Duke Power
A Duke Energy Company

M. S. Tuckman
Executive Vice President
Nuclear Generation

Duke Power Company
A Duke Energy Company
EC07H
526 South Church Street
P.O. Box 1006
Charlotte, NC 28201-1006

(704) 382-2200 OFFICE
(704) 382-4360 FAX

January 31, 2002

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

Subject: Response to Requests for Additional Information in Support of the Staff Review of the Application to Renew the Facility Operating Licenses of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2

Docket Nos. 50-369, 50-370, 50-413 and 50-414

Dear Sir:

By letter dated June 13, 2001, Duke Energy Corporation (Duke) submitted an Application to Renew the Facility Operating Licenses of McGuire Nuclear Station and Catawba Nuclear Station (Application). The staff is reviewing the information provided in the Application and by letter dated November 21, 2001 identified areas where additional information (RAI) is needed to complete its review of the Severe Accident Mitigation Analysis portion of the McGuire Environmental Report contained within the Application. Please note that in an NRC telecon summary dated December 6, 2001, the staff provided revised RAIs 5 and 8. Duke responses to the staff requests for additional information are provided in Attachment 1 to this letter. None of the responses in Attachment 1 contain any commitments.

If there are any questions, please contact Bob Gill at (704) 382-3339.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachment

A085

Affidavit

M. S. Tuckman, being duly sworn, states that he is Executive Vice President, Nuclear Generation Department, Duke Energy Corporation; that he is authorized on the part of said Corporation to sign and file with the U. S. Nuclear Regulatory Commission the attached responses to staff requests for additional information relative to its review of the Application to Renew the Facility Operating Licenses of McGuire Nuclear Station and Catawba Nuclear Station, Docket Nos. 50-369, 50-370, 50-413 and 50-414 dated June 13, 2001, and that all the statements and matters set forth herein are true and correct to the best of his knowledge and belief. To the extent that these statements are not based on his personal knowledge, they are based on information provided by Duke employees and/or consultants. Such information has been reviewed in accordance with Duke Energy Corporation practice and is believed to be reliable.

M. S. Tuckman

M. S. Tuckman, Executive Vice President
Duke Energy Corporation

Subscribed and sworn to before me this 31ST day of JAN 2002.

Mary P. Nelms
Notary Public

My Commission Expires:

JAN 22, 2006

xc: (w/ Attachment)

L. A. Reyes
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, GA 30303

D. B. Matthews
Director, Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Senior NRC Resident Inspector
McGuire Nuclear Station

Senior NRC Resident Inspector
Catawba Nuclear Station

C. P. Patel
Senior Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

J. H. Wilson
Senior Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

C. I. Grimes
Program Director, License Renewal and Environmental Impacts
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

R. L. Franovich
Senior Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

R. E. Martin
Senior Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

xc: (w/ Attachment)

Henry J. Porter
Assistant Director, Division of Waste Management
Bureau of Land & Waste Management
S.C. Department of Health and Environmental Control
2600 Bull St.
Columbia, SC 29201

R. M. Fry
Director, Division of Radiation Protection
North Carolina Department of Environment, Health, and Natural Resources
3825 Barrett Drive
Raleigh, NC 27609

North Carolina Municipal Power Agency Number 1
1427 Meadowwood Boulevard
P.O. Box 29513
Raleigh, NC 27626

North Carolina Electric Membership Corporation
P.O. Box 27306
Raleigh, NC 27611

Piedmont Municipal Power Agency
121 Village Drive
Greer, SC 29651

Saluda River Electric Cooperative, Inc.
P. O. Box 929
Laurens, SC 29360

Attachment 1
Application to Renew the Operating Licenses of
McGuire Nuclear Station and Catawba Nuclear Station

Responses to NRC Requests for Additional Information
Concerning the McGuire Severe Accident Mitigation Alternatives Analysis
NRC Letter dated November 21, 2001

RAI 1:

Please provide the following information related to the 1997 update to the McGuire probabilistic risk assessment (PRA) and individual plant evaluation (IPE) that form the basis for the severe accident mitigation alternatives (SAMA) analysis:

a. a description of the major changes made to the Level 1 and 2 PRA/IPE previously reviewed by the staff, and their respective impacts on core damage frequency (CDF) and release frequency;

b. a description of the internal and external peer review process used for the updated PRA/IPE; and

c. justification for the estimated steam generator tube rupture (SGTR) - induced core damage frequency of 7.8×10^{-10} per reactor year, which is very low compared to the results of other studies for similar plants (e.g., NUREG-1150 study for Sequoyah shows a value of 7×10^{-6} per year).

Response to RAI 1a:

The Level 1 changes associated with the McGuire PRA Revision 2 model are:

- Updated the data in the models (component reliability, unavailabilities, initiating event frequencies, CCF, and HRA)
- Converted to a single top fault tree from a sequence based solution
- Incorporated plant modifications
- Model enhancement and error corrections as appropriate (e.g., better treatment of 1 versus 2 unit LOOP initiating events)

The most significant data changes are those related to diesel generator performance. Following the IPE, Duke proceeded with a program to improve the DG reliability at McGuire. The reliability improvement that occurred significantly reduced the CDF contributed by the LOOP and Tornado initiators. To a lesser extent, the seismic results are also impacted by the DG reliability data.

The most important changes to the Level 2 analysis included:

- Incorporating an emergency operating procedure change that reduced the likelihood of restarting a reactor coolant pump following core damage, thus reducing the potential for thermally induced steam generator tube rupture
- Modification to the containment event tree logic regarding the potential for corium contact with the containment liner
- Recognized that the refueling water storage tank inventory would drain through a failed reactor vessel in some sequences (e.g., station blackout); this was factored into the CET logic and quantification

Another important change occurred in the ISLOCA evaluation. The generic database adopted for the Revision 2 analysis had significantly higher failure rates for valve ruptures. This resulted in a significant increase in the CDF contributed by the ISLOCA, an important risk contributor.

These changes resulted in a large decrease in the potential for thermally induced steam generator tube ruptures and a slight increase in the potential for early containment failure as a result of corium contact with the containment liner.

Some specific comparisons are made below. The LOOP, tornado, and seismic CDF results are the most sensitive to the DG reliability data.

Core Damage Frequency Estimates

	McGuire IPE (PRA Rev. 1) Core Damage Frequency	McGuire PRA Rev. 2 (1997 Update) Core Damage Frequency
TOTAL	7.4E-05 per year	4.9E-05 per year
LOOP	1.1E-05 per year	2.6E-06 per year
Tornado	1.9E-05 per year	6.5E-06 per year
Seismic	1.4E-05 per year	1.1E-05 per year

Containment Failure Frequencies

Containment Failure Mode	McGuire IPE (PRA Revision 1) Containment Failure Freq.	McGuire PRA Revision 2 (1997 Update) Containment Failure Freq.
Steam Generator Tube Rupture	1.1E-06 per year	2.5E-08 per year
ISLOCA	8.0E-09 per year	2.2E-07 per year
Containment Isolation Failure	4.7E-07 per year	2.2E-07 per year
Early Containment Failure	3.1E-06 per year	3.5E-06 per year
Late Containment Failure	3.0E-05 per year	2.0E-05 per year
Basemat Melt Through	3.8E-06 per year	2.5E-06 per year
No Containment Failure	3.5E-05 per year	2.2E-05 per year

Response to RAI 1b:

The internal review occurs during the conduct of the PRA. Analysis notebooks are prepared by the responsible engineer and independently reviewed and approved. An external peer review was conducted on the original McGuire PRA by the EPRI Nuclear Safety Analysis Center.

Duke Energy is currently participating in the WOG PRA certification program. The McGuire PRA was reviewed in the fall of 2000. The review process focused primarily on the Revision 3 model which was in development at that time. However, the review team did review the Revision 2 material as necessary when the Revision 3 material was incomplete. In general, the review team found that the Duke PRA processes are sufficient to support applications requiring risk significance determination.

Response to RAI 1c:

The McGuire SGTR model incorporated in both the IPE and in the 1997 update relied upon success criteria established during the IPE development. Where applicable, the system success criteria were established with the then current version of the MAAP code. Furthermore, a sequence was categorized as a success because core damage occurred beyond 24 hours, even though a safe stable state had not been attained, this is inconsistent with what is now the generally accepted industry practice. As a result of comments received during the McGuire peer review process, these success criteria were revisited. The new MAAP results showed core damage to occur where the original analysis did not. The outdated success criteria are judged to be the most significant contributors to the comparatively low SGTR initiated CDF previously reported.

The SGTR analysis is being completely revisited in Revision 3 to the McGuire PRA, which is still in development. This new analysis estimates the CDF for SGTR at 5.3E-07 per year, which is more in line with similar plants, and an estimated public risk of approximately 4 person-Rem. Performing a benefit analysis on this new information yields a maximum (completely eliminating SGTR) total averted present worth for the 20 year license renewal period on the order of $101,000 (includes averted offsite person-Rem, averted onsite property damage costs, averted onsite exposure costs, averted offsite property damage, and averted power replacement costs). From a cost benefit standpoint, it seems unlikely that a cost beneficial alternative could be implemented to further reduce the SGTR risk based on such a low benefit estimate.

RAI 2:

Please provide an estimate of the uncertainties associated with the calculated core damage frequency and risk for internal and external events for McGuire, and the rationale for not explicitly considering these uncertainties in the SAMA analyses. This is of particular interest in light of the fact that, for some risk contributors, alternative/additional SAMAs could be postulated that offer much of the benefit of the evaluated SAMAs at a substantially lower cost (see Question 6 below).

Response to RAI 2:

The uncertainty analysis for the McGuire PRA Revision 2 Level 1 produced 5th, 50th, and 95th percentile values for the core damage frequency of 1.2E-05, 3.3E-05, and 1.3E-04 per year respectively. The point estimate for the McGuire PRA core damage frequency is 4.9E-05 per year and this is the CDF estimate used in the SAMA analyses. Other SAMA analyses were reviewed for insights into what scope of analysis satisfied the expectations for a SAMA analysis. None of the analyses reviewed included an assessment of the impact of uncertainty on the conclusions of the SAMA. No specific requirement to consider the uncertainty was identified in the regulations regarding the SAMA analysis.

A quantitative evaluation of the uncertainties in the Level 2 and 3 analyses are beyond the scope of the current PRA program at Duke. Qualitatively, the uncertainty associated with the containment performance and the offsite consequence analysis are judged to be larger than those associated with the core damage frequency calculation. NUREG-1150 results have been reviewed for insights into the expected uncertainties. The NUREG-1150 analysis included a treatment of the uncertainties in the containment performance modeling, but did not include uncertainties in the offsite consequence analysis. Figure 5.10 of NUREG-1150 indicates that the uncertainty range in the 50 mile population dose is approximately 2 orders of magnitude (5th to 95th percentile). The 95th percentile value is approximately 5 times the mean value. These results would be expected to be representative of the uncertainties in the McGuire analysis.

The large margin between the estimated costs and benefits as evaluated in the McGuire SAMA suggest that the conclusions of the analysis would have been unlikely to change if a comprehensive uncertainty analysis could have been included.

RAI 3:

Attachment K (Page 20) of the environmental report states, "For the McGuire containment the conditional probability of having an early release of fission products to the public from early containment failures, isolation failures, and containment bypass following a severe accident is estimated to be approximately 9%." Using the results from the updated McGuire PRA, and considering both internally- and externally-initiated events, please provide:

a. the core damage frequency from events involving station blackout (SBO), including a breakdown into slow SBO and fast SBO;

b. the conditional containment failure probabilities (both "early" and "late") in core damage events involving SBO; and

c. a comparison of the conditional early containment failure probability for McGuire to the conditional early containment failure probabilities reported in a recent NRC-sponsored study by Sandia National Laboratory -- "Assessment of the DCH Issue for Plants with Ice Condenser Containments," NUREG/CR-6427. Also, provide a discussion of the models and assumptions in the McGuire PRA that account for the major differences.

Response to RAI 3a:

The slow station blackout CDF is estimated to be 2.3E-05/year and the fast station blackout CDF is estimated to be 2.7E-07/year. These totals include both the internal and external initiators.

Response to RAI 3b:

Containment failure probabilities are developed for each plant damage state(PDS) in the PRA. For those PDSs included in the slow station blackout frequency, the conditional containment failure probabilities for early failure fell into a range from 0.15 to 0.19. For the fast SBO PDSs, the probabilities ranged from 0.16 to 0.26. The late containment failure conditional probabilities fell into ranges of 0.34 to 0.56 and 0.17 to 0.36 for the slow and fast SBOs respectively.

Response to RAI 3c:

The conditional containment failure probabilities at vessel breach (station blackouts) from NUREG/CR-6427 are 1.0 for the condition with high pressure melt ejection (HPME) and 0.551 for a non-HPME sequence. The McGuire PRA containment event tree (CET) analysis estimates conditional early containment failure probabilities of approximately 0.73 and 0.22 respectively for those conditions. Early containment failure is defined to occur prior to, at, or within 5 hours following vessel breach. For the HPME condition, the largest contribution to containment failure occurs at vessel breach.

The primary difference is the amount of hydrogen assumed to be in containment. According to Appendix B of NUREG/CR-6427, the amount of in-vessel oxidation assumed was equivalent to 58.8% of the clad reacted, and that this level "... corresponds to the high end of the distribution for the fraction of zirconium oxidized ...the median is about 40 percent oxidized."

The Duke analysis estimated the hydrogen released to the containment with a sequence specific analysis using version 3B of the MAAP code. The fraction of clad oxidized (typically 14% to 53%) is less than the ~59% value that is applied in NUREG/CR-6427. As a result, the peak pressure that occurs during the burn is lower. In the Duke HPME case, hydrogen combustion at vessel breach and corium contact with the containment liner contribute approximately equally to the early containment failure probability.

The availability of an ignition source is another important difference. A review of the Figure 4.2 in NUREG/CR-6427 suggests that for the low pressure at vessel breach cases, the hydrogen is assumed to ignite at vessel breach. This assumption is conservative and the McGuire analysis assumed that a random ignition source would be required with a probability of occurrence of 0.25. For the high pressure at vessel breach cases, both NUREG/CR-6427 and the McGuire analyses assume a very high, essentially 1.0 probability of ignition.

Another significant difference is that the Duke analysis considers the possibility that too little hydrogen is generated in vessel for a burn to occur. This is assigned a probability of only 0.1.

RAI 4:

In light of the issues raised in NUREG/CR-6427 concerning the likelihood of early containment failure in SBO events, please provide a reevaluation of the benefits associated with the hydrogen control measures in Table 5-1 (install back-up power to igniters, install containment inerting) assuming a containment response consistent with the findings in NUREG/CR-6427 (i.e., using the containment failure probabilities for direct containment heating (DCH) and non-DCH events provided in Tables 4.21 and 4.24 of NUREG/CR-6427, respectively).

Response to RAI 4:

The risk and benefit values presented in the SAMA analysis (Table 5-1) are 5.5 person-rem with an averted risk value of $121,000 (based on McGuire PRA results).

The weighted average of the containment failure probabilities associated with DCH and non-DCH events from NUREG/CR-6427 (Tables 4.21 and 4.24) is estimated to be 58% for McGuire. The risk and benefit values reevaluated using this NUREG/CR-6427 estimated containment failure probability for McGuire yields an estimated 21.0 person-rem with an averted risk value of $462,000. This result over estimates the benefit to the extent that not all of the early containment failure risk can be eliminated by providing hydrogen control. The other early containment failure modes are still present.

Note: The original version of RAI 5 contained in the NRC transmittal letter dated November 21, 2001 was revised by the NRC telecon summary dated December 6, 2001 to read as follows:

RAI 5:

Based on the McGuire PRA used for the SAMA evaluation, please provide the frequency and population exposure (person-rem within 50 miles) for each containment failure mode (radiological release mode), and a breakdown of the population dose (person-rem per year) by containment end-state (similar to Table 5-4 in NUREG-1437, Supplement 2). Identify which of these release modes most closely represents each of the following scenarios:

a. Early containment failure (i.e., at or around the time of vessel breach) due to hydrogen combustion resulting from a SBO with containment sprays unavailable, and a dry reactor cavity

b. Late containment failure (i.e., within a few hours after vessel breach) due to hydrogen combustion resulting from a SBO with containment sprays unavailable, and a dry reactor cavity

c. Late containment failure (i.e., at on or about 24 hours after the start of core damage) due to gradual containment overpressurization in a SBO with containment sprays unavailable, and a dry reactor cavity

d. No containment failure, containment sprays unavailable, and a dry reactor cavity.

Note: The above scenarios have been lettered to facilitate response.

Response to RAI 5:

In the McGuire PRA Revision 2 analysis there are 31 release categories (radiological release modes). Table RAI 5-1 contains a list of these 31 release categories along with the frequencies and annual person-rem contribution from internal and external initiators. A description of the release categories (e.g., RC501 is defined as an early containment failure, at or around the time of vessel breach, with a wet cavity -- no ex-vessel release of fission products) can be found in Section 6.3.3 (Release Category Definitions) of the McGuire IPE report. Table RAI 5-2 presents a breakdown of the population dose (person-rem per year) by containment end-state (similar to Table 5-4 in NUREG-1437, Supplement 2).

In the McGuire PRA, scenarios that involve a SBO event with inventory available in the refueling water storage tank are expected to result in a wet cavity following reactor vessel breach. The design of the plant consists of an open path from the refueling water storage tank to the reactor coolant system. In the event of low

pressure in the reactor coolant system, such as would be expected to exist after reactor vessel breach, much of the water remaining in the refueling water storage tank will drain into the reactor coolant system and into the cavity through the failed vessel. Therefore, sequences with ex-vessel releases tend to be of low frequency.

Response to RAI 5a:

The scenario described in RAI 5a closely represents release category RC502 (with a total frequency of 1.58E-07 per year and an annual person-rem of 1.68E-01 -- see Table RAI 5-1). Note that these results include early containment failures from all modeled phenomena, not just hydrogen combustion.

Response to RAI 5b:

No exact match to the scenario described in RAI 5b exists in the McGuire PRA. In the McGuire PRA Level 2 analysis the containment failures are early if they occur within 5 hours following reactor vessel breach and late if they occur beyond 5 hours following reactor vessel breach. Thus, the result would be between RC502 identified above and RC606 and RC704, identified below.

Response to RAI 5c:

The scenario described in RAI 5c closely represents release category RC606 (catastrophic containment failure with a total frequency of 8.50E-07 per year and an annual person-rem of 3.08E-01 -- see Table RAI 5-1) and release category RC704 (benign containment failure with a total frequency of 1.03E-07 per year and an annual person-rem of 6.41E-03 -- see Table RAI 5-1). Note that these results include early containment failures from all modeled phenomena, not just gradual overpressurization.

Response to RAI 5d:

The scenario described in RAI 5d closely represents release category RC904 (with a total frequency of 8.30E-09 per year and an annual person-rem of 2.99E-05 -- see Table RAI 5-1).

Table RAI 5-1 Release Category (Frequencies and Person-rem) Results

Release Category	Type	Frequency	Person-rem
RC102	Internal	3.88E-11	1.76E-04
	External	5.63E-11	2.55E-04
	Total	9.51E-11	4.31E-04
RC104	Internal	7.87E-11	5.63E-04
	External	7.07E-13	5.06E-06
	Total	7.94E-11	5.68E-04
RC204	Internal	2.22E-07	2.62E+00
	External	0.00E+00	0.00E+00
	Total	2.22E-07	2.62E+00
RC301	Internal	4.61E-08	3.43E-03
	External	1.45E-07	1.08E-02
	Total	1.91E-07	1.42E-02
RC302	Internal	5.12E-09	7.74E-03
	External	1.63E-08	2.46E-02
	Total	2.14E-08	3.23E-02
RC303	Internal	7.15E-10	4.65E-05
	External	1.32E-09	8.57E-05
	Total	2.03E-09	1.32E-04
RC304	Internal	7.94E-11	1.10E-04
	External	1.49E-10	2.05E-04
	Total	2.28E-10	3.15E-04
RC401	Internal	1.16E-09	3.24E-05
	External	0.00E+00	0.00E+00
	Total	1.16E-09	3.24E-05
RC402	Internal	1.29E-10	8.22E-05
	External	1.63E-10	1.04E-04
	Total	2.91E-10	1.86E-04
RC403	Internal	1.17E-11	3.60E-07
	External	0.00E+00	0.00E+00
	Total	1.17E-11	3.60E-07
RC404	Internal	1.30E-12	9.32E-07
	External	1.92E-11	1.38E-05
	Total	2.05E-11	1.47E-05
RC501	Internal	7.13E-07	1.12E+00
	External	2.67E-06	4.19E+00
	Total	3.38E-06	5.31E+00
RC502	Internal	3.01E-08	3.20E-02
	External	1.28E-07	1.36E-01
	Total	1.58E-07	1.68E-01
RC601	Internal	3.46E-06	1.30E+00
	External	3.30E-06	1.24E+00
	Total	6.76E-06	2.54E+00
RC602	Internal	3.26E-06	5.54E-01
	External	5.81E-06	9.88E-01
	Total	9.07E-06	1.54E+00
RC603	Internal	9.32E-08	1.21E-01
	External	1.99E-07	2.59E-01
	Total	2.92E-07	3.80E-01
RC604	Internal	1.57E-07	9.96E-02
	External	5.78E-07	3.66E-01
	Total	7.35E-07	4.66E-01
RC605	Internal	3.94E-08	1.87E-03
	External	5.32E-08	2.51E-03
	Total	9.26E-08	4.38E-03
RC606	Internal	1.76E-07	6.37E-02
	External	6.74E-07	2.44E-01
	Total	8.50E-07	3.08E-01
RC607	Internal	1.38E-09	2.25E-04
	External	3.91E-09	6.37E-04
	Total	5.29E-09	8.62E-04
RC608	Internal	1.54E-08	7.28E-03
	External	6.57E-08	3.10E-02
	Total	8.12E-08	3.83E-02
RC701	Internal	3.95E-07	1.33E-03
	External	3.90E-07	1.32E-03
	Total	7.86E-07	2.65E-03
RC702	Internal	3.79E-07	1.41E-03
	External	7.09E-07	2.64E-03
	Total	1.09E-06	4.05E-03
RC703	Internal	4.54E-09	3.88E-05
	External	6.34E-09	5.42E-05
	Total	1.09E-08	9.30E-05
RC704	Internal	2.13E-08	1.32E-03
	External	8.22E-08	5.09E-03
	Total	1.03E-07	6.41E-03
RC801	Internal	1.57E-06	3.26E-03
	External	4.34E-07	8.99E-04
	Total	2.01E-06	4.16E-03
RC802	Internal	3.25E-07	8.76E-04
	External	1.23E-07	3.32E-04
	Total	4.47E-07	1.21E-03
RC901	Internal	1.58E-05	3.30E-02
	External	4.57E-06	9.56E-03
	Total	2.04E-05	4.26E-02
RC902	Internal	3.17E-08	8.66E-05
	External	4.52E-08	1.23E-04
	Total	7.69E-08	2.10E-04
RC903	Internal	1.35E-06	6.87E-04
	External	6.65E-07	3.39E-04
	Total	2.01E-06	1.03E-03

Table RAI 5-1 Release Category (Frequencies and Person-Rem) Results (continued)

RC904	Internal	1.64E-09	5.91E-06
	External	6.66E-09	2.40E-05
	Total	8.30E-09	2.99E-05

Table RAI 5-2 Breakdown of Population Dose by Containment End-State (Total Dose = 13.5 person-rem per year)

Containment End-State	% of Total Dose Internal Initiators	% of Total Dose External Initiators	% of Total Dose All Initiators
Steam Generator Tube Rupture	<0.1	<0.1	<0.1
Interfacing Systems LOCA	19.4	0.0	19.4
Containment Isolation Failure	0.1	0.3	0.4
Early Containment Failure	8.5	32.1	40.6
Late Containment Failure	15.9	23.3	39.2
Basemat Melt Through	<0.1	<0.1	<0.1
No Containment Failure	0.3	0.1	0.4
Total	**44.2**	**55.8**	**100**

RAI 6:

Attachment K (Page 24) states, "The cost to implement any of the containment performance improvement alternatives listed in Table 5-1 for McGuire will range anywhere from a few million dollars to tens of millions of dollars...." It is not clear why lower cost improvements that can achieve much of the benefit have not been considered in the evaluation of alternatives. Specifically, for containment hydrogen control, a severe accident management strategy to power a subset of igniters from a portable generator, or the use of passive auto catalytic recombiners (PARs) would cost less than one million dollars and provide a risk reduction similar to the SAMAs evaluated in the Environmental Report (e.g., install backup power to igniters, install containment inerting system). Please provide a discussion of any lower cost improvements that also were considered. If none were considered, please provide an explanation for not doing so, particularly for hydrogen control.

Response to RAI 6:

The containment performance SAMAs considered in the McGuire analysis were compiled from the Watts Bar SAMA analysis with additional alternatives drawn from NUREG-1560. Furthermore, the cost estimates for many of these alternatives were also obtained the Watts Bar SAMA. The SAMA analysis conservatively estimated the potential benefit of providing a backup power supply to the igniters of $238,000. This value was lower than the cost associated with the identified alternative. This level of estimated benefit suggested that no practical alternatives were likely to exist and no further evaluation for additional alternatives was conducted. The alternatives suggested above are evaluated here.

PASSIVE HYDROGEN CONTROL SYSTEM

Other studies (Calvert Cliffs and Arkansas Nuclear One SAMA analyses) have estimated the cost to install a passive hydrogen control system to be on the order of $750,000. This estimated cost is more than 3 times the conservatively estimated potential benefit. If it is assumed that these two estimates are "close enough" to warrant further investigation, a more accurate estimate of the benefit would consider the following. The Containment Event Tree analysis reveals that the early containment failure mode is most sensitive to the availability of the igniters and that the late containment failure probability is much less affected. A better estimate of the benefit is obtained by considering only the early containment failure mode. While this method does neglect the benefit derived from a small reduction in the late containment failure frequency, the reduction in the early containment failure frequency is conservatively evaluated by considering its complete elimination. From the SAMA, the estimated benefit of eliminating the early containment failure frequency is $121,000. This is substantially lower than the estimated cost of a passive hydrogen control system.

AC POWER TO A SUBSET OF IGNITERS

The other alternative suggested is a severe accident management strategy to power a subset of igniters from a portable generator. The cost of such an alternative may be in the range of the expected benefit, though a more detailed evaluation of the benefit may be appropriate. However, the proposed alternative is not judged to have sufficient technical merit to warrant consideration at this time. To our knowledge, there are no publicly available analyses to demonstrate that powering the igniters alone, without the air return fans to provide mixing, actually reduces the containment failure probability. The only relevant analysis that Duke has performed in-house (performed using HECTR) resulted in calculated hydrogen concentrations in various regions of the ice condenser to be in the range of 24 to 39 percent by volume at the time that a burn in the ice condenser upper plenum first occurred. The risk of a detonation in the ice condenser was judged to be a legitimate concern and the analysis concluded that the air return fans are required to assure properly controlled burning of the hydrogen.

The McGuire CET analysis assumes that both fans and igniters are required for effective hydrogen control. Both NUREG/CR-6427 and NUREG-1150 make reference to the potential for detonable concentrations of hydrogen in the ice condenser when the fans are not operating. From Section 3.2.2.2 of NUREG/CR-6427 "... hydrogen can reach detonable concentrations in the ice condenser before concentrations in the upper plenum ... are high enough to initiate a burn ..." The NUREG does go on to say that any detonation for this scenario would be limited to the ice condenser which is less vulnerable to impulsive loadings than the upper plenum. The NUREG-1150 evaluation of the Sequoyah hydrogen ignition system contained the following observation, " ...when power is recovered ... if the igniters are turned on before the air-return fans have diluted the hydrogen concentration at or above the ice beds, the ignition could trigger a detonation or deflagration that could fail containment."

Powering the igniters, all or some subset, without also providing for a well mixed containment atmosphere may not represent a reduction in public health risk. The potential for a containment challenging detonation or deflagration may be increased over the alternative strategy of allowing the containment to become steam inerted and recovering from this condition following restoration of emergency power. Duke notes that the potential for random ignition may still exist and would have to factored into a comprehensive risk assessment. In the absence of a state-of-the-art analysis of hydrogen transport/combustion and containment performance for the condition of powered igniters and no operating air return fans, such a plant modification is judged to be inappropriate.

DEDICATED LINE FROM COWANS FORD HYDRO-ELECTRIC STATION TO MCGUIRE
In addition to the specific question related to an alternate ac power source for hydrogen mitigation, Duke has previously investigated using the Cowans Ford hydro-electric station as an alternate ac power source for McGuire. Any connection between the two stations using overhead lines between the switchyards would be subject to a number of common mode failures (e.g., tornados, switchyard events) of both the offsite power sources and the alternate ac source. Minimizing this potential for common mode failure was assumed to be needed in order to achieve any meaningful benefit. Therefore, the analyzed design included underground routing of the cable from Cowans Ford to McGuire. The estimated cost of this arrangement was slightly in excess of $3 million, far in excess of the benefit that can be derived by the elimination of the station blackout contribution to the core damage frequency (~ $300,000 from RAI Response Table 8-1).

RAI 7:

SAMAs for reducing CDF appear to have been identified through an examination of the top 200 internal and external cut sets from the McGuire PRA, i.e., those that make up at least 0.06 percent of CDF individually (Section 4.2 of Attachment K), and through the use of basic event importance rankings (Section 4.3 of Attachment K).

a. What is the total percentage contribution of these 200 cutsets to CDF?

b. Because some potential SAMAs could impact or eliminate a large number of cutsets, please explain why the method described is viewed as sufficient to identify all potentially cost-beneficial SAMAs aimed at reducing CDF.

c. Please explain why the list of potential SAMAs obtained in the manner described above is viewed as sufficient given that some SAMAs involving the addition of new systems to the plant would not necessarily be identifiable this way.

Response to RAI 7a:

The McGuire PRA Revision 2 estimated total core damage frequency is 4.9E-05 per year (see page 9 of Attachment K of the Application). The sum of the McGuire PRA Revision 2 top 100 internal cut sets is 2.5E-05 per year, and the sum of the top 100 external cut sets is 1.9E-05 per year. Therefore, the estimated core damage frequency for these 200 cut sets is 4.4E-05 per year which represents approximately 90% of the total core damage frequency.

Response to RAI 7(b):

The process used in the McGuire SAMA analysis is described in Sections 4.3 and 4.4 of Attachment K of the Environmental Report.

The top cutsets and the importance rankings were used to help identify potential SAMAs. The risk reduction worth shows the benefit of making an event perfectly reliable. Therefore, using the importance ranking as a process to identify the most significant SAMAs provides a means of showing the potential benefit of those SAMAs for all cutsets.

In addition, the actual benefit calculations were performed on the entire cutset file. The basic event(s) affected by the SAMA under consideration were modified in the cutset file. Therefore, all cutsets containing the event(s), not just those in the printed list, were included in the assessment of the benefit.

Response to RAI 7c:

Table 4-2 of Attachment K of the Environmental Report identifies those SAMAs that were considered for their potential to reduce the CDF. SAMAs numbered 2 through 5 all represent alternatives requiring the addition of new systems or components. We believe these results demonstrate that identifying the important hardware failures and human actions through the process described does identify opportunities for new systems to be added as mitigation alternatives.

Note: The original version of RAI 8 contained in the NRC transmittal letter dated November 21, 2001 was revised by the NRC telecon summary dated December 6, 2001 to read as follows:

RAI 8:

The SAMA analysis assessed benefits in terms of averted offsite person-Rem (public dose) but did not include other averted costs that should be included in accordance with the Regulatory Analysis Guidelines (NUREG/BR-0184). The SAMA analysis should be modified to include all potential averted costs associated with each potential improvement, in particular, replacement power costs, and for potential containment-related SAMAs, the averted offsite property damage costs. In addition, a sensitivity study should be performed to assess the value of SAMAs over the remainder of the current operating license and the license renewal period.

Response to RAI 8:

The response to RAI 8 is provided in two parts. The first part which follows directly provides an analysis of the potential averted costs associated with each potential improvement to reduce core damage frequency including replacement power costs, and for potential containment-related SAMAs, the averted offsite property damage costs for the 20-year license renewal period.

Averted Power Replacement Cost for the 20-year License Renewal Period

The McGuire SAMA analysis, in addition to averted offsite person-rem (public dose), considered averted onsite property damage costs, averted onsite exposure costs, and averted offsite property damage (contained in Attachment K -- see pages 10-13 and Table 4-1 and Table 4-2 of the Application). The only factor Duke did not include in the McGuire SAMA analysis to reduce core damage frequency is the averted power replacement cost. Calculations have now been performed to estimate the potential SAMA benefits for averted power replacement (APR) cost. The results are presented in RAI Response Table 8-1.

It should be noted that in general the potential reductions in core damage frequency and person-Rem risk are estimated for the McGuire SAMA analyses represent the maximum reductions. The actual reductions likely to be achieved would be somewhat less than these estimated values based on the assumption of totally eliminating the associated failure mode. Therefore, the results of the cost-benefit comparisons are considered conservative.

The assumptions and methodology for the averted power replacement benefit calculation:

Averted Power Replacement Cost

The Duke estimate of the annual power replacement cost for McGuire is based on an assumed discount rate of 7% for the 20-year license renewal period.

The estimated present power replacement costs of a severe accident occurring in each year of the license renewal period is given by (equation from NUREG/BR-0184 page 5.44):

$PV_{RP} = [\$1.2E{+}08/0.07][1 - \exp(-0.07 * 20)]^2$

$PV_{RP} = \$9.73E{+}08$

Then, to estimate the net present value of power replacement over the 20-year license renewal (equation from NUREG/BR-0184 page 5.44):

$U_{RP} = [PV_{RP}/0.07][1 - \exp(-0.07 * 20)]^2$

$U_{RP} = \$7.89E{+}09$

Averted Power Replacement Cost (APRC) = U_{RP} * (Change in annual CDF)

Since the averted power replacement cost from the NUREG is in 1990 dollars, an assumption is made to include a 4% inflation rate over 11 years to bring the value into 2001 dollars; therefore,

APRC for 20-year license renewal period = $1.21E+10 * (Change in annual CDF)

The results as presented in RAI Response Table 8-1 show that the cost of implementation of any of the SAMAs still far exceeds the benefit. Therefore, the conclusions of the McGuire SAMA analysis do not change – that is; none of the SAMA alternatives are cost beneficial.

Averted Offsite Property Damage Cost for Containment-related SAMAs for the 20-year License Renewal Period

In the McGuire SAMA analysis, the containment-related SAMA benefit calculations are based on averted offsite person-rem (public dose) and does not include averted offsite property damage benefit assessments. NUREG/CR-6349 provides an

estimated value of $3000/person-rem to be used for offsite consequences (includes both offsite property damage and health-related costs) in performing value-impact analyses.

Averted Offsite Consequences (property damage and public risk) for Containment-related SAMAs

Averted Offsite Costs = $3000/person-Rem * [1 – exp(-0.07 * 20)]/0.07 * (Change in Person-rem)

Averted Offsite Costs for 20-years = $3.23E+04 * (Change in annual Risk)

The results for the containment-related SAMAs, that include averted property damage and public risk, are presented in RAI Response Table 8-2. The results of this sensitivity study still show that none of the alternatives presented in the McGuire SAMA analysis are cost-beneficial.

Sensitivity Study for Remainder of Current Operating and License Renewal Period.

The second part of the response to RAI 8 provides an analysis of the potential averted costs associated with each potential improvement including replacement power costs, and for potential containment-related SAMAs, the averted offsite property damage costs for the remainder of the current operating license and 20-year license renewal period.

An analysis based on a 43-year period (current and life extension for Unit 2) has been performed. The results are presented in RAI Response Tables 8-3 and 8-4. The results of this sensitivity study still show that none of the alternatives presented in the McGuire SAMA analysis are cost-beneficial.

The following equations represent the calculation for benefit estimates for the 43-year current license and license renewal period combined:

Averted Offsite Person-rem (APE)

Averted Public Health Exposure Costs = $2000/person-Rem * [1 – exp(-0.07 * 43)]/0.07 * (Change in Person-Rem)

APE for 43-years = $2.72E+04 * (Change in annual Risk)

Averted Onsite Cleanup Cost (ACC)

The estimated cleanup and decontamination cost for severe accidents is \$1.5 billion (from NUREG/BR-0184 page 5.42). This cost is the sum of equal costs over a 10 year cleanup period. At a 7% discount rate, the present value of this stream of costs is \$1.1 billion.

The net present value of cleanup and decontamination over the license renewal period is estimated from (equation from NUREG/BR-0184 page 5.43):

$U_{CD} = [\$1.1E+09/0.07][1 - \exp(-0.07 * 43)]$

$U_{CD} = \$1.49E+10$

Then,

$\boxed{\text{ACC for 43-years} = \$1.49E+10 * \text{(Change in annual CDF)}}$

Averted Onsite Exposure Cost (AOE)

Assume a discount rate of 7% over the 43-year current and license renewal period.

Immediate Dose (see NUREG/BR-0184 pages 5.30 – 5.33)

$W_{IO} = \$2000/\text{person-Rem} * 3300 \text{ person-Rem} * [1 - \exp(-0.07 * 43)]/0.07 * \text{(Change in CDF)}$

where, 3300 person-Rem = best estimate (from NUREG/BR-0184 page 5.30)
$W_{IO} = \$8.96E+07 * \text{(Change in annual CDF)}$

Long-Term Dose (see NUREG/BR-0184 pages 5.31 – 5.33)

$W_{LTO} = \$2000/\text{person-Rem} * 20{,}000 \text{ person-Rem} * [(1 - \exp(-0.07 * 43))/0.07] * [(1 - \exp(-0.07 * 10))/(0.07 * 10)] * \text{(Change in CDF)}$

where, 20,000 person-Rem = best estimate (from NUREG/BR-0184 page 5.31)

Assume the doses accrue over a 10-year period

$W_{LTO} = \$3.91E+08 * \text{(Change in annual CDF)}$

AOE = $W_{IO} + W_{LTO}$ = ($8.96E+07 + $3.91E+08) * (Change in annual CDF)

AOE for 43-years = $4.80E+08 * (Change in annual CDF)

Averted Offsite Property Damage Cost (AOEC)

In 1990 dollars ≈ $2.46E+08 (assumed from NUREG/BR-0184 Table 5.6 on page 5.38)

Inflating to the year 2001 dollars ≈ $3.79E+08 (assume 4% inflation)

Assume a 7% discount rate for the 43 year current and license renewal period

AOEC = [$3.79E+08/0.07][1 – exp(-0.07 * 43)] * (Change in CDF)

AOEC for 43-years = $5.14E+09 * (Change in annual CDF)

Averted Power Replacement Cost

Assuming an inflation rate of 4% over 11 years to bring the value into 2001 dollars;

APRC for 43-years = $3.08E+10 * (Change in annual CDF)

Averted Offsite Consequences (property damage and public risk) for Containment-related SAMAs

Averted Offsite Costs = $3000/person-Rem * [1 – exp(-0.07 * 43)]/0.07 * (Change in Person-rem)

Averted Offsite Costs for 43-years = $4.07E+04 * (Change in annual Risk)

For the seismic initiators (with a CDF of 1.1E-5 in the SAMA Report) if one assumes that the seismic CDF risk can be completely eliminated, the averted power replacement cost is estimated to be $133,000 for the 20-year license renewal period (or $339,000 for the 43-year current license and license renewal period combined). The seismic CDF is calculated to occur at earthquake accelerations much greater than the design basis earthquake (0.5 g versus 0.1 g) and no simple fixes are found to substantially reduce the seismic risk. As discussed in the McGuire SAMA report, many plant systems would need to be substantially upgraded to significantly increase their seismic ruggedness. The cost of these substantial upgrades in the plant systems seismic ruggedness is much higher than the averted cost of all the benefits.

RAI RESPONSE TABLE 8-1 Summary of Averted Power Replacement Benefit Calculation (20-year license renewal period)

Potential Alternative	Severe Accident Sequences (Basic Event)	Averted Power Replacement Costs	Total Present Worth	Cost of Alternative
Man SSF 24 hours a day with a trained operator This SAMA would eliminate the time factor associated with an operator being dispatched to the SSF. Therefore, for this analysis it is assumed that the DHE events associated with the operators failing to align SS Sys. for operation in time are completely eliminated since there would be no transition time associated with dispatching an operator to start the SSF.	• Loss of RN, failure of operators to align SS Sys. for operation, filter (Standby Makeup Pump) restricts flow, failure to align RV Cooling/other Unit RN • Vital I&C Fire causes a Loss of RN, failure of operators to align SS Sys. for operation, failure to use other Unit or remote control during fire • Loss of 4160V Essential Bus and failure to align SS Sys. for operation (NNVSSFADHE) AND • Tornado causes LOOP, DG 1A and 1B fail to fun, operators fail to initiate SS Sys. operation (NNVSSFBDHE)	$134,000	$380,000	>$5 M
Install automatic swap over to high pressure recirculation. This SAMA would eliminate the operator action required for manual swap over – DHE event.	LOCA cut sets with failure of operators to establish high pressure recirculation (TRECIRCDHE)	$121,000	$291,000	>$1 M
Install automatic swap to RV Cooling/other Unit RN system upon loss of RN This SAMA would eliminate the operator action required to manually align backup cooling to NV pumps.	Loss of RN, failure of operators to align SS Sys. for operation, filter (Standby Makeup Pump) restricts flow, failure to align RV Cooling/other Unit RN (RNUNIT2RHE)	$107,000	$275,000	>$1 M
Install third diesel For this SAMA it is assumed that failures associated with the two diesels already installed (run, start and common cause failures) would be eliminated.	Tornado causes LOOP, DG 1A and 1B fail, and operators fail to initiate SS Sys. operation (JDG001ADGR + JDG001BDGR + JDG001ADGS + JDG001BDGS + JDG1ARNCOM)	$102,000	$304,000	>$2 M
Install automatic swap to other Unit	Vital I&C Fire causes a Loss of RN, failure of operators to align SS Sys. for operation, failure to use other Unit or remote control during fire (FIREFLDRHE)	$35,000	$106,000	>$1 M
Increase test frequency of Standby Makeup Pump flow path (currently tested quarterly)	Loss of RN, failure of operators to align SS Sys. for operation, filter (Standby Makeup Pump) restricts flow, failure to align RV Cooling/other Unit RN (NNVSMUPFLF)	$22,000	$62,000	>$0.4 M
Replace reactor vessel with stronger vessel	Failure of reactor pressure vessel with failure to prevent core damage following an reactor pressure vessel breach (RPV)	$12,000	$30,000	>$1 M

RAI RESPONSE TABLE 8-2

Containment Performance - Averted Offsite Consequence Cost Benefit Results
(Modified Table 5-1 of the SAMA Analysis)
20-year license renewal period

Containment Failure Mode (CFM)	Potential Containment Performance Alternatives To Mitigate CFM	Percentage Of Time Severe Accidents Will End In Particular CFM	Total Person-Rem Risk	Present Worth Of Averted Offsite Property Damage and Public Risk
Late Containment Failures	1. Install independent containment spray system 2. Install filtered containment vent system 5. Install backup power to igniters 8. Install backup power to air return fans 9. Install containment inerting system	41 %	5.3	$171,000
Containment Bypass ISLOCA SGTR	3. Install additional containment bypass instrumentation (ISLOCA) 4. Add independent source of feedwater to reduce induced SGTR	< 1 % (ISLOCA and SGTR combined)	2.6 – ISLOCA < 0.1 – SGTR	$84,000 (ISLOCA) < $3200 (SGTR)
Early Containment Failures	1. Install independent containment spray system 2. Install filtered containment vent system 5. Install backup power to igniters 6. Install reactor cavity flooding system 8. Install backup power to air return fans 9. Install containment inerting system	7 %	5.5	$178,000
Basemat Melt Through	6. Install reactor cavity flooding system 7. Install core retention device	5 %	< 0.1	< $3200

RAI RESPONSE TABLE 8-3 Summary of Benefit Calculations (43-year current license and license renewal period)

Potential Alternative	Severe Accident Sequences (Basic Event)	Averted Public Exposure	Averted Onsite Cleanup Costs	Averted Onsite Exposure	Averted Offsite Property Damage	Averted Power Replacement Costs	Total Present Worth
Man SSF 24 hours a day with a trained operator This SAMA would eliminate the time factor associated with an operator being dispatched to the SSF. Therefore, for this analysis it is assumed that the DHE events associated with the operators failing to align SS Sys. for operation in time are completely eliminated since there would be no transition time associated with dispatching an operator to start the SSF.	• Loss of RN, failure of operators to align SS Sys. for operation, filter (Standby Makeup Pump) restricts flow, failure to align RV Cooling/other Unit RN • Vital I&C Fire causes a Loss of RN, failure of operators to align SS Sys. for operation, failure to use other Unit or remote control during fire • Loss of 4160V Essential Bus and failure to align SS Sys. for operation (NNVSSFADHE) AND • Tornado causes LOOP, DG 1A and 1B fail to fun, operators fail to initiate SS Sys. Operation (NNVSSFBDHE)	$8.7E+04	$1.6E+05	$5.3E+03	$5.7E+04	$3.4E+05	$6.5E+05
Install automatic swap over to high pressure recirculation. This SAMA would eliminate the operator action required for manual swap over – DHE event.	LOCA cut sets with failure of operators to establish high pressure recirculation (TRECIRCDHE)	$1.1E+04	$1.5E+05	$4.8E+03	$5.1E+04	$3.1E+05	$5.2E+05
Install automatic swap to RV Cooling/other Unit RN system upon loss of RN This SAMA would eliminate the operator action required to manually align backup cooling to NV pumps.	Loss of RN, failure of operators to align SS Sys. for operation, filter (Standby Makeup Pump) restricts flow, failure to align RV Cooling/other Unit RN (RNUNIT2RHE)	$3.3E+04	$1.3E+05	$4.2E+03	$4.5E+04	$2.7E+05	$4.8E+05
Install third diesel For this SAMA it is assumed that failures associated with the two diesels already installed (run, start and common cause failures) would be eliminated.	Tornado causes LOOP, DG 1A and 1B fail, and operators fail to initiate SS Sys. operation (JDG001ADGR + JDG001BDGR + JDG001ADGS + JDG001BDGS + JDG1ARNCOM)	$8.4E+04	$1.3E+05	$4.0E+03	$4.3E+04	$2.6E+05	$5.2E+05
Install automatic swap to other Unit	Vital I&C Fire causes a Loss of RN, failure of operators to align SS Sys. for operation, failure to use other Unit or	$3.0E+04	$4.3E+04	$1.4E+03	$1.5E+04	$8.9E+04	$1.8E+05

RAI RESPONSE TABLE 8-3 Summary of Benefit Calculations (43-year current license and license renewal period)

Potential Alternative	Severe Accident Sequences (Basic Event)	Averted Public Exposure	Averted Onsite Cleanup Costs	Averted Onsite Exposure	Averted Offsite Property Damage	Averted Power Replacement Costs	Total Present Worth
	remote control during fire (FIREFLDRHE)						
Increase test frequency of Standby Makeup Pump flow path (currently tested quarterly)	Loss of RN, failure of operators to align SS Sys. for operation, filter (Standby Makeup Pump) restricts flow, failure to align RV Cooling/other Unit RN (NNVSMUPFLF)	$1.4E+04	$2.7E+04	$8.6E+02	$9.3E+03	$5.5E+04	$1.1E+05
Replace reactor vessel with stronger vessel	Failure of reactor pressure vessel with failure to prevent core damage following an reactor pressure vessel breach (RPV)	< $2.7E+03	$1.5E+04	$4.8E+02	$5.1E+03	$3.1E+04	$5.4E+04

RAI RESPONSE TABLE 8-4

Containment Performance - Averted Offsite Consequence Cost Benefit Results
(Modified Table 5-1 of the SAMA Analysis)
43-year current license and license renewal period

Containment Failure Mode (CFM)	Potential Containment Performance Alternatives To Mitigate CFM	Percentage Of Time Severe Accidents Will End In Particular CFM	Total Person-Rem Risk	Present Worth Of Averted Offsite Property Damage and Public Risk
Late Containment Failures	3. Install independent containment spray system 4. Install filtered containment vent system 6. Install backup power to igniters 10. Install backup power to air return fans 11. Install containment inerting system	41 %	5.3	$216,000
Containment Bypass ISLOCA SGTR	7. Install additional containment bypass instrumentation (ISLOCA) 8. Add independent source of feedwater to reduce induced SGTR	< 1 % (ISLOCA and SGTR combined)	2.6 – ISLOCA < 0.1 – SGTR	$106,000 (ISLOCA) < $4100 (SGTR)
Early Containment Failures	3. Install independent containment spray system 4. Install filtered containment vent system 9. Install backup power to igniters 10. Install reactor cavity flooding system 10. Install backup power to air return fans 11. Install containment inerting system	7 %	5.5	$224,000
Basemat Melt Through	8. Install reactor cavity flooding system 9. Install core retention device	5 %	< 0.1	< $4100

RAI 9:

Page 23 of Attachment K states that "...almost all of the large early release frequency (LERF) is attributable to the ISLOCA [interfacing systems loss-of-coolant-accident] initiator." However, Table 5-1 (Page 27) indicates that the conditional "Early Containment Failures" probability is 7 percent and that the ISLOCA & SGTR combined is <1 percent. Please define what is meant by "early" in LERF, how it is different from the "early" in Table 5-1, and whether/how this impacts the SAMA analysis.

Response to RAI 9:

McGuire PRA Revision 2 LERF Definition and Methodology

The McGuire PRA is a full scope Level 3 PRA. As such, health effects such as early fatalities and whole-body person-rem to the surrounding population are calculated. The concept of LERF has been developed as a surrogate for the early fatality risk. The regulatory guides applicable to risk informed regulation have adopted LERF as one of the acceptance criteria. A definition of LERF derived from the Level 3 PRA analysis is established for McGuire as follows.

LERF is the sum of the frequencies of those release categories identified as having a meaningful potential for early fatalities. A meaningful potential for early fatalities is defined as a mean conditional value of early fatalities $\geq$ 0.5 from the off-site consequence results. The results for the McGuire PRA Revision 2 analysis determined that only ISLOCA and SGTRs satisfy this definition. The LERF result presented for the McGuire PRA is based on the early fatality risk results of the Level 3 PRA and may be different in its development that the LERF estimate developed when no Level 3 analysis is available. The use of "early" in this context is that the release occurs early enough that evacuation was not sufficient to prevent a meaningful potential for early fatalities.

McGuire PRA Revision 2 Early Containment Failure Definition

"Early" in early containment failures is defined as releases that occur prior to, at, or early after reactor vessel breach (within 5 hours following reactor vessel breach). The early containment failure release categories did not contribute to LERF because the plant specific evaluation of the early fatality risk produced a mean conditional probability of early fatalities that was less than 0.5.

Because the offsite consequence results themselves are used on the cost/benefit analysis, the definition of LERF adopted in the McGuire PRA has no impact on the SAMA analysis.

RAI 10:

Provide a discussion of the meteorological data and emergency planning assumptions used in performing the SAMA analysis. Provide an assessment of the impact of the license renewal period on emergency planning assumptions (i.e., effects of increased population).

Response to RAI 10:

Meteorological Data

MACCS2 requires a file of hourly meteorological data consisting of wind speed, wind direction, atmospheric stability category, and precipitation. This MACCS2 meteorological data file contains data for one year, that is, 8760 entries for a 365-day year. The McGuire site meteorological tower was recently relocated, for the McGuire SAMA analysis new meteorological data was obtained from the new tower location for the time period January 1, 1999 through December 31, 1999. It is assumed that the meteorological data for this time period is representative and is as valid as any other year that might be selected.

Population Data and Emergency Planning (Evacuation)

The MACCS2 models allow the input of site-specific population data as a function of distance and direction from the reactor site.

The McGuire PRA Revision 2 and the SAMA off-site consequence analyses use three distinct evacuation schemes in order to adequately represent evacuation time estimates for the permanent resident population, the transient population, and the special facility population (schools, hospitals, etc.) The three groups are defined by the time delay from initial notification to start of evacuation. For each evacuation scheme, the fraction of the population starting their evacuation is included. For the permanent resident evacuation schemes, it was assumed that 5 percent of the population would delay evacuation for 24 hours after being warned to evacuate. This is a conservative assumption. (NUREG 1150 used a value of 0.5 percent.) The delay time and fraction of population for the remaining two schemes was developed from information given in the latest update (1993)to the McGuire evacuation time estimate study for the 10 mile Emergency Planning Zone (EPZ). The evacuation schemes include additional information such as evacuation distance, average evacuation speed, sheltering, and shielding considerations.

In the McGuire evacuation model, only the 10-mile Emergency Planning Zone (EPZ) is assumed to be involved in the initial evacuation. For personnel outside of the 10-mile EPZ, the MACCS2 model assumes that they will wait 24 hours before evacuating (provided that radiological conditions warrant evacuation).

The McGuire SAMA analysis assessed the impact of population increases on off-site consequences. The SAMA analysis is based on the evacuation model described above (no change for the license renewal period) with an estimated 50 mile population for the year 2040.

RAI 11:

Figure 6.1 of NUREG/CR-6427 displays fragility curves for McGuire. Is this curve similar to the curves used in the current McGuire PRA? If not, please explain the differences.

Response to RAI 11:

NUREG/CR-6427 states that the plant-specific IPE fragility curves were used in the analysis. The McGuire containment fragility curve has not been changed since the McGuire IPE; therefore, the McGuire curve in Figure 6.1 of NUREG/CR-6427 is the same as used in the current McGuire PRA.