Statement for License Renewal of Nuclear Plants (NUREG-1437); (2) the Environmental Report submitted by NMC; (3) consultation with other Federal, State, and Local agencies; (4) the staff's own independent review; and (5) the staff's consideration of public comments.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
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