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

Edwin I. Hatch Nuclear Plant
Additional Information Related to the Staff's Review of Severe
Accident Mitigation Alternatives (TAC Nos. MA8096 and MA8098)

Gentlemen:

By letter dated May 30, 2000, the NRC requested additional information (RAI) related to the review of severe accident mitigation alternatives for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. This letter formally submits Southern Nuclear's (SNC) response to the 18 RAIs identified in the NRC May 30, 2000, letter to SNC.

If you have any questions regarding this submittal, please contact this office.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

HLS/JTD

Enclosure: SNC Response to SAMA RAIs

cc: Southern Nuclear Operating Company
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Mr. J. H. Wilson, Environmental Project Manager – Hatch License Renewal
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Mr. L. A. Reyes, Regional Administrator
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A001

ENCLOSURE 1

OFFICE OF NUCLEAR REACTOR REGULATION
REQUEST FOR ADDITIONAL INFORMATION (RAI)
RELATED TO THE STAFF'S REVIEW OF
SEVERE ACCIDENT MITIGATION ALTERNATIVES
RELATED TO LICENSE RENEWAL FOR THE EDWIN I. HATCH NUCLEAR PLANT,
UNITS 1 AND 2 (TAC NOS. MA8096 AND MA8098)

1. **"The original individual plant examination (IPE), as well as the upgrades to address the 1998 power uprate, were based on the RISKMAN "Large event tree, small fault tree" model. The severe accident mitigation alternative (SAMA) analysis is based on a subsequent conversion of the RISKMAN model to a cutset and fault tree analysis (CAFTA) "linked fault tree" model that also included other modeling changes. The risk profile in this updated model appears to be different than that in the IPE (the core damage frequency has decreased, while the frequencies of the five release classes/sequences reported in Section 2.0 of the SAMA submittal are about a factor of 2 to 6 higher than reported in the IPE). To support using the updated risk model in the SAMA identification and evaluation processes, please provide the following:"**

   a. **"A specific reference for the probabilistic risk assessment (PRA) study, and a description of the internal and external peer review of the Level 1, 2, and 3 portions of the study."**

      The E. I. Hatch Probabilistic Safety Analysis (PSA) model used for the *Environmental Report – Operating License Renewal Stage* (ER) SAMA review is a conversion from the IPE model originally submitted to the NRC. IPE model, Revision 1b was used for the conversion. The IPE model was run with RISKMAN Event Tree linking software. The present model uses a CAFTA Fault Tree linking software.

      Due to the distinct change in software, the present Hatch Unit 1 model is identified as follows: Hatch 1 PSA model, Revision 0.

      The Hatch 1 PSA model, Revision 0, includes Level 1 and Level 2 components. It was developed by PLG-EQE (Pickard, Lowe, and Garrick) and was reviewed by Southern Nuclear Company PSA engineering staff. Because this model was developed from the original IPE model, all review pertinent to its content from the original IPE review still applies. In December 2000, this model will be submitted for a Nuclear Engineering Institute (NEI) Peer Review Certification. This certification is based on the GE Owner's Group methodology and is considered acceptable by the proposed *ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*, Revision 12 draft.

      The Hatch Unit 2 PSA model is also a CAFTA model that was converted from the Unit 1 CAFTA model. This method of conversion was acceptable due to the

similarity between plant units. SNC performed and reviewed this conversion with consultation provided by PLG-EQE. The Unit 1 and Unit 2 Plant designs are considered sufficiently similar such that the results of the Unit 1 CAFTA model were applied to both units for the ER SAMA analysis.

The model developed specifically for the Environmental Report Level 3 SAMA analysis is the Melcor Accident Consequence Code System (MACCS2), version 1.12. MACCS2 and its immediate predecessor, MACCS, were developed at Sandia National Laboratory. The code has undergone extensive peer review both in this country and in Europe. The code has been applied to a wide range of hypothetical and real radionuclide accident releases, including extensive use on the Chernobyl release. It is, by far, the most widely used accident consequence code in the U.S. and has been routinely used in license applications to the NRC (e.g., Calvert Cliffs, Peach Bottom). "MACCS2, version 1.12, is the latest version of the NRC's official severe-accident consequence-analysis code." (Reference 10) For the Hatch SAMA analysis, an internal independent experienced accident risk analyst reviewed the Level 3 input parameter development, the use of these parameters in the analysis, the MACCS2 computer input files, and the computer model output files. The latter two were performed to assure that the computer analysis correctly simulated the HATCH SAMA impacts and that the results were interpreted correctly. Questions/comments were resolved between the reviewer and the performing analyst.

b. **"A description of the Level 1 and Level 2 risk profiles, results, and insights in terms of the major contributors (hardware and human failures) to the core damage frequency (CDF) and release frequencies."**

The tables and pie graphs presented in figures 1.b-1 and 1.b-2 provide CDF and large early release fraction (LERF) information regarding Unit 1 major initiating events. The response to RAI question 1.e provides the insight in terms of major contributors (hardware and human failures).

The calculated basic release frequencies for the ER Phase 2 SAMAs are provided in the table 1.b-1. Computer files containing cutsets for the various SAMAs were processed and release frequencies for the various LERF sequences were identified. The probabilities for each occurrence were summed to calculate the release frequency for each sequence as a function of each Phase 2 SAMA.

Table 1.b-1 Calculated Basic Release Frequencies

Phase 2 SAMA	SEQUENCE 2	4	5	11	15	Sum of Annual Risk
	Frequency					
Baseline (also P2-3, P2-10, P2-15)	1.793E-06	7.433E-07	1.656E-07	7.433E-07	9.240E-10	3.446E-06
P2-2	1.793E-06	7.433E-07	1.656E-07	7.433E-07	9.240E-10	3.446E-06
P2-5	1.760E-06	7.432E-07	1.656E-07	7.432E-07	9.240E-10	3.413E-06
P2-7	1.793E-06	7.415E-07	1.656E-07	7.415E-07	9.240E-10	3.443E-06
P2-8	1.767E-06	7.423E-07	1.656E-07	7.423E-07	9.240E-10	3.418E-06
P2-11	1.792E-06	7.433E-07	1.656E-07	7.433E-07	9.240E-10	3.445E-06
P2-12	1.793E-06	7.433E-07	1.656E-07	7.433E-07	9.240E-10	3.446E-06
P2-14	1.793E-06	7.433E-07	1.656E-07	7.433E-07	9.240E-10	3.446E-06

Initiator Distribution Hatch Unit 1 CDF

EVENT	%	CDF	EVENT DESCRIPTION
ATWS	4.3	6.92E-07	Anticipated Transient Without Scram
FL-BUSC	8.4	1.36E-06	Loss of 600VAC Bus C
FL-LODC	18	2.90E-06	Loss of Station Battery A
FL-LOPSW	4.1	6.56E-07	Loss of Plant Service Water
IORV	3.7	5.97E-07	Inadvertently Opened SRV
LOFW	20.2	3.27E-06	Loss of Feedwater
LOSP	16.7	2.70E-06	Loss of Offsite Power
MLOCA	2	3.28E-07	Medium Break LOCA
MSIVC	7.3	1.17E-06	MSIV Closure
TTRIP	3.9	6.29E-07	Turbine Trip
OTHER EVENTS	11.3	1.85E-06	Various System Line Breaks Outside Containment
TOTALS	100	1.62E-05	

% Initiator Distribution

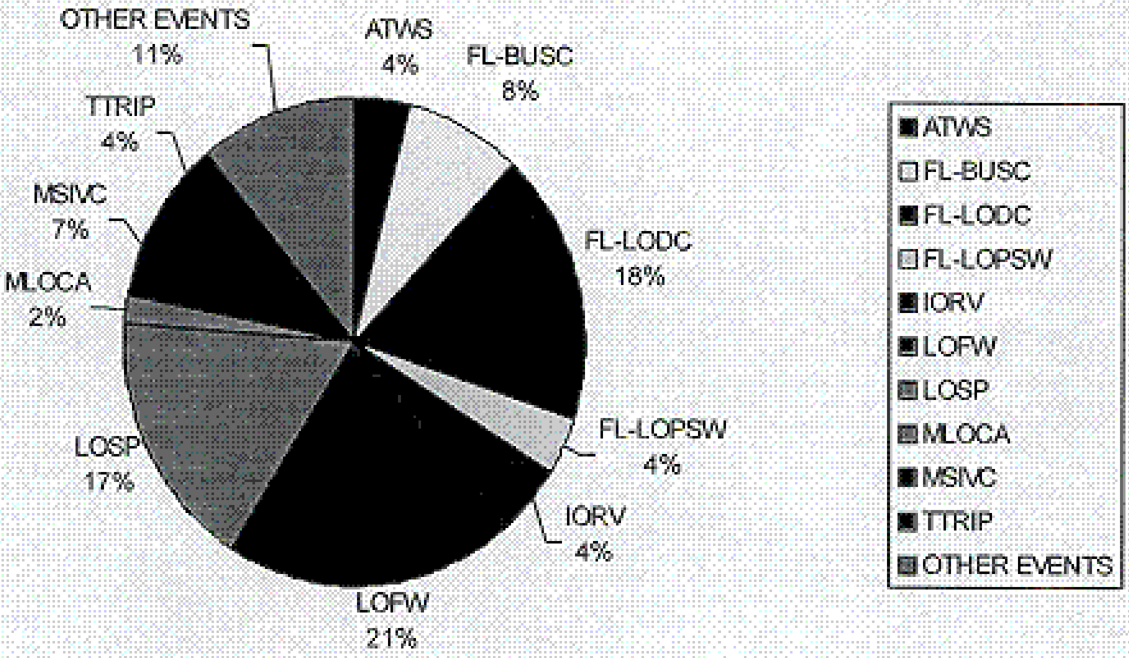


Figure 1.b-1 Initiator Distribution Hatch Unit CDF

C-1

Initiator Distribution Hatch Unit 1 LERF

EVENT	%	CDF	EVENT DESCRIPTION
ATWS	0.85	2.28E-08	Anticipated Transient Without Scram
FL-BUSC	16.3	4.38E-07	Loss of 600VAC Bus C
FL-LODC	2.6	7.04E-08	Loss of Station Battery A
FL-LOPSW	3.6	9.68E08	Loss of Plant Service Water
LOFW	3.2	8.73E-08	Loss of Feedwater
LOSP	49.6	1.34E-06	Loss of Offsite Power
MSIVC	4.5	1.20E-07	MSIV Closure
SCRAM	4.6	1.24E-07	Reactor Scram
TTRIP	5.6	1.50E-07	Turbine Trip
VSEQ	3.2	8.62E-08	HI/LO Pipe System Interface Breaks
OTHER EVENTS	5.95	1.60E-07	Various System Line Breaks Outside Containment
TOTALS	100	2.69E-06	

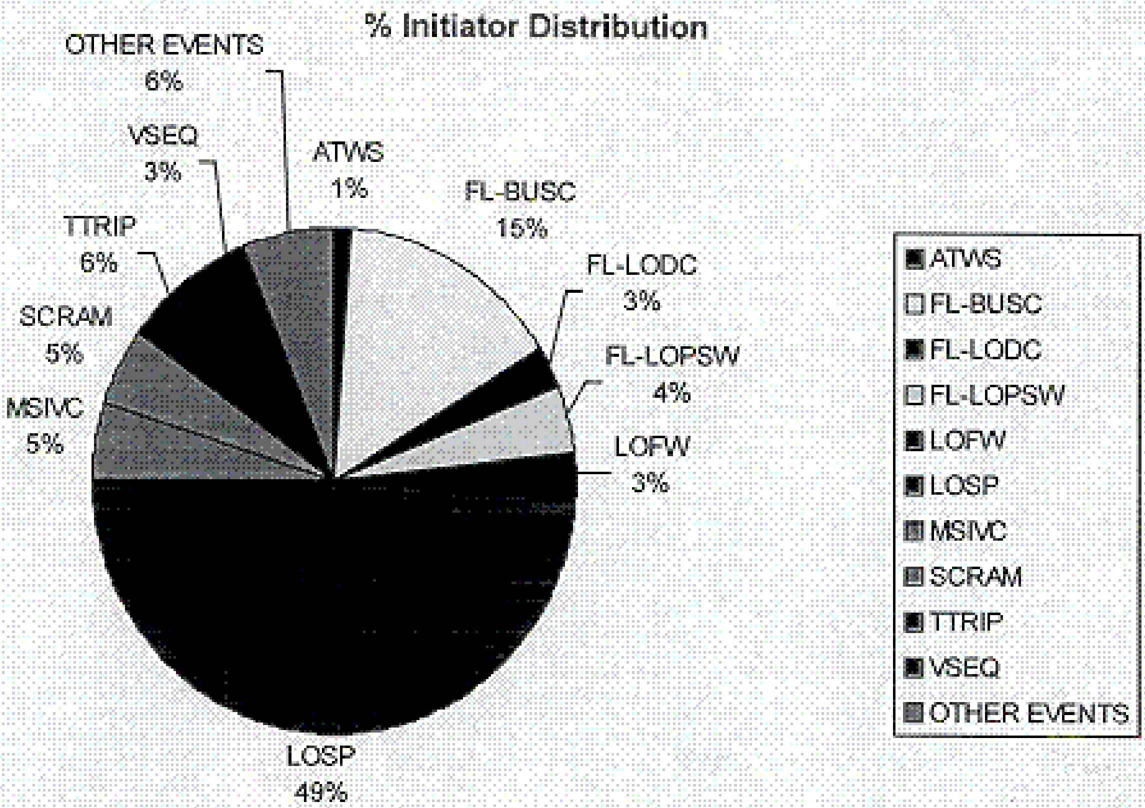


Figure 1.b-2 Initiator Distribution Hatch Unit 1 LERF

C-2

c. **"A listing of the dominant Level 1 accident sequences including the sequence logic in terms of event tree top events, and a mapping of these sequences to the Level 2 release categories. Provide sufficient supporting material to allow an outside PRA reviewer to understand the sequences and mappings."**

The *PRA Conversion Accident Sequence Analysis Notebook* (Attachment 1) provides a complete sequence description for the PSA model covering both Level 1 and Level 2 trees. The *PRA Conversion Project Model Integration and Quantification Work Package* (Attachment 2) describes how the process was performed. In addition, Attachment 2, section 4.2.9, identifies how the core damage sequences are integrated into the containment event tree (CET) which is used to obtain LERF information.

d. **"A characterization of the major differences in the core damage frequency and large release frequency contributors from those reported in the IPE, and the reasons for these differences."**

CDF has decreased with the CAFTA model as opposed to what it was under RISKMAN. For the Unit 1 RISKMAN model CDF was 2.2E-05 as opposed to 1.62E-05 for CAFTA. The LERF value for the Unit 1 RISKMAN model was 3.5E-06 as opposed to 2.7E-06 for CAFTA.

The percentage change for these numbers, while perhaps significant for specific applications, is not unexpected for a change in model types. More detail was added with regard to plant equipment support features in the CAFTA model as opposed to the RISKMAN model. This is particularly evident in the electric plant modeling. Fault trees for the RISKMAN model were quantified at as low a cutoff as possible. Split fractions (devices used to propagate failure of functions and equipment through the event trees) were then calculated from these cutsets. As a result, the split fractions accounted for as much failure probability as possible. The event trees were then run at much higher cutoffs for overall model quantification speed considerations.

There are no split fractions in CAFTA; instead, cutsets are propagated all the way through the model at a single cutoff value. An analysis is limited by the number of cutsets that can be saved and the run time of the model. Cutsets which miss the cutoff are truncated while the rest are summed for an overall CDF. The conservatism of the split fractions applied to the event tree model, as compared to the cutoff frequency of the CAFTA model cutsets, largely accounts for the difference in overall CDF and LERF frequencies.

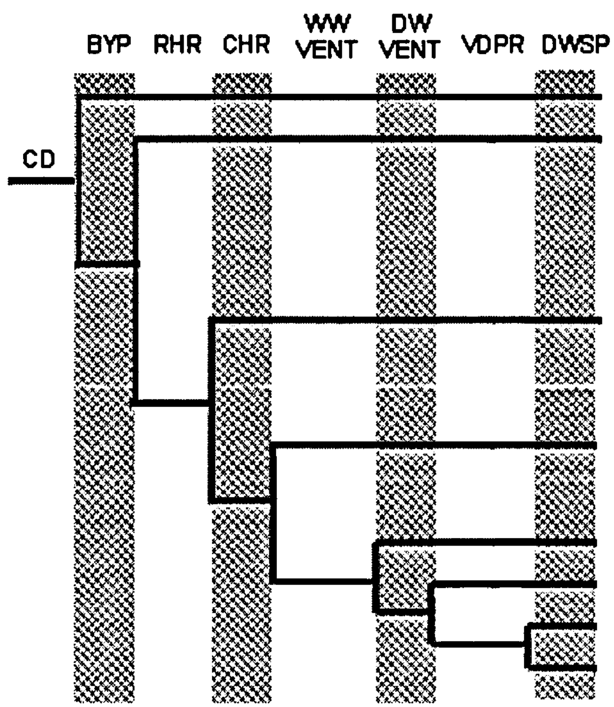
The LERF model is based on a CET developed by Fauske and Associates (see Hatch Containment Event Tree, figure 1.d-1). This tree was used to construct the LERF model for the Hatch PSA. Details of this construction are described in Attachment 2 with Attachment 1 describing the core damage model interface. Comparison of this tree to the IPE version shows a great simplification. Due to the direction of the nuclear PSA industry and the NRC, only large early releases were considered. The Hatch definition of LERF is as follows: A rapid, unscrubbed release of airborne aerosols within 6 hours of the time between vessel failure and containment failure. In

keeping with the screening criteria used in the IPE, large release is essentially category D or greater than 10 percent volatile fission products. In select cases, due to the closeness of the values, a category C is placed into the LERF category as well. LERF is also considered for direct primary containment unscrubbed venting and containment bypass scenarios.

The RISKMAN LERF model was based on a much more detailed CET but produced only four LERF sequences. These sequences (see Table 7.1-1), in addition to the other 10 non-LERF sequences presented in the IPE documentation, were used for Modular Accident Analysis Program (MAAP) code source term calculations. The same LERF sequences plus one extra (a new sequence formulated for the CAFTA model) are evaluated in the present PSA model. Despite support feature additions, the overall containment failure evaluation of the model cutsets showed that the same 14 sequences are still valid. The extra sequence (Sequence 15) identifies venting directly from the drywell after vessel failure. Comparing sequence frequency data between models for Sequences 2, 4, 5, and 11 show the following differences.

Sequence 2 describes the station blackout (SBO) case and has an IPE frequency of occurrence in the 1E-07 range. Due to the high level of detail used in binning RISKMAN model sequences in the IPE, intermediate release information which pertains to release within 24 hours was not included with those considered to be LERF. These sequences account for the increase in occurrence frequency because the new CAFTA model does not distinguish, but instead, accounts for intermediate and early release in its LERF category. This discussion holds for Sequences 4 and 11 as well. Sequence 5, which is containment bypass, uses a more detailed model for the various high energy line breaks outside containment, which is more conservative than what was previously used in the IPE model. Sequence 15 being a new sequence has no IPE references.

Hatch Containment Event Tree



END STATE	ANALYZED SEQUENCE	RELEASE CAT	LERF	ΔT
CB	5	D	YES	0
OPD	4	D	YES	0
	11	D	YES	0
OPW	3	B	NO	0
	10	C	NO	0
CN	1	A	NO	0
	7	A	NO	0
	8	A	NO	
VW	6	A	NO	0
	13	A	NO	0
VD	15	D	YES	0
OT	9	B	NO	17
OPD	12	B	NO	12
OT	2	C	YES	5

Figure 1.d-1 Hatch Containment Event Tree

e. **"A list of key equipment failures and human actions that dominate CDF and large release frequency, and the results of any supporting importance analyses (e.g., using Fussell-Vesely and/or Risk Reduction importance measures) indicating those equipment failures and human actions having greatest potential worth for reducing risk at Hatch."**

The *Basic Event Importance Report* (Attachment 3) provides a basic event importance report with event descriptions. *The Top 100 CDF Cutsets and Top 100 LERF Cutsets* are provided in Attachments 4 and 5, respectively. These documents provide the details of the key equipment failures and human actions that dominate the CDF and LERF and have the greatest potential for reducing risk at Plant Hatch.

2. **"Studies at other commercial nuclear power plants have shown that external events can be the dominating contributors to the overall core damage frequency and overall risk to the public. However, only two SAMA candidates for Hatch appear to involve external events and two other candidates address internal flooding concerns. Please discuss how plant-specific external event insights were considered in the SAMA identification process. Also, for those SAMAs intended primarily for internal events, describe how any added benefits in external events were considered in developing risk reduction estimates."**

A review of Plant Hatch IPEEE and NUREG-1560, *Individual Plant Examination (IPE) Program: Perspective on Reactor Safety and Plant Performance* was performed and the following SAMAs were identified that addressed external events during Phase I of the SAMA evaluation:

No.	SAMA ID	SAMA Title
1.	SAMA 68	Develop a severe weather procedure
2.	SAMA 75	Bury offsite power lines
3.	SAMA 114	Increase seismic ruggedness of plant components

SAMAs 68 and 114 have already been implemented at Plant Hatch. SAMA 75 did not pass the initial screening criteria to move into Phase II.

Plant specific insights were considered in our review of the Individual Plant Examination for External Events (IPEEE) and related to seismic events, internal fires, high winds, external floods, transportation and nearby facility accidents and other events. The review of the Plant Hatch IPEEE results indicated that external events are not dominating contributors to the overall CDF and overall risk to the public as discussed in the following paragraphs.

A seismic margins analysis was performed for Plant Hatch which showed all equipment within the IPEEE-seismic scope to possess a high-confidence low-probability of failure (HCLPF) capacity of at least 0.3 g peak ground acceleration (PGA). This information shows that seismic concerns are a very low threat to core damage in the Hatch PSA.

CDF contribution due to high winds, including tornadoes, was concluded to be less than 1E-06 per year and, as a consequence, considered to be an insignificant contribution to CDF.

External flooding affects on CDF was estimated in the IPEEE Report for Plant Hatch to be less than 1E-08 per year, which also is an insignificant contribution.

Internal fires, examined in the IPEEE, produced a conservatively modeled CDF contribution of 7.5E-06 per year for Unit 1 and 5.4E-06 per year for Unit 2. The fire analysis for Hatch is a PSA performed with a modified version of the IPE RISKMAN model.

No other external (or special internal) events were identified that posed any significant threat of severe accident at Plant Hatch. All the information described above is addressed in the Hatch IPEEE submittal to the NRC.

Georgia Power Company's final response (dated January 26, 1996) to Generic Letter 88-20, Supplement 4 provides the results of the IPEEE evaluations for Hatch Units 1 and 2. The Summary and Conclusion of the evaluation states:

The major finding from this examination is that Plant Hatch has no fundamental weaknesses or vulnerabilities to severe accident risk in regard to the external events related to seismic, fire, high winds, floods, transportation and nearby facility accidents, and other external hazards.

Therefore, the Plant Hatch IPEEE did not identify weaknesses or vulnerabilities due to external events and it demonstrated that external events are not dominating contributors to the overall core damage frequency and overall risk to the public.

3. **"It is not apparent that insights from the plant-specific risk study have been used to identify potential means of further reducing the risk at Hatch. For example, based on the IPE, battery depletion and main steam isolation valve closure events are important contributors to loss of high-pressure injection, yet neither of these contributors are addressed by SAMAs. There appear to be numerous other plant-specific insights that were not addressed in the SAMA submittal. In this regard, please provide:"**

   a. **"A discussion of the extent that the above plant-specific risk insights were used to identify potential SAMAs. If plant-specific insights were not considered, justify how the SAMA analysis can be considered to have identified "those SAMA candidates that have the most potential for reducing CDF and person-**

rem risk" at Hatch, as stated in Section 1 of Attachment F to Appendix D of the Environmental Report."

The Hatch IPE and IPEEE were used to develop the plant-specific insights for the ER SAMA candidates. This was done with a focus on providing for increased reliability for existing mitigation systems or alternates to existing mitigation systems. However, as described in answer to Question 2, Hatch IPEEE concluded that Plant Hatch has no fundamental weaknesses or vulnerabilities to severe accident risk in regard to the external events related to seismic, fire, high winds, floods, transportation and nearby facility accidents, and other external hazards. The SAMA candidates developed as a result of Hatch IPE are identified in table 14a which is being provided in response to Question 14.a. As indicated in answer to Question 3.b, procedural changes and plant modifications identified by Hatch IPE have already been implemented.

Also, Severe Accident Mitigation Design Alternatives for Limerick, the SAMAs developed as a result of the Watts Bar IPE review, NUREG-1560, NUREG-1437, NUREG-0498, NUREG-1462, and other references listed in table 14a in response to Question 14a were considered while developing the SAMA candidates for Plant Hatch.

One of the examples that the RAI notes in particular is battery depletion effects on loss of high-pressure injection. A number of SAMAs address improving station battery reliability and extending power availability. These Phase I SAMAs are as follows:

SAMA ID Number	Title
57	Provide Additional DC Battery Capacity
58	Use Fuel Cells Instead of Lead-Acid Batteries
61	Incorporate An Alternate Battery Charging Capability
62	Increase/Improve DC Bus Load Shedding
63	Replace Existing Batteries with More Reliable Ones

The above SAMAs were evaluated for their cost benefit based upon maximum risk reduction.

The second example that the RAI notes in particular is the loss of high-pressure injection caused by main steam isolation valve (MSIV) closure. Phase 1 SAMA 107, specifically addresses installation of a 50% capacity motor-driven pump to provide a redundant and independent (i.e., not dependent upon steam for motive power) source of high-pressure coolant injection.

b. **"A description of potential design enhancements identified through the IPE and follow-on studies and the disposition/status of these items. For those that have not been implemented, provide an assessment of them within the context of SAMAs."**

The procedural changes and design modifications identified through the Hatch IPE process have already been implemented. These include:

1) Installation of a hardened containment vent to ensure the design adequately addressed the dominant loss of decay heat removal (DHR).

2) removal of the common plant service water (PSW) discharge valve in Unit 1 eliminated a significant contributor to loss of all PSW.

3) Control building (CB) HVAC duct modifications and changes in procedures to allow continued operation of electrical equipment in the CB following loss of HVAC.

4) Procedural changes to initiate the purge mode of the main control room (MCR) cooling on loss of control room chillers.

5) Intake structure ventilation system modifications to the to ensure that the failure in the HVAC system will not lead to a failure of the PSW or residual heat removal service water (RHRSW) pumps.

6) Procedural changes to allow tripping of RHR and core spray (CS) pumps in emergency core cooling system (ECCS) rooms to allow continued operation of one pump with loss of room cooling.

7) Procedural changes to allow cross connection of PSW cooling water to RHRSW pump motors, with loss of one division of PSW.

8) Modification to allow the swing chiller compressor for control room HVAC to be powered by either division of electrical power.

9) SBO rule modifications including replacing station service battery chargers and enhancing procedures dealing with loss of ventilation.

4. **"The offsite risk estimate for Hatch appears to be based on only five of the 15 release classes/sequences in the updated Level 2 PRA. Although the five sequences appear to include the large early release sequences, several additional sequences have either substantially larger release frequencies or only slightly lower release fractions (e.g., Sequence 12). The risk associated with the other 10 sequences should also be included in order to provide a complete picture of risk. Please provide the frequency and consequences (person-rem and economic) for all 15 sequences. If**

these additional sequences impact the results by more than about 10 percent, please revise the SAMA benefit evaluations."

The original RISKMAN model used for the IPE, which was submitted to the NRC, evaluated the large early release sequences as well as others to satisfy the requirements of Generic Letter 88-20. As discussed in the response to Question 3b, the design enhancements and procedural changes identified through the IPE process have already been implemented at Plant Hatch. (Responses to Questions 1c and 1d provide the details on the conversion from the RISKMAN model to our current PSA model.) The current PSA was refined during the conversion process to focus on LERF as the accepted measure for containment performance. Input for this decision came from Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis* and widely accepted industry practices. Regulatory Guide 1.174 and industry experience indicate that LERF is the principal accepted measure for containment performance. Subsequent to meeting the requirements of Generic Letter 88-20, there was no longer a need to include the detail of intermediate and late releases to measure containment performance in the Level 2 model. Therefore, this LERF-based approach was adopted in the development of the new CAFTA Level 2 model for Plant Hatch.

Plant Hatch does not currently have a Level 3 model. The MACCS2 model used for the ER SAMA analysis was developed specifically to perform the required evaluation. There is currently very little guidance on what should go into this type model; therefore, the license renewal evaluation was performed around the presently accepted Level 2 modeled information. Source term information is available for the non-LERF sequences to evaluate whether or not they contribute to LERF, but occurrence frequency was not generally modeled for these sequences and is not needed to define Level 2 output. Without occurrence frequency information, it is not possible to adequately address the contributions from all of the other sequences, which address late containment failure with our current model. SNC considers that LERF sequences collect the majority of the source term based on information shown in table 7.1-1 provided in the response to Question 7, Releases for Analyzed Sequences, and as illustrated by quantifying the results for two of the largest releases of the non-LERF sequences (12 and 14). The frequency and consequences for Sequences 12 (high-pressure transient with loss of containment heat removal (CHR)) and 14 (SBO with containment isolation failure) are:

Sequence	Frequency	Population dose, 0-50 mile (risk in person-rem)	Total economic costs, 0-50 miles (risk in $)
12	2.0E-7	0.10	110.00
14	3.1E-9	0.0008	1.10
(12+14) Percent of LERF		3.4%	1.3%

SNC does not consider modeling of the remaining eight sequences to be necessary to adequately address the risk of the SAMA candidates evaluated by the ER.

5. **"Please provide a breakdown of leading contributors to dose consequences (e.g., containment bypass, early containment failure, late containment failure, intact containment). Results may be presented in either a table or figure that provides general risk insights-percent contributions to the population dose."**

Release mode contributors were classified according to the time between vessel failure and containment failure; 6 hours or less is considered early containment failure. Sequences 12 and 14 are used to represent the late containment failure mode; these sequences result in two of the largest releases of this failure mode. The mean (expected) annual 0-50 mile population dose risk by release mode contributor is:

Release Mode	Sequence	Population Dose (Person-rem)	Contribution (%)
Containment bypass	5 (Loss-of-coolant accident (LOCA) Outside Containment)	0.17	5.44
Early containment failure	2 (SBO), 4 (Loss of loss of containment heat removal (CHR)/Drywell Failure), 11 (Anticipated transient without scram (ATWS) Drywell Failure)	2.8	91.21
Late containment failure	12 (High pressure transient w/loss of CHR), 14 (SBO w/containment isolation failure)	0.10	3.32
Intact containment (venting)	15 (High pressure transient w/Venting)	0.0009	0.03
TOTAL		**3.1**	**100**

6. **"It is our understanding that release fractions as determined in a report by FAI, Inc. entitled, "Level II Process Plant Hatch," (FAI/98088, March 1999) were used in this submittal. That report states that release fractions were estimated using modular accident analysis program (MAAP) calculations for representative events in each containment event tree endstate. Please clarify what version of MAAP was used for these calculations. Please provide release fractions for radionuclide groups (not only noble gases, but also I, Cs, Te, Sr, Ru, La, Ba, and Ce) so that results can**

be compared with values predicted in NUREG-1150 for the Peach Bottom plant (also a BWR-4 in a Mark I containment)."

MAAP 3.0B BWR Revision 10 was used to generate radionuclide release fractions as well as to provide containment analysis details for the present Hatch PSA models.

The LERF release fractions used in the analysis were:

Sequence	Xe/Kr	I	Cs	Te	Sr	Ru	La	Ce	Ba
2	1	0.1	0.82	0.06	0	0	0	0.003	0.01
4	1	0.35	0.39	0.11	0.005	0	0	0.004	0.001
5	1	0.92	0.93	0.16	0.018	0.13	0	0.006	0.02
11	1	0.08	0.13	0.13	0.01	0	0.002	0.008	0.008
15	0.83	0.31	0.24	0.065	0.008	0.003	0	0.005	0.003

7. **"Because it is a dominant contributor to plant risk, please discuss differences in the MAAP results presented in the IPE station blackout (SBO) sequence (Sequence 2 in tables 4.7-9 and 4.7-10) and the SBO sequence from the fail as is (FAI) report. Differences include timing of key events and release fractions. Also, please clarify why the differences (timing and frequencies) for Unit 1 and Unit 2 in the IPE don't exist in the FAI report. Finally, clarify why the source term bin 2 is release category D in the IPE, but release Category C in the current submittal."**

Notation for Question 7: FAI stands for Fauske and Associates, Inc.

(The reader should refer to table 7.1-1, sheets 1 through 7, for the following discussion.) Sequence 2, SBO, was subject to a MAAP code revision. Column 8.01, IPE, (table 7.1-1) relates to the original IPE results evaluated with MAAP 3.0B BWR Revision 8.01. Column 10, IPE, reflects IPE results recalculated with the present version, MAAP 3.0B BWR Revision 10. Revision 10 of MAAP removed the ability of the Automatic Depressurization System (ADS) to provide open relief valves beyond a certain drywell pressure, which precludes their operation. This revision ultimately prolonged the operation of reactor core isolation cooling (RCIC) thus increasing the time to core uncovery during a SBO. This modification was not in MAAP Rev 8.01 and, as a result, provided different core uncovery timing numbers.

Column 10, UPRATE, provides the MAAP results for Sequence 2 using MAAP 3.0B BWR Revision 10. The numbers in Column 10 are effected by the overall increased power of the uprate and its associated decay heat contribution. In addition, changes in certain instrumentation setpoints were also made in the MAAP parameter file, which had small effects on the results. The increased power of uprate resulted in a radionuclide release increase.

The FAI report for uprated Plant Hatch uses common Unit 1 and 2 sequence descriptions for MAAP modeling. The differences between units are accounted for in the MAAP

parameter files for each unit. The differences (such as setpoints) are not considered to be significant enough to warrant two sets of sequence descriptions. This is discussed in internal SNC calculations and was presented in this way for the extended power uprate application. For the purposes of the license renewal application, the Unit 1 Level 1 and 2 models were used. The differences, likewise, do not warrant two sets of evaluations. The FAI report, which describes the new CET model, is considered a supplement to existing documentation regarding the detail of model, construction, and phenomenological evaluation. It is not intended to be a comprehensive description.

The MAAP code revision of 8.01 to 10.0 discussed above resulted in a reduction in radionuclide release for Sequence 2 that made it a category C release. Extended power uprate resulted in a release increase for Sequence 2, but not enough for it to be a category D. The current Hatch Level 2 model continues to consider this sequence as a LERF based on timing.

Table 7.1-1 (Sheet 1 of 7)

Releases for Analyzed Sequences

Code Version: MAAP 3.0B BWR Revision........	8.01	10	10
Parameter File (IPE or Uprate):	IPE	IPE	Uprate
Sequence No. 1			
Sequence Type : Medium LOCA			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	0.226	0.226	0.201
Time of Vessel Failure (h)	2.177	2.146	1.903
Time of Containment Failure (h)	—	—	—
Fraction of Zr Reacted in Vessel	0.2161	0.2192	0.2523
UO_2 in Pedestal (lbm)	90700.0	90742.2	97567.4
UO_2 in Drywell (lbm)	138000.0	138391.1	144527.0
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	2.5400	2.5718	2.5355
Volatile FP Release (%)	0.0054	0.0055	0.0036
Nonvolatile FP Release (%)	0.0005	0.0005	0.0005
Release Category	A	A	A

Sequence No. 2			
Sequence Type : SBO			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	4.935	5.039	5.326
Time of Vessel Failure (h)	8.528	8.244	8.118
Time of Containment Failure (h)	9.689	12.491	13.099
Fraction of Zr Reacted in Vessel	0.1934	0.1787	0.2273
UO_2 in Pedestal (lbm)	38200	28739.4	34372.1
UO_2 in Drywell (lbm)	191000	168798.3	180151.3
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	100	100	100
Volatile FP Release (%)	9.8800	6.8329	7.9651
Nonvolatile FP Release (%)	0.3023	0.0409	0.0979
Release Category	D	C	C

Table 7.1-1 (Sheet 2 of 7)

Releases for Analyzed Sequences

Code Version: MAAP 3.0B BWR Revision........	8.01	10	10
Parameter File (IPE or Uprate):	IPE	IPE	Uprate
Sequence No. 3 Sequence Type : Loss of CHR/ Torus Failure			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	31.392	20.386	20.058
Time of Vessel Failure (h)	37.053	24.891	23.912
Time of Containment Failure (h)	29.271	24.894	23.915
Fraction of Zr Reacted in Vessel	0.2024	0.1802	0.2242
UO_2 in Pedestal (lbm)	61300	44351.9	46960.8
UO_2 in Drywell (lbm)	73900	115739.1	117591.2
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	100	100	100
Volatile FP Release (%)	1.0480	0.5501	0.2865
Nonvolatile FP Release (%)	0.0100	0.0037	0.0020
Release Category	C	B	B

Sequence No. 4 Sequence Type : Loss of CHR/ Drywell Failure			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	31.379	20.386	20.058
Time of Vessel Failure (h)	36.891	24.891	23.912
Time of Containment Failure (h)	29.149	24.894	23.915
Fraction of Zr Reacted in Vessel	0.2007	0.1802	0.2242
UO_2 in Pedestal (lbm)	65400	45060.7	46405.7
UO_2 in Drywell (lbm)	68200	114384.8	117417.4
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	100	100	100
Volatile FP Release (%)	27.8260	29.7464	32.1777
Nonvolatile FP Release (%)	0.0455	0.0985	0.2138
Release Category	D	D	D

Table 7.1-1 (Sheet 3 of 7)

Releases for Analyzed Sequences

Code Version: MAAP 3.0B BWR Revision........	8.01	10	10
Parameter File (IPE or Uprate):	IPE	IPE	Uprate
Sequence No. 5 Sequence Type : LOCA Outside Containment			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	0.051	0.072	0.070
Time of Vessel Failure (h)	1.411	1.626	1.241
Time of Containment Failure (h)	—	—	—
Fraction of Zr Reacted in Vessel	0.0923	0.109	0.1449
UO_2 in Pedestal (lbm)	31700	31525.1	35418.4
UO_2 in Drywell (lbm)	197000	197595	206647.5
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	100	100	100
Volatile FP Release (%)	76.7690	75.7257	75.7777
Nonvolatile FP Release (%)	2.9780	6.0352	3.3441
Release Category	D	D	D

Sequence No. 6 Sequence Type : High Pressure Transient w/Venting			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	0.629	0.667	0.648
Time of Vessel Failure (h)	2.949	2.864	2.491
Time of Containment Failure (h)	—	—	—
Fraction of Zr Reacted in Vessel	0.1737	0.1632	0.2067
UO_2 in Pedestal (lbm)	94200	93414	101949.7
UO_2 in Drywell (lbm)	135000	135718.7	140146.0
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	100	100	100
Volatile FP Release (%)	0.0049	0.0033	0.0081
Nonvolatile FP Release (%)	0.0000	0.0000	0.0000
Release Category	A	A	A

Table 7.1-1 (Sheet 4 of 7)

Releases for Analyzed Sequences

Code Version: MAAP 3.0B BWR Revision........	8.01	10	10
Parameter File (IPE or Uprate):	IPE	IPE	Uprate
Sequence No. 7 Sequence Type : ATWS w/Injection Failure			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	0.078	0.079	0.069
Time of Vessel Failure (h)	1.657	1.573	1.259
Time of Containment Failure (h)	—	—	—
Fraction of Zr Reacted in Vessel	0.1398	0.1391	0.1795
UO_2 in Pedestal (lbm)	92700	92554	99400.8
UO_2 in Drywell (lbm)	136000	136576.1	142693.0
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	2.4000	2.3801	2.3934
Volatile FP Release (%)	0.0002	0.0002	0.0002
Nonvolatile FP Release (%)	0.0000	0.0000	0.0000
Release Category	A	A	A

Sequence No. 8 Sequence Type : Large LOCA			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	0.021	0.021	0.021
Time of Vessel Failure (h)	0.984	0.997	0.839
Time of Containment Failure (h)	—	—	—
Fraction of Zr Reacted in Vessel	0.0733	0.0755	0.0957
UO_2 in Pedestal (lbm)	93700	92856	98654.7
UO_2 in Drywell (lbm)	135000	136274.7	143438.6
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	2.0600	2.1967	2.2636
Volatile FP Release (%)	0.0019	0.0019	0.0017
Nonvolatile FP Release (%)	0.0000	0.0000	0.0000
Release Category	A	A	A

Table 7.1-1 (Sheet 5 of 7)

Releases for Analyzed Sequences

Code Version: MAAP 3.0B BWR Revision........	8.01	10	10
Parameter File (IPE or Uprate):	IPE	IPE	Uprate
Sequence No. 9			
Sequence Type : Low Pressure Transient			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	0.110	0.131	0.130
Time of Vessel Failure (h)	1.502	1.619	1.247
Time of Containment Failure (h)	14.794	13.912	18.323
Fraction of Zr Reacted in Vessel	0.0321	0.032	0.0401
UO_2 in Pedestal (lbm)	28400	28358.7	30419.5
UO_2 in Drywell (lbm)	201000	200774.9	211673.1
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	100	100	90.3484
Volatile FP Release (%)	2.0450	1.5420	0.8605
Nonvolatile FP Release (%)	0.0012	0.0028	0.0006
Release Category	C	C	B

Sequence No. 10			
Sequence Type : ATWS Torus Failure			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	0.930	0.706	0.618
Time of Vessel Failure (h)	3.167	2.347	1.978
Time of Containment Failure (h)	0.760	0.706	0.718
Fraction of Zr Reacted in Vessel	0.0972	0.0466	0.0742
UO_2 in Pedestal (lbm)	31700	31712.3	35127.7
UO_2 in Drywell (lbm)	197000	197396.8	206913.4
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	100	100	100
Volatile FP Release (%)	1.7550	2.4362	1.0652
Nonvolatile FP Release (%)	0.0613	0.0151	0.0164
Release Category	C	C	C

Table 7.1-1 (Sheet 6 of 7)

Releases for Analyzed Sequences

Code Version: MAAP 3.0B BWR Revision........	8.01	10	10
Parameter File (IPE or Uprate):	IPE	IPE	Uprate
Sequence No. 11 Sequence Type : ATWS Drywell Failure			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	0.940	0.706	0.618
Time of Vessel Failure (h)	3.216	2.330	1.972
Time of Containment Failure (h)	0.763	0.712	0.723
Fraction of Zr Reacted in Vessel	0.1035	0.0474	0.0701
UO_2 in Pedestal (lbm)	31100	32096.1	35089.3
UO_2 in Drywell (lbm)	198000	197003.5	206946.5
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	100	100	100
Volatile FP Release (%)	11. 4700	23.9213	12.6186
Nonvolatile FP Release (%)	0.2290	0.3461	0.3803
Release Category	D	D	D

Sequence No. 12 Sequence Type : High Press. Transient w/Loss of CHR			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	0.629	0.667	0.648
Time of Vessel Failure (h)	2.949	2.864	2.491
Time of Containment Failure (h)	15.061	15.092	14.484
Fraction of Zr Reacted in Vessel	0.1737	0.1632	0.2067
UO_2 in Pedestal (lbm)	45000	44799.2	42773.6
UO_2 in Drywell (lbm)	184000	184329.5	199317.7
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	100	100	100
Volatile FP Release (%)	5.3780	5.4914	6.6014
Nonvolatile FP Release (%)	0.1970	0.1966	0.2170
Release Category	C	C	C

Table 7.1-1 (Sheet 7 of 7)

Releases for Analyzed Sequences

Code Version: MAAP 3.0B BWR Revision........	8.01	10	10
Parameter File (IPE or Uprate):	IPE	IPE	Uprate
Sequence No. 13			
Sequence Type : Medium LOCA w/Venting			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	0.375	0.380	0.377
Time of Vessel Failure (h)	1.937	1.933	1.708
Time of Containment Failure (h)	—	—	—
Fraction of Zr Reacted in Vessel	0.1713	0.1749	0.2328
UO_2 in Pedestal (lbm)	76800	76937.9	99848.8
UO_2 in Drywell (lbm)	103000	102746.9	142247.4
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	100	100	100
Volatile FP Release (%)	0.0097	0.0106	0.0033
Nonvolatile FP Release (%)	0.0000	0.0000	0.0001
Release Category	A	A	A

Sequence No. 14			
Sequence Type : SBO w/CI			
CORE/CONTAINMENT RESPONSE			
Time of Core Uncovery (h)	4.970	4.519	5.332
Time of Vessel Failure (h)	8.563	7.670	8.134
Time of Containment Failure (h)	—	—	39.773
Fraction of Zr Reacted in Vessel	0.194	0.181	0.2258
UO_2 in Pedestal (lbm)	39000	29754.8	34903.4
UO_2 in Drywell (lbm)	190000	167780.9	179416.5
ENVIRONMENTAL RELEASE @40 h			
Noble Release (%)	100	100	100
Volatile FP Release (%)	0.7130	1.0805	1.3932
Nonvolatile FP Release (%)	0.0070	0.0057	0.0033
Release Category	B	C	C

8. **"The Hatch model assumes that drywell venting would only be used if the wetwell vent is unavailable, and indicates that the frequency of drywell venting would be 9E-10/year (Sequence 15). This assumption is more restrictive than the generic guidance provided in the BWROG Emergency Procedure and Severe Accident Guidelines, which permits the use of the drywell vent for pressure and hydrogen control, independent of the wetwell vent. The model also does not appear to account for drywell venting to facilitate containment flooding and reactor pressure vessel injection, in accordance with RC/F-1 through -6 of the severe accident guidelines. Thus, the Hatch model may understate the offsite risk associated with drywell venting. Please describe the basis for the drywell venting assumption and justify that the assumption is consistent with the plant-specific guidance on containment venting at Hatch. Also, describe the risk associated with drywell venting to facilitate containment flooding and reactor pressure vessel injection, and how it is reflected in the Hatch model. If the PRA models/assumptions are not consistent with plant-specific procedures and guidance, please provide a revised estimate of the risk posed by drywell venting, and a value/impact analysis of modifying the procedures/guidance to further limit drywell venting."**

The basis for drywell venting, Sequence 15, is the Plant Hatch Emergency Operating Procedures (EOPs) and the Severe Accident Guidelines (SAGs). In Plant Hatch EOPs, the preferred vent path for containment pressure and /or hydrogen control is via the suppression chamber (torus) -- unless the torus vent capability has failed or torus water level is at or above 300 inches. Under the SAGs scenario, it is assumed that the core is exiting a failed reactor vessel. However, since there is no water source for debris coverage, venting for primary containment flooding is not a consideration. Once the core debris has breached the reactor vessel, reactor vessel venting is not allowed by the SAGs.

Reactor vessel venting may be used in the attempt to help the external or internal water supply cover the core, although in-vessel core considerations are not addressed by Sequence 15. There are sequences evaluated for radioactive release for the Level 2 model that do not have an associated containment failure. These sequences are potential candidates for direct (unscrubbed) release due to venting for flooding or from reactor vessel venting. Sequences 1, 7, and 8 are described in the response to Question 7 (table 7.1-1). In these cases, water is available for debris cooling, and there are mechanisms available for removing the containment heat load. The associated EOP/SAGs assume conditions such that the inability to meet reactor vessel level requirements has lead to primary containment flooding and, thus, entry into the SAG. Event timing is such that for the LOCAs (Sequences 1 and 8), reactor vessel venting may possibly be performed. For Sequence 7, reactor vessel venting would serve no useful purpose because water from the primary containment flooding source would not have a path to the core until vessel failure, at which time SAG guidance would prohibit reactor pressure vessel (RPV) venting.

When reactor vessel venting is performed, the main condenser serves as the hold-up volume. If circulating water is available, the low flowrate steam is slowly condensed. If circulating water is not available, the main condenser vents are opened, and the steam partially condenses with the balance vented to the turbine building. Turbine building

ventilation provides some filtration, then the noncondensables are passed to the reactor building vent stack. The flowrate associated with this path is also small, and the holdup contribution from the various components encountered along the path is very significant. This release, which only occurs for a brief time before the vessel fails in Sequences 1 and 8, is small because the pertinent radionuclide concentration is small, and the pathway retention is high. This release, while early, is not considered large.

Any necessary venting via the unscrubbed drywell pathway to allow primary containment flooding would be fairly late in the scenario. Initially, the venting would be from the torus until the level reached 300 inches or this vent pathway failed. Venting from the unscrubbed pathway would be intermittent—only enough to maintain containment pressure within limits and prevent the flooding source from being placed in a shutoff head condition. Within time, if successful, the debris would be covered and the consequences of release would be reduced. It is believed that the existing LERF categories and analyses bound this release condition.

Sequences 1, 7, and 8 do not produce large releases (note in table 7.1-1 that they all are category A releases). These sequences are not considered LERF sequences and are not specifically modeled for Level 2 considerations.

Thus the PRA model and assumptions are consistent with Plant Hatch specific guidance for drywell venting and the offsite risk associated with drywell venting is not understated.

9. **"The SAMA submittal indicates that the population growth rate used in the projection out to 2030 was assumed to be the same as that projected between 1990 and 2000. Please provide this assumed growth rate. The second paragraph on Page F-3 indicates that Reference 2 (NUREG-1150) lists 1990 population data by county and projected county population growth rates. This reference citation appears incorrect. Please provide the correct reference."**

a) Population projections for 2000 were determined using the growth rate between 1990 and 2000. Population projections for 2020, 2030, and 2040 were determined using the growth rate between 2000 and 2010. The annual growth rates (i.e., population in year I+1/population in year I), by sector and distance, beginning in the year 2000 are described in table 9a.

b) The reference citation was incorrect and should be:

M. Sik, Georgia Governor's Office of Planning and Budget, Atlanta Georgia, personal communications with J.B. Hovey, Tetra Tech NUS, Inc., Aiken, South Carolina, "1980 and 1990 Census Counts and 2000 and 2010 Population Projections, 1997 Estimates," April 2, 1999.

Table 9a Annual Growth Rates

Sector	0-1 mile	1-2 miles	2-3 miles	3-4 miles	4-5 miles	5-10 miles	10-20 miles	20-30 miles	30-40 miles	40-50 miles	50-mile total
N	--	1.008	1.009	--	1.009	1.009	1.009	1.008	1.006	1.007	1.008
NNE	--	1.000	--	--	1.012	1.009	1.009	1.007	1.004	1.007	1.006
NE	--	--	--	1.010	1.009	1.009	1.008	1.007	1.007	1.014	1.012
ENE	--	--	--	--	1.000	1.009	1.007	1.007	1.008	1.014	1.010
E	--	--	--	--	1.007	1.007	1.007	1.006	1.010	1.016	1.015
ESE	--	--	1.007	--	--	1.007	1.007	1.006	1.003	1.006	1.005
SE	--	--	1.008	1.007	1.007	1.007	1.007	1.008	1.008	1.008	1.008
SSE	--	--	1.007	1.007	1.007	1.007	1.007	1.007	1.007	1.007	1.007
S	--	1.008	1.007	1.007	1.007	1.007	1.007	1.003	1.006	1.009	1.008
SSW	--	1.006	1.007	1.007	1.008	1.007	1.007	1.003	1.003	1.006	1.004
SW	--	1.007	1.007	1.006	1.009	1.009	1.009	1.010	1.011	1.010	1.010
WSW	--	--	1.008	--	1.010	1.009	1.009	1.009	1.010	1.008	1.009
W	--	1.007	--	1.008	--	1.009	1.008	1.009	1.009	1.007	1.008
WNW	--	--	--	1.007	--	1.007	1.005	1.006	1.007	1.003	1.006
NW	--	--	--	1.010	1.008	1.008	1.006	1.005	1.009	1.010	1.008
NNW	--	1.000	1.009	1.009	1.009	1.009	1.007	1.006	1.005	1.008	1.006
TOTAL	--	1.007	1.008	1.008	1.008	1.008	1.008	1.006	1.007	1.011	1.009

The "total" column and row are calculated by dividing the total populations in sequential years.
"-" indicates no population within that sector.

10. **"Please provide an explanation of: (1) how the risk would change if population projections were based on the end of the renewal period (2034 and 2038 for Units 1 and 2) rather than 2030; and (2) what, if any, transient population considerations were factored into the risk determination."**

a) The growth rates indicated in the response to Question 9 were applied to the 2030 population used in the Level 3 analysis to obtain 2034 and 2038 population distributions. They were then used in an exposure analysis analogous to that performed for the Level 3 study. The resulting risk was seen to increase by ~4% for 2034 and ~8% for 2038, relative to 2030. Given that the events and release frequencies are independent of population, and that the risk is insensitive to evacuation assumptions (see response to Question 12), then the risk will be roughly proportional to local population increases (weighted by exposure at those locales). The increase in risk calculated is, indeed, roughly proportional to the "total (over the 50-mile radius)" population growth rate (see response to Question 9).

b) Transient populations were not considered in the risk determination due to the rural setting of Plant Hatch and the small assumed transient population within 50 miles of the site.

11. **"The SAMA submittal does not provide sufficient detail about the release sequences to readily determine if the times specified for declaring a general emergency are appropriate. Please provide this information."**

Emergency Planning considerations were specifically excluded from license renewal in the rulemaking process (55FR 29053). The times specified for declaring a general emergency are appropriate and are consistent with our emergency plan. The answer provided below provides more detail concerning timing of the release sequences in our current PSA model.

Times (from scram) of core uncovery, vessel failure, and containment failure for each of the LERF sequences are as follows:

Sequence No	2	4	5	11	15
Core Uncovery (h)	5.326	20.058	0.070	0.618	0.647
Vessel Failure (h)	8.118	23.912	1.241	1.972	2.490
Containment Failure (h)	13.099	23.915	---	0.723	---
General Emergency Declaration (h)	1.000	20.058	0.070	0.25	0.25

12. **"Justify why evacuation times based on the current evacuation study would remain valid for the end of the renewal period (2034 and 2038), given the projected increase in population."**

Emergency Planning considerations with a specific emphasis on population increases were specifically excluded from license renewal in the rulemaking process (55FR 29053). Plant Hatch does not have a current level 3 model. The MACC2S was developed specifically to address the required evaluation for the ER. This analysis is a snap shot in time. SNC does not consider this an appropriate question for license renewal. However, the following justification is provided to clarify the lack of impact the evacuation parameter assumptions have on our analysis.

The risks for the Hatch site are insensitive to evacuation parameter assumptions because the 10-mile radius emergency planning zone (EPZ) is located in a rural area of low population (the 0-10 mile population is 2% of the 0-50 mile population). Furthermore, conservative assumptions were made in choosing these parameters. For example, it was assumed that the entire population within the EPZ would evacuate at the speed of the slowest subpopulation (special need persons under adverse conditions). This speed is approximately half of the evacuation speed indicated for the general population (under adverse conditions) in the current evacuation study.

To illustrate the insensitivity of the evacuation parameter assumptions, the conditional population dose was recalculated for the LERF sequences with the evacuation speed arbitrarily set at one-half of that used in the Hatch Level 3 analysis:

Sequence	2	4	5	11	15
Dose-Conditional (Person-rem)	1.06E+06	1.02E+06	1.16E+06	7.03E+05	1.13E+06

When compared to the Level 3 model results described in the response to Question 14.c the differences are inconsequential.

13. **"Please provide a discussion of why 1997 meteorological data were used and justify why this can be considered a representative year."**

The 1997 meteorological data that was used in the SAMA analysis is a representative year for the Plant Hatch region. We have run ground level X/Qs (concentration/ source term) for 1995 through 1997 and for each individual year. The results indicate that 1997 had the highest X/Q values of the 3 years. Therefore, the results indicate that 1997 was a conservative set of data in comparison to the 3-year period of 1995 through 1997. Table 13 provides the results of the comparison.

TABLE 13. 1995-1997 X/Q COMPARISON												
Direction	1995-1997			1995			1996			1997		
	800m	3200m	8047m	800m	3200m	8047m	800m	3200m	8047m	800m	3200m	8047m
N	1.96E-5	1.11E-6	2.35E-7	1.75E-5	9.89E-7	2.09E-7	2.02E-5	1.14E-6	2.43E-7	2.10E-5	1.21E-6	2.56E-7
NNE	3.06E-5	1.73E-6	3.73E-7	2.19E-5	1.22E-6	2.62E-7	3.44E-5	1.93E-6	4.20E-7	3.56E-5	2.04E-6	4.39E-7
NE	3.51E-5	1.98E-6	4.31E-7	2.65E-5	1.49E-6	3.22E-7	3.81E-5	2.15E-6	4.68E-7	4.09E-5	2.31E-6	5.03E-7
ENE	3.19E-5	1.78E-6	3.94E-7	2.28E-5	1.28E-6	2.78E-7	3.29E-5	1.82E-6	4.05E-7	4.03E-5	2.24E-6	4.99E-7
E	2.31E-5	1.29E-6	2.80E-7	1.84E-5	1.05E-6	2.24E-7	2.31E-5	1.28E-6	2.79E-7	2.79E-5	1.53E-6	3.38E-7
ESE	1.48E-5	8.20E-7	1.72E-7	1.25E-5	6.99E-7	1.43E-7	1.55E-5	8.52E-7	1.81E-7	1.63E-5	9.11E-7	1.93E-7
SE	1.21E-5	6.67E-7	1.37E-7	1.14E-5	6.28E-7	1.28E-7	9.94E-6	5.44E-7	1.11E-7	1.50E-5	8.32E-7	1.73E-7
SSE	8.34E-6	4.65E-7	9.28E-8	8.76E-6	4.80E-7	9.58E-8	7.92E-6	4.42E-7	9.07E-8	8.34E-6	4.66E-7	9.17E-8
S	5.65E-6	3.08E-7	6.16E-8	6.18E-6	3.37E-7	6.70E-8	4.80E-6	2.56E-7	5.19E-8	5.97E-6	3.22E-7	6.58E-8
SSW	6.74E-6	3.72E-7	7.53E-8	6.40E-6	3.52E-7	7.11E-8	6.92E-6	3.88E-7	7.94E-8	6.89E-6	3.76E-7	7.53E-8
SW	1.58E-5	8.87E-7	1.76E-7	1.57E-5	8.69E-7	1.71E-7	1.41E-5	7.93E-7	1.59E-7	1.76E-5	9.99E-7	1.99E-7
WSW	1.39E-5	7.92E-7	1.58E-7	1.49E-5	8.41E-7	1.66E-7	1.24E-5	7.09E-7	1.44E-7	1.43E-5	8.25E-7	1.64E-7
W	1.04E-5	5.97E-7	1.22E-7	1.10E-5	6.28E-7	1.25E-7	8.03E-6	4.63E-7	9.49E-8	1.22E-5	7.01E-7	1.45E-7
WNW	1.05E-5	6.10E-7	1.25E-7	1.15E-5	6.66E-7	1.35E-7	1.07E-5	6.19E-7	1.29E-7	9.28E-5	5.43E-7	1.10E-7
NW	1.27E-5	7.27E-7	1.48E-7	1.19E-5	6.73E-7	1.36E-7	1.24E-5	7.21E-7	1.47E-7	1.36E-5	7.87E-7	1.60E-7
NNW	1.31E-5	7.44E-7	1.54E-7	1.08E-5	6.00E-7	1.23E-7	1.60E-5	9.07E-7	1.92E-7	1.26E-5	7.25E-7	1.49E-7

14. **"Discuss how the risk reduction benefits and costs associated with implementing each SAMA were estimated. Please include the following:"**

a. "An indication of the source (reference) for each SAMA."

The following table 14a provides the source reference for each SAMA candidate.

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
1	Cap downstream piping of normally closed component cooling water drain and vent valves.	SAMA would reduce the frequency of a loss of component cooling event, a large portion of which was derived from catastrophic failure of one of the many single isolation valves.	N/A	1	—
2	Enhance loss of component cooling procedure to facilitate stopping reactor coolant pumps.	SAMA would reduce the potential for reactor coolant pump (RCP) seal damage due to pump bearing failure.	B	2	—
3	Enhance loss of component cooling procedure to present desirability of cooling down reactor coolant system (RCS) prior to seal LOCA.	SAMA would reduce the potential for RCP seal failure.	B	2	—
4	Provide additional training on the loss of component cooling.	SAMA would potentially improve the success rate of operator actions after a loss of component cooling (to restore RCP seal damage).	B	2	—
5	Provide hardware connections to allow another essential raw cooling water system to cool charging pump seals.	SAMA would reduce effect of loss of component cooling by providing a means to maintain the centrifugal charging pump seal injection after a loss of component cooling.	N/A	1 2	—
5A	Procedure changes to allow cross connection of motor cooling for RHRSW pumps.	SAMA would allow continued operation of both RHRSW pumps on a failure of one train of PSW.	C, 1.4.1 of IPE	12	—
6	Proceduralize shedding component cooling water loads to extend component cooling heatup on loss of essential raw cooling water.	SAMA would increase time before the loss of component cooling (and reactor coolant pump seal failure) in the loss of essential raw cooling water sequences.	B	2	—

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
7	Increase charging pump lube oil capacity.	SAMA would lengthen the time before centrifugal charging pump failure due to lube oil.	N/A	2	—
8	Eliminate the RCP thermal barrier dependence on component cooling such that loss of component cooling does not result directly in core damage.	SAMA would prevent the loss of recirculation pump seal integrity after a loss of component cooling. Watts Bar Nuclear Plant IPE said that they could do this with essential raw cooling water connection to charging pump seals.	N/A	2	—
9	Add redundant DC control power for PSW pumps C & D.	SAMA would increase reliability of PSW and decrease core damage frequency due to a loss of SW.	None	3	2-7
10	Create an independent RCP seal injection system, with a dedicated diesel.	SAMA would add redundancy to RCP seal cooling alternatives, reducing CDF from loss of component cooling or service water or from a station blackout event.	B	1	—
11	Use existing hydro-test pump for RCP seal injection.	SAMA would provide an independent seal injection source, without the cost of a new system.	B	4	—
12	Replace ECCS pump motor with air-cooled motors.	SAMA would eliminate ECCS dependency on component cooling system.	N/A	1	—
13	Install improved RCS pumps seals.	SAMA would reduce probability of RCP seal LOCA by installing RCP seal O-ring constructed of improved materials	B	1	—
14	Install additional component cooling water pump.	SAMA would reduce probability of loss of component cooling leading to RCP seal LOCA.	B	1	—
15	Prevent centrifugal charging pump flow diversion from the relief valves.	SAMA modification would reduce the frequency of the loss of RCP seal cooling if relief valve opening causes a flow diversion large enough to prevent RCP seal injection.	B	1	—

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
16	Change procedures to isolate RCP seal letdown flow on loss of component cooling, and guidance on loss of injection during seal LOCA.	SAMA would reduce CDF from loss of seal cooling.	B	1	—
17	Implement procedures to stagger high-pressure safety injection (HPSI) pump use after a loss of service water.	SAMA would allow HPSI to be extended after a loss of service water.	N/A	1	—
18	Use fire protection system pumps as a backup seal injection and high-pressure makeup.	SAMA would reduce the frequency of the RCP seal LOCA and the SBO CDF.	B	1	—
19	Enhance procedural guidance for use of cross-tied component cooling or service water pumps.	SAMA would reduce the frequency of the loss of component cooling water and service water.	C	1	2-10
20	Procedure enhancements and operator training in support system failure sequences, with emphasis on anticipating problems and coping.	SAMA would potentially improve the success rate of operator actions subsequent to support system failures.	D (various SAMAs for specific systems)	1 2	—
21	Improved ability to cool the residual heat removal heat exchangers.	SAMA would reduce the probability of a loss of decay heat removal by implementing procedure and hardware modifications to allow manual alignment of the fire protection system or by installing a component cooling water cross-tie.	D, 29 and 30	1	—
22	Provide reliable power to control building fans.	SAMA would increase availability of control room ventilation on a loss of power.	None	1	2-15
23	Provide a redundant train of ventilation.	SAMA would increase the availability of components dependent on room cooling.	D, 22 and 25	2	—

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
24	Procedures for actions on loss of HVAC.	SAMA would provide for improved credit to be taken for loss of HVAC sequences (improved affected electrical equipment reliability upon a loss of control building HVAC).	C	1 12	—
25	Add a diesel building switchgear room high temperature alarm.	SAMA would improve diagnosis of a loss of switchgear room HVAC.		1	
		Option 1: Install high temp alarm	None		2-5A
		Option 2: Redundant louver and thermostat	None		2-5B
26	Create ability to switch fan power supply to DC in an SBO event.	SAMA would allow continued operation in an SBO event. This SAMA was created for reactor core isolation cooling system room at Fitzpatrick Nuclear Power Plant.	N/A	1	—
27	Delay containment spray actuation after large LOCA.	SAMA would lengthen time of RWST availability.	N/A	2	—
28	Install containment spray pump header automatic throttle valves.	SAMA would extend the time over which water remains in the RWT, when full CS flow is not needed	N/A	4 8	—
29	Install an independent method of suppression pool cooling.	SAMA would decrease the probability of loss of containment heat removal.	E	5 6	—
30	Develop an enhanced drywell spray system.	SAMA would provide a redundant source of water to the containment to control containment pressure, when used in conjunction with containment heat removal.	E	5 6	—
31	Provide dedicated existing drywell spray system.	SAMA would provide a source of water to the containment to control containment pressure, when used in conjunction with containment heat removal. This would use an existing spray loop instead of developing a new spray system.	E	5 6	—

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
32	Install an unfiltered hardened containment vent.	SAMA would provide an alternate decay heat removal method for non-ATWS events, with the released fission products not being scrubbed.	C	5 6	—
33	Install a filtered containment vent to remove decay heat.	SAMA would provide an alternate decay heat removal method for non-ATWS events, with the released fission products being scrubbed.		5 6	
		Option 1: Gravel Bed Filter	E		—
		Option 2: Multiple Venturi Scrubber	E		—
34	Install a containment vent large enough to remove ATWS decay heat.	Assuming that injection is available, this SAMA would provide alternate decay heat removal in an ATWS event.	E	5 6	—
35	Create/enhance hydrogen recombiners with independent power supply.	SAMA would reduce hydrogen detonation at lower cost, Use either a new, independent power supply, a nonsafety-grade portable generator, existing station batteries, or existing AC/DC independent power supplies.	E	5 11	—
35A	Install hydrogen recombiners.	SAMA would provide a means to reduce the chance of hydrogen detonation.	E (Unit 1) C (Unit 2)	11	—
36	Create a passive design hydrogen ignition system.	SAMA would reduce hydrogen denotation system without requiring electric power.	E	4	—
37	Create a large concrete crucible with heat removal potential under the basemat to contain molten core debris.	SAMA would ensure that molten core debris escaping form the vessel would be contained within the crucible. The water cooling mechanism would cool the molten core, preventing a melt-through of the basemat.	E	5 6	—
38	Create a water-cooled rubble bed on the pedestal.	SAMA would contain molten core debris dropping on to the pedestal and would allow the debris to be cooled.	E	5 6	—

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
39	Provide modification for flooding the drywell head.	SAMA would help mitigate accidents that result in the leakage through the drywell head seal.	E	5 6	—
40	Enhance fire protection system and/or standby gas treatment system hardware and procedures.	SAMA would improve fission product scrubbing in severe accidents.	C	6	2-4
41	Create a reactor cavity flooding system.	SAMA would enhance debris coolability, reduce core concrete interaction, and provide fission product scrubbing.	E	1 3 7 8	2-16
42	Create other options for reactor cavity flooding.	SAMA would enhance debris coolability, reduce core concrete interaction, and provide fission product scrubbing.	D - See 41	1	—
43	Enhance air return fans (ice condenser plants).	SAMA would provide an independent power supply for the air return fans, reducing containment failure in SBO sequences.	N/A	1	—
44	Create a core melt source reduction system.	SAMA would provide cooling and containment of molten core debris. Refractory material would be placed underneath the reactor vessel such that a molten core falling on the material would melt and combine with the material. Subsequent spreading and heat removal form the vitrified compound would be facilitated, and concrete attack would not occur.	E	9	—
45	Provide a containment inerting capability.	SAMA would prevent combustion of hydrogen and carbon monoxide gases.	C	7 8	—
46	Use the fire protection system as a backup source for the containment spray system.	SAMA would provide redundant containment spray function without the cost of installing a new system.	None	4	2-2

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
47	Install a secondary containment filter vent.	SAMA would filter fission products released from primary containment.	C (standby gas treatment system (SGTS))	10	—
48	Install a passive containment spray system.	SAMA would provide redundant containment spray method without high cost.	E	10	—
49	Strengthen primary/secondary containment.	SAMA would reduce the probability of containment overpressurization to failure.	E	10 11	—
50	Increase the depth of the concrete basemat or use an alternative concrete material to ensure melt-through does not occur.	SAMA would prevent basemat melt-through.	E	11	—
51	Provide a reactor vessel exterior cooling system.	SAMA would provide the potential to cool a molten core before it causes vessel failure, if the lower head could be submerged in water.	D—See 41	11	—
52	Construct a building to be connected to primary/secondary containment that is maintained at a vacuum.	SAMA would provide a method to depressurize containment and reduce fission product release.	N/A	11	—
53	Not used.		None		—
54	Proceduralize alignment of spare diesel to shutdown board after loss of offsite power and failure of the diesel normally supplying it.	SAMA would reduce the SBO frequency.	C (with current swing diesel generator)	2	—
55	Not used.		None		—

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
56	Provide an additional diesel generator.	SAMA would increase the reliability and availability of onsite emergency AC power sources.	E	1 3 7 11	—
57	Provide additional DC battery capacity.	SAMA would ensure longer batter capability during an SBO, reducing the frequency of long-term SBO sequences.	E	1 3 7 11 12	—
58	Use fuel cells instead of lead-acid batteries.	SAMA would extend DC power availability in an SBO.	E	11	—
59	Procedure to cross-tie high-pressure core spray diesel.	SAMA would improve core injection availability by providing a more reliable power supply for the high-pressure core spray pumps.	N/A	1	—
60	Improve 4.16-kV bus cross-tie ability.	SAMA would improve AC power reliability.	None	1	2-11
61	Incorporate an alternate battery charging capability.	SAMA would improve DC power reliability by either cross-tying the AC busses, or installing a portable diesel-driven battery charger.	E	1 8 9	—
62	Increase/improve DC bus load shedding.	SAMA would extend battery life in an SBO event.	E	1 8	—
63	Replace existing batteries with more reliable ones.	SAMA would improve DC power reliability and thus increase available SBO recovery time.	N/A	11	—
63A	Mod for DC Bus A reliability.	SAMA would increase the reliability of AC power and injection capability. Loss of DC Bus A causes a loss of main condenser, prevents transfer from the main transformer to offsite power, and defeats one half of the low vessel pressure permissive for LPCI/CS injection valves.	C	1	2-13

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
64	Create AC power cross-tie capability with other unit.	SAMA would improve AC power reliability.	E	1 8 9	—
65	Create a cross-tie for diesel fuel oil.	SAMA would increase diesel fuel oil supply and thus diesel generator, reliability.	C	1	—
66	Develop procedures to repair or replace failed 4-kV breakers.	SAMA would offer a recovery path from a failure of the breakers that perform transfer of 4.16-kV nonemergency busses from unit station service transformers, leading to loss of emergency AC power.	C	1	2-9
67	Emphasize steps in recovery of offsite power after an SBO.	SAMA would reduce human error probability during offsite power recovery.	C	1	—
68	Develop a severe weather conditions procedure.	For plants that do not already have one, this SAMA would reduce the CDF for external weather-related events.	C	1 13	—
69	Develop procedures for replenishing diesel fuel oil.	SAMA would allow for long-term diesel operation.	C	1	—
70	Install gas turbine generator.	SAMA would improve onsite AC power reliability by providing a redundant and diverse emergency power system.	E	1	—
71	Not used.		None		—
72	Create a backup source for diesel cooling. (Not from existing system)	This SAMA would provide a redundant and diverse source of cooling for the diesel generators, which would contribute to enhanced diesel reliability.	D, 73	1	—
73	Use fire protection system as a backup source for diesel cooling.	This SAMA would provide a redundant and diverse source of cooling for the diesel generators, which would contribute to enhanced diesel reliability.	None	1	2-8
74	Provide a connection to an alternate source of offsite power.	SAMA would reduce the probability of a loss of offsite power event.	E	1	—

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
75	Bury offsite power lines.	SAMA could improve offsite power reliability, particularly during severe weather.	E	1	—
76	Replace anchor bolts on diesel generator oil cooler.	Millstone Nuclear Power Station found a high seismic SBO risk due to failure of the diesel oil cooler anchor bolts. For plants with a similar problem, this would reduce seismic risk. Note that these were Fairbanks Morse DGs.	D, See 114	1	—
77	Change undervoltage (UV), auxiliary feedwater actuation signal (AFAS) block and high pressurizer pressure actuation signals to 3-out-of-4, instead of 2-out-of-4 logic.	SAMA would reduce risk of 2/4 inverter failure.	N/A	1	—
78	Provide DC power to the 120/240-V vital AC system from the Class 1E station service battery system instead of its own battery.	SAMA would increase the reliability of the 120-VAC Bus.	None	12	2-12
79	Install a redundant spray system to depressurize the primary system during a steam generator tube rupture (SGTR).	SAMA would enhance depressurization during a SGTR.	N/A	1	—
80	Improve SGTR coping abilities.	SAMA would improve instrumentation to detect SGTR, or additional system to scrub fission product releases.	N/A	1 4 11	—
81	Add other SGTR coping abilities.	SAMA would decrease the consequences of an SGTR.	N/A	4 10 11	—
82	Increase secondary side pressure capacity such that an SGTR would not cause the relief valves to lift.	SAMA would eliminate direct release pathway for SGTR sequences.	N/A	10 11	—

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
83	Replace steam generators (SG) with a new design.	SAMA would lower the frequency of an SGTR.	N/A	1	—
84	Revise emergency operating procedures to direct that a faulted SG be isolated.	SAMA would reduce the consequences of an SGTR.	N/A	1	—
85	Direct SG flooding after a SGTR, prior to core damage.	SAMA would provide for improved scrubbing of SGTR releases.	N/A	10	—
86	Implement a maintenance practice that inspects 100% of the tubes in a SG.	SAMA would reduce the potential for an SGTR.	N/A	11	—
87	Locate residual heat removal (RHR) inside of containment.	SAMA would prevent intersystem LOCA (ISLOCA) out the RHR pathway.	A	10	—
88	Not used.		None		—
89	Install additional instrumentation for ISLOCAs.	SAMA would decrease ISLOCA frequency by installing pressure of leak monitoring instruments in between the first two pressure isolation valves on low-pressure inject lines, RHR suction lines, and HPSI lines.	A	3 4 7 8	—
90	Increase frequency for valve leak testing.	SAMA could reduce ISLOCA frequency.	A	1	—
91	Improve operator training on ISLOCA coping.	SAMA would decrease ISLOCA effects.	A	1	—
92	Install relief valves in the CC System.	SAMA would relieve pressure buildup from an RCP thermal barrier tube rupture, preventing an ISLOCA.	A	1	—

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
93	Provide leak testing of valves in ISLOCA paths.	SAMA would help reduce ISLOCA frequency. At Kewaunee Nuclear Power Plant, four MOVs isolating RHR from the RCS were not leak tested.	A	1	—
94	Revise EOPs to improve ISLOCA identification.	SAMA would ensure LOCA outside containment could be identified as such. Salem Nuclear Power Plant had a scenario where an RHR ISLOCA could direct initial leakage back to the pressurizer relief tank, giving indication that the LOCA was inside containment.	A	1	—
95	Ensure all ISLOCA releases are scrubbed.	SAMA would scrub all ISLOCA releases. One example is to plug drains in the break area so that the break point would cover with water.	A	1	—
96	Add redundant and diverse limit switches to each containment isolation valve.	SAMA could reduce the frequency of containment isolation failure and ISLOCAs through enhanced isolation valve position indication.	A	1	—
97	Modify swing direction of doors separating turbine building basement from areas containing safeguards equipment.	SAMA would prevent flood propagation, for a plant where internal flooding from turbine building to safeguards areas is a concern.	D, See 99	1	—
98	Improve inspection of rubber expansion joints on main condenser.	SAMA would reduce the frequency of internal flooding, for a plant where internal flooding due to a failure of circulating water system expansion joints is a concern.	D, See 99	1	—
99	Implement internal flood prevention and mitigation enhancements.	This SAMA would reduce the consequences of internal flooding.	None	1	2-14

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
100	Implement internal flooding improvements such as those implemented at Fort Calhoun.	This SAMA would reduce flooding risk by preventing or mitigating: • a rupture in the RCP seal cooler of the component cooling system • an ISLOCA in a shutdown cooling line, • an auxiliary feedwater (AFW) flood involving the need to remove a watertight door.	D—See 99	1	—
101	Install a digital feedwater upgrade.	This SAMA would reduce the chance of a loss of main feedwater following a plant trip.	C	1	—
102	Perform surveillances on manual valves used for backup AFW pump suction.	This SAMA would improve success probability for providing alternative water supply to the AFW pumps.	N/A	1	—
103	Install manual isolation valves around AFW turbine-driven steam admission valves.	This SAMA would reduce the dual turbine-driven AFW pump maintenance unavailability.	N/A	1	—
104	Install accumulators for turbine-driven AFW pump flow control valves (CVs).	This SAMA would provide control air accumulators for the turbine-driven AFW flow CVs, the motor-driven AFW pressure CVs and SG power-operated relief valves (PORVs). This would eliminate the need for local manual action to align nitrogen bottles for control air during a LOOP.	N/A	4 8	—
105	Proceduralize intermittent operation of HPCI.	SAMA would allow for extended duration of HPCI availability.	None	1	2-3
106	Increase the reliability of safety relief valves. (Adding signals to add electrical signal to open automatically).	SAMA reduces the probability of a certain type of medium break LOCA. Hatch evaluates medium LOCA initiated by an MSIV closure transient with a failure of SRVs to open. Reducing the likelihood of the failure for SRVs to open, subsequently reduces the occurrence of this medium LOCA.	C	12	—

Table 14a. Disposition of initial SAMAs investigated.

Phase I SAMA ID number	SAMA title	Result of potential enhancement	Screening criterion*	Reference Number	Phase II SAMA ID number**
107	Install motor-driven feedwater pump.	SAMA would increase the availability of injection subsequent to MSIV closure.	E	1 12	—
108	Enhance procedure to instruct operators to trip unneeded RHR/CS pumps on loss of room ventilation.	SAMA increases availability of required RHR/CS pumps. Reduction in room heat load allows continued operation of required RHR/CS pumps, when room cooling is lost.	C, IPE 1.4.1	12	—
109	Increase available net positive suction head (NSPH) for injection pumps.	SAMA increases the probability that these pumps will be available to inject coolant into the vessel by increasing the available NPSH for the injection pumps.	C	1	—
110	Increase the safety relief valve (SRV) reseat reliability.	SAMA addresses the risk associated with dilution of boron caused by the failure of the SRVs to reseat after standby liquid control (SLC) injection.	E	1	—
111	Reduce DC dependency between high-pressure injection system and ADS.	SAMA would ensure containment depressurization and high-pressure injection upon a DC failure.	N/A	1	—
112	Modify (RWCU) for use as a decay heat removal system and proceduralize use.	SAMA would provide an additional source of decay heat removal.	C	1	2-6
113	Use control rod drive (CRD) for alternate boron injection.	SAMA provides an additional system to address ATWS with SLC failure or unavailability.	C	1	2-1
114	Increase seismic ruggedness of plant components.	SAMA would increase the availability of necessary plant equipment during and after seismic events.	C	11 13	—
115	Allow cross connection of uninterruptable compressed air supply to opposite unit.	SAMA would increase the ability to depressurize containment using the hardened vent.	C	12 13	—

* N/A indicates that the proposed SAMA is not applicable to the Hatch BWR-4/Mark I design.
A indicates that the proposed SAMA is related to mitigation of an ISLOCA. Per IN-92-36, and its supplement, ISLOCA contributes little risk for boiling water reactors, because of the lower primary pressures. Because of the low risk contribution due to ISLOCA, this SAMA has not been developed further.

B indicates that the proposed SAMA is related to RCP seal leakage. A review of NUREG-1560 indicates that although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs.
C indicates that the proposed SAMA has already been installed at Hatch.
D indicates that similar item is addressed under other proposed SAMAs.
E indicates that SAMA did not pass initial screening to move into Phase II—no Phase II number assigned.
** ID numbers in parenthesis show SAMAs initially considered but dropped from Phase II analysis (e.g. already implemented at Plant Hatch or did not pass the screening criteria).

References for Table 14a

1. NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," Volume 2, NRC, December 1997.
2. Letter from Mr. M. O. Medford (Tennessee Valley Authority) to NRC Document Control Desk, dated September 1, 1992, "Watts Bar Nuclear Plant Units 1 and 2 – Generic Letter (GL) – Individual Plant Examination (IPE) for Severe Accident Vulnerabilities – Response"
3. NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," Volume 1, Table 5.36 Listing of SAMDAs considered for the Comanche Peak Steam Electric Station, NRC, May 1996.
4. Letter from Mr. D. E. Nunn (Tennessee Valley Authority) to NRC Document Control Desk, dated October 7, 1994, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 – Severe Accident Mitigation Design Alternatives (SAMDA) – Response to Request for Additional Information (RAI)"
5. "Cost Estimate for Severe Accident Mitigation Design Alternatives, Limerick Generating Station for Philadelphia Electric Company," Bechtel Power Corporation, June 22, 1989.
6. NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," Volume 1, Table 5.35, Listing of SAMDAs considered for the Limerick, NRC, May 1996.
7. Letter from Mr. W. J. Museler (Tennessee Valley Authority) to NRC Document Control Desk, dated October 7, 1994, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 – Severe Accident Mitigation Design Alternatives (SAMDA)."
8. NUREG-0498, "Final Environmental Statement related to the operation of Watts Bar Nuclear Plant, Units 1 and 2," Supplement No. 1, NRC, April 1995.
9. Letter from Mr. D. E. Nunn (Tennessee Valley Authority) to NRC Document Control Desk, dated June 30, 1994. "Watts Bar Nuclear Plant (WBN) Unit 1 and 2 – Severe Accident Mitigation Design Alternatives (SAMDAs) Evaluation from Updated Individual Plant Evaluation (IPE)."
10. Letter form N. J. Liparulo (Westinghouse Electric Corporation) to NRC Document Control Desk, dated December 15, 1992, "Submittal of Material Pertinent to the AP600 Design Certification Review."
11. NUREG-1462, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design," NRC, August 1994.
12. Hatch IPE
13. Hatch IPEEE

b. "The bases for the preliminary cost estimates for each of the SAMA candidates for which a cost estimate was made, and the bases for the final cost estimates for the nine SAMAs in table 7 of the SAMA submittal."

The preliminary cost estimate for each Phase I SAMA is based on the total cost involved in performing engineering, utility cost, procurement and construction costs. Credit was taken for the past experiences on total cost with similar modifications either estimated or actually performed for other plants for the purposes of the preliminary estimate for Phase I.

The final cost estimates for Phase II SAMA items were developed based on the estimated cost of engineering, utility, procurement, and construction for the proposed modification, similar to Phase I costs. However, Phase II estimates were developed considering Hatch's specific plant design further in detail as compared to Phase I estimates.

The engineering cost includes preparation of the design change package, coordination with site, and site support required during implementing of the package. Utility cost includes preparation of the design implementation package and generation of the as-built notices after the design change is implemented. Procurement costs include the cost of materials needed to implement the design package. Every effort was made to include the supplier input for the material cost. Construction costs include the cost of performing the physical changes to the plant per the design package. Also, the cost of training to the plant personnel was included in the Phase II estimates.

c. "Estimates of the ΔCDF and Δperson-rem for each of the 43 unique Hatch SAMA candidates. Also provide the calculations showing how these values were obtained."

During the conference call between the NRC and Southern Company on June 8, 2000, it was determined that the scope of this question should be redefined to only include the SAMAs in table 7 of the ER (i.e., 10 SAMAs). These are the only SAMAs for which ΔCDF and Δperson-rem values were calculated

<u>ΔCDF</u>
See Attachment 6.

<u>Δperson-rem</u>
The conditional dose for each sequence was taken from the MACCS2 results for "Overall Results Combining 2 Emergency Response Cohorts" (i.e., the 95% of the population evacuated and the 5% not evacuated). The population dose reported was the mean effective dose equivalent (EDE) whole body dose, total life, 0-50 miles. The conditional dose for each sequence is:

Sequence	2	4	5	11	15
Dose-Conditional (Person-rem)	1.06E+06	1.02E+06	1.15E+06	7.02E+05	1.13E+06

These conditional doses were multiplied by the release frequencies given in item 1.b, resulting in the 0-50 mile population dose risk in person-rem. The results are:

Sequence	2	4	5	11	15	Sum Of Annual Dose Risk
P2-7	1.90E+00	7.58E-01	1.90E-01	5.21E-01	1.05E-03	3.37E+00
P2-2	1.90E+00	7.60E-01	1.90E-01	5.22E-01	1.05E-03	3.37E+00
P2-5	1.86E+00	7.59E-01	1.90E-01	5.22E-01	1.05E-03	3.34E+00
P2-8	1.87E+00	7.59E-01	1.90E-01	5.21E-01	1.05E-03	3.34E+00
P2-12	1.90E+00	7.60E-01	1.90E-01	5.22E-01	1.05E-03	3.37E+00
P2-14	1.90E+00	7.60E-01	1.90E-01	5.22E-01	1.05E-03	3.37E+00
P2-11	1.90E+00	7.60E-01	1.90E-01	5.22E-01	1.05E-03	3.37E+00

15. **"Uncertainties in the core damage frequency, public risk, risk reduction estimates and cost estimates all contribute to uncertainties in the value-impact analyses for each SAMA. Factors of three to five are common in the Level 1 PRA alone. Please justify why uncertainties were not considered in the value-impact analysis. Explain the influence that uncertainties could have on the results of the SAMA analysis, including SAMA screening and dispositioning, if the impact of uncertainties were explicitly accounted for in the analysis."**

The basic event values used in the CAFTA-based Plant Hatch PSA are, for the most part, based on statistical distributions. The mean value from these distributions is substituted into the PSA model as the actual basic event value. This in itself addresses uncertainty on an event level. CAFTA, however, does not conveniently lend itself to propagating this uncertainty through the model for an overall result. It is for this reason that the new Hatch PSA does not have an uncertainty analysis.

Even if all uncertainties associated with the SAMA with having the highest total benefit (SAMA P2-5, with a total benefit of $2,492) had the impact of increasing the benefit by a factor of five (for a total benefit of $12,460), this SAMA would still be much lower than it's cost of implementation ($100,000 per unit). Thus, the net benefit of the SAMA would be negative, and the SAMA would not be justified on a cost-benefit basis.

16. **"For SAMA 2-8, "Use Fire Protection as a Backup to Diesel Generator Cooling," the description indicates that Diesel Generator 1B already has an alternate cooling water supply. This would seem to imply that the scope and cost of implementing this SAMA would differ for Unit 1 and Unit 2. However, only one implementation**

cost and one risk reduction benefit are listed. Please identify the diesel generators on which this SAMA would be implemented, and confirm whether the cost and risk reduction estimates are for Unit 1, Unit 2, or both units."

There are a total of five emergency diesel generators (EDGs) 1A, 1B, 1C, 2A, and 2C at Plant Hatch for Units 1 & 2. The EDG 1B is a swing diesel generator and is shared between Units 1 & 2. The cooling water to EDGs is supplied from the safety-related portion of the PSW. Swing EDG 1B has its own dedicated standby service water pump and backed by Division I and Division II of the Unit 1 PSW system. Loss of any of the division of the PSW system will result in loss of the corresponding EDG, and redundancy is provided by the swing EDG. An alternate source of cooling water was proposed for EDGs 1A, 1B, 1C, 2A, and 2C from a nearby fire hydrant. (Although the swing EDG is provided with its own dedicated standby service water pump and backed by Division I and Division II of the Unit 1 PSW system, the modifications are also recommended for EDGs 1B.) The cost of modification for each EDG is the same. Since both units share EDG 1B, the cost of the modification associated with EDG 1B is split between the two units.

17. **"Section 4 indicates that an initial list of 115 SAMAs was reduced to 43 unique, applicable SAMAs, and subsequently reduced to 16 SAMAs for further analysis. However, it appears that only 114 SAMAs are accounted for in table 6 and when the screening is performed there would be 42 unique SAMAs and 15 candidates for further analysis. Please address this inconsistency. Also, clarify why SAMA 41 is designated as an "E", but is still assigned Phase II number "2-16"."**

Table 6 only accounts for 114 SAMAs. The following discussion explains the reduction of candidates and the screening process.

<u>SAMA Candidates and Screening Process:</u>

An initial list of 114 SAMA candidates was developed from insights from the Plant Hatch IPE and IPEEE results, lists of severe accident mitigation design alternatives at other nuclear power plants, NRC documents, and documents related to advanced power reactor designs. This initial list was then screened to remove those that were not applicable to Plant Hatch due to design differences.

Twenty-six of the initial 114 candidate SAMAs were removed from further consideration as they did not apply to the BWR-4/Mark I design used at Plant Hatch. These 26 SAMAs were 1, 5, 7, 8, 12, 17, 26, 27, 28, 43, 52, 59, 63, 77, 79, 80, 81, 82, 83, 84, 85, 86, 102, 103, 104, and 111.

An additional nine candidates were removed from consideration because they were related to mitigation of an ISLOCA. According to NRC Information Notice 92-36 and its supplement, ISLOCA contributes little risk for boiling water reactors because of the

lower primary pressures. A review of the Hatch 1 PSA model agreed with this conclusion. These nine SAMAs were 87, 89, 90, 91, 92, 93, 94, 95, and 96.

Eleven SAMA candidates were related to Reactor Coolant Pump seal leakage. NUREG-1560 indicates that although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to core damage frequency in BWRs. A review of the Hatch 1 PSA model agreed with this conclusion. Therefore, the following 11 candidates were removed from further consideration: 2, 3, 4, 6, 10, 11, 13, 14, 15, 16, and 18.

The following 16 SAMA candidates were found to be in place at HNP and were thus dropped from further consideration: 5A, 24, 32, 45, 47, 54, 65, 67, 68, 69, 101, 106, 108, 109, 114, 115.

The following 10 SAMA candidates were of sufficient similarity to other SAMA candidates that they were either combined or dropped from further consideration: 20, 21, 23, 42, 51, 72, 76, 97, 98, 100.

This left 42 unique SAMA candidates that were applicable to HNP and were of potential value in averting the risk of severe accidents. A preliminary cost estimate was prepared for each of these candidates to focus on those that had the possibility of having a positive benefit and to eliminate those whose costs were clearly beyond the possibility of any corresponding benefit.

When the screening cutoff of $500,000 was applied, 26 candidates were eliminated that were more expensive than any possible offsetting benefit. These 26 SAMAs were 29, 30, 31, 33, 34, 35, 35A, 36, 37, 38, 39, 44, 48, 49, 50, 56, 57, 58, 61, 62, 64, 70, 74, 75, 107, and 110.

This left the following 16 candidates for further analysis: 9, 19, 22, 25, 40, 41, 46, 60, 63A, 66, 73, 78, 99, 105, 112, 113.

Level II SAMA Analysis

For each of the 16 remaining SAMA candidates, a more detailed conceptual design was prepared along with a more detailed estimated cost. This information was then used to evaluate the effect of the candidate changes upon the plant safety model.

During the Level II analysis, it was determined that five of the SAMA candidates (SAMA numbers 40, 63A, 66, 112, 113) were adequately covered by existing plant design and procedures. In addition, the Phase II costing for one of the candidates (SAMA number 41) was found to be in excess of the $500,000 screening criterion (refer to response to Question 14.b for details on cost development). As a result, these six SAMA candidates (SAMA numbers 40, 41, 63A, 66, 112, 113) were dropped from further consideration.

The 10 SAMAs that remained for detailed analysis are as follows:

SAMA Number	Phase II Number
9	2-7
19	2-10
22	2-15
25	2-5A
46	2-2
60	2-11
73	2-8
78	2-12
99	2-14
105	2-3

18. **"In general, the candidate SAMAs focus on hardware changes that tend to be expensive to implement. While hardware changes may often provide the greatest risk reduction, consideration should be given to other options that provide marginally smaller risk reductions but with much smaller implementation costs. For example, instead of adding another service water (SW) pump to improve SW reliability, risk could be reduced by determining the causes for failures in the existing SW pumps and adjusting the preventive maintenance program or procedures to address the dominant failure modes. Please justify why these type of options were not considered as alternative SAMAs to address the major risk contributors at Hatch."**

The following SAMA candidates involving procedure changes or training enhancements were evaluated for applicability during Phase I. Some of the SAMA titles are based on applicability to a pressurized water reactor (PWR) plant. However, an effort was made to relate each SAMA item to Plant Hatch systems:

SAMA ID Number	Title
2	Enhance loss of component cooling procedure to facilitate stopping RCPs. The purpose of the procedure enhancement is to avoid RCP seal damage on loss of component cooling system. The SAMA was related to reactor building closed cooling water (RBCCW) system and recirculation pumps for Plant Hatch.
3	Enhance loss of component cooling procedure to present desirability of cooling down RCS prior to seal LOCA. The purpose of the procedure enhancement is to provide clear guidance to cool down RCS prior to seal LOCA minimizing the potential for seal

SAMA ID Number	Title
	damage. The SAMA was related to RBCCW system and recirculation pumps for Plant Hatch.
4	Additional training on the loss of component cooling. The purpose of the SAMA item is to ensure availability of the component cooling water system on loss-of-offsite power by aligning emergency source of power to improve the success rate of operator actions to avoid RCP seal damage. The SAMA was related to RBCCW system and recirculation pumps for Plant Hatch.
5A	Procedures changes to allow cross connection of motor cooling for reactor heat removal service water (RHRSW) pump motors, when one division of PSW is failed (Hatch IPE).
6	On loss of essential raw cooling water, proceduralize shedding component cooling water loads to extend component cooling heatup. The SAMA was related to RBCCW system and recirculation pumps for Plant Hatch.
16	Change procedures to isolate RCP seal letdown flow on loss of component cooling and guidance on loss of injection during seal LOCA. The SAMA was related to RBCCW system and recirculation pumps for Plant Hatch.
17	Implement procedures to stagger HPSI pump use after a loss of service water. This item was not found applicable to BWR-4/Mark I design.
18	Use of the fire protection system pumps as a backup seal injection and high-pressure makeup source.
19	Procedural guidance for use of cross-tied component cooling or service water pumps. This SAMA item was related to PSW and RBCCW systems for Plant Hatch.
20	Procedure enhancements and operator training in support system failure sequences, with emphasis on anticipating problems and coping. This SAMA item is generic in nature and was considered to be addressed as a part of the overall task.
24	Procedures for actions on loss of HVAC (Hatch IPE).
40	Enhance fire protection system and/or SGTS hardware and procedures.

SAMA ID Number	Title
	offsite power and failure of the diesel normally supplying it. Hatch design already provides for a swing diesel.
59	Procedure to crosstie high-pressure core spray diesel. This SAMA item is not applicable to Plant Hatch.
60	Improve 4.16-kV bus crosstie ability. The purpose of the SAMA item is to improve AC power reliability.
62	Increase/improve DC bus load shedding. This SAMA item would extend the life of the battery life in an SBO event. This item also included hardware changes.
66	Develop procedures to repair or replace failed 4 kV breakers.
67	Emphasize steps in recovery of offsite power after an SBO.
68	Develop a severe weather conditions procedure.
69	Develop procedures for replenishing diesel fuel oil.
84	Revise emergency operating procedures to direct that a faulted steam generator be isolated. This SAMA item is not applicable to BWRs.
91	Improve operator training on ISLOCA coping.
94	Revise EOPs to improve ISLOCA identification.
98	Improve inspection of rubber expansion joints on main condenser.
105	Proceduralize intermittent operation of HPCI.
108	Procedure to instruct operators to trip unneeded RHR/CS pumps on loss-of-room ventilation (Hatch IPE).

Review indicates that SAMA items 5A, 24, 40, 54, 66, 67, 68, 69, and 108 have already been incorporated into Plant Hatch design. Items 17, 59, and 84 are not applicable to the Hatch BWR-4/Mark I design. Items 2, 3, 4, 6, 16, and 18 are related to RCP seal leakage. Based on the review of NUREG-1560, although RCP seal leakage is important for PWRs, recirculation pump leakage for BWRs does not significantly contribute to CDF and were not evaluated further. Items 20 and 98 have been addressed by other SAMA items, considered as a part of the review. Items 91 and 94 are related to mitigation of an

ISLOCA. Due to the low risk contribution of ISLOCA, these items were not developed further.

Items 19, 60, 62, and 105 were considered for detailed review. Item 62 did not meet the screening criteria for consideration under Phase II. Items 19, 60, and 105 were evaluated under Phase II. It was discovered that intent of SAMA Item 19 is already met by the existing Hatch design and the risk reduction benefit for the other SAMA items did not justify cost of implementation of these items.

The regulations governing license renewal acknowledge that the scope of the Maintenance Rule and the License Renewal Rule are the same. One of the objectives of the Maintenance Rule is to identify the root causes of maintenance-preventable failures of risk-significant components and adjust maintenance practices accordingly to reduce or eliminate those failures. Thus, while SAMA candidates which do not involve hardware changes were considered as discussed above, developing a program to determine the causes for failures of risk-significant components or systems (such as SW pumps) and to adjust accordingly the preventive maintenance program or procedures is considered to be duplicative of the intent of the Maintenance Rule and was not proposed as a SAMA candidate.

References

1. PSA-h-98-005, Calculation for Evaluation of Extended Power Uprate on E. I. Hatch PRA.

2. REES-h-97-001, Changes in Timing of Core Melt Progression and Expected Releases due to Extended Power Uprate.

3. Plant Hatch IPE Submittal to the NRC, Volumes I and II, in response to Generic Letter 88-20.

4. Plant Hatch PSA Model, Unit 1, Revision 0.

5. Plant Hatch Revision 4 Emergency Operating Procedures.

6. Plant Hatch Revision 0 Severe Accident Guidelines (SAGs).

7. Regulatory Guide 1.174.

8. ASME Proposed Standard for Probabilistic Risk Assessment For Nuclear Power Plant Applications, Revision 12, Draft, May 30, 2000.

9. FAI/98-88, Level II Process for Plant Hatch, Fauske and Associates, March 16, 1999.

10. Blaha, James S., Assistant for Operations OEDO to NRC Commissioners, April 8, 1999.