5.0 Environmental Impacts of Postulated Accidents

Environmental issues associated with postulated accidents were discussed in the *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* (GEIS), NUREG-1437 (NRC 1996; 1999).[a] The GEIS included a determination of whether the analysis of the environmental issue could be applied to all plants and whether additional mitigation measures would be warranted. Issues were then assigned a Category 1 or a Category 2 designation. As set forth in the GEIS, Category 1 issues are those that meet all of the following criteria:

(1) The environmental impacts associated with the issue have been determined to apply either to all plants or, for some issues, to plants having a specific type of cooling system or other specified plant or site characteristics.

(2) A single significance level (i.e., SMALL, MODERATE, or LARGE) has been assigned to the impacts (except for collective offsite radiological impacts from the fuel cycle and from high-level waste and spent fuel disposal).

(3) Mitigation of adverse impacts associated with the issue has been considered in the analysis, and it has been determined that additional plant-specific mitigation measures are likely not to be sufficiently beneficial to warrant implementation.

For issues that meet the three Category 1 criteria, no additional plant-specific analysis is required unless new and significant information is identified.

Category 2 issues are those that do not meet one or more of the criteria of Category 1, and therefore, additional plant-specific review for these issues is required.

This chapter describes the environmental impacts from postulated accidents that might occur during the license renewal term.

5.1 Postulated Plant Accidents

A Category 1 issue in 10 CFR Part 51, Subpart A, Appendix B, Table B-1, related to postulated accidents that is applicable to Edwin I. Hatch Nuclear Plant (HNP) is listed in Table 5-1. The Southern Nuclear Operating Company (SNC) stated in its Environmental Report (ER; SNC 2000a) that it is not aware of any new and significant information associated with the renewal of

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

Table 5-1. Category 1 Issue Applicable to Postulated Accidents During the Renewal Term

ISSUE—10 CFR Part 51, Subpart A, Appendix B, Table B-1	GEIS Sections
POSTULATED ACCIDENTS	
Design-Basis Accidents (DBAs)	5.3.2; 5.5.1

the HNP operating licenses (OLs). No significant new information has been identified by the staff during its review. Therefore, the staff concludes that there are no impacts related to this issue beyond those discussed in the GEIS. For this issue, the GEIS concluded that the impacts are SMALL, and plant-specific mitigation measures are not likely to be sufficiently beneficial to be warranted.

A brief description of the staff's review and the GEIS conclusions, as codified in Table B-1, follows.

Design-Basis Accidents (DBAs). Based on information in the GEIS, the Commission found that "The NRC staff has concluded that the environmental impacts of design- basis accidents are of small significance for all plants." The staff has not identified any significant new information during its independent review of the SNC ER, the staff's site visit, the scoping process, its review of public comments, or its evaluation of other available information. Therefore, the staff concludes that there are no impacts of DBAs beyond those discussed in the GEIS.

A Category 2 issue related to postulated accidents that is applicable to HNP is listed in Table 5-2.

Table 5-2. Category 2 Issue Applicable to Postulated Accidents During the Renewal Term

ISSUE—10 CFR Part 51, Subpart A, Appendix B, Table B-1	GEIS Sections	10 CFR 51.53(c)(3)(ii) Subparagraph	SEIS Section
POSTULATED ACCIDENTS			
Severe Accidents	5.3.3; 5.3.3.2 5.3.3.3; 5.3.3.4; 5.3.3.5; 5.4.; 5.5.2	L	5.2

Severe Accidents. Based on information in the GEIS, the Commission found that "The probability weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to groundwater, and societal and economic impacts from severe accidents are small for all plants. However, alternatives to mitigate severe accidents must be considered for all plants that have not considered such alternatives."

The staff has not identified any significant new information with regard to the consequences from severe accidents during its independent review of the SNC ER, the staff's site visit, the scoping process, its review of public comments, or its evaluation of other available information. Therefore, the staff concludes that there are no impacts of severe accidents beyond those discussed in the GEIS. However, in accordance with 10 CFR 51.53(c)(3)(ii)(L), the staff has reviewed severe accident mitigation alternatives (SAMAs) for HNP. The results of its review are discussed in Section 5.2.

5.2 Severe Accident Mitigation Alternatives

Title 10 of the Code of Federal Regulations, Part 51.53(c)(3)(ii)(L) requires that license renewal applicants consider alternatives to mitigate severe accidents if the staff has not previously evaluated SAMAs for the applicant's plant in an environmental impact statement or related supplement or in an environmental assessment. The purpose of this consideration is to ensure that plant design changes with the potential for improving severe accident safety performance are identified and evaluated. SAMAs have not been previously considered for HNP; therefore, the following sections address those alternatives.

5.2.1 Introduction

SNC submitted an assessment of SAMAs for HNP as part of the ER (SNC 2000a). This assessment was based on the *Hatch 1 Probabilistic Safety Assessment* (PSA), Revision 0 (an updated version of the Individual Plant Examination [IPE, SNC 1992]) for core damage frequency (CDF) estimation and containment performance, and a separate Level 3 model for the ER SAMA risk determination. In identifying and evaluating potential SAMAs, SNC considered the insights from the HNP IPE and Individual Plant Examination for External Events (IPEEE, SNC 1996a) as well as several recent SAMA analyses for other plants (Limerick, Watts Bar, and Comanche Peak) and other industry documentation, such as NUREG-1560 (NRC 1997a), NUREG-1462 (NRC 1994a), and the GEIS (NRC 1996; 1999), that discuss potential plant improvements. SNC identified and evaluated 114 SAMA candidates. As discussed below, this list was reduced to 42 unique SAMA candidates because the remainder were either not applicable to boiling-water reactors (BWRs), related to phenomena that are not risk-significant in BWRs, or similar to other SAMAs being considered. Other SAMAs were excluded because they had already been implemented at HNP to address insights and recommendations from the HNP PSA and IPE. The study concluded that none of the remaining SAMAs was cost-beneficial.

Based on a review of the SAMA assessment, the NRC issued a request for additional information (RAI) to SNC by letter dated May 30, 2000 (NRC 2000a). Major issues concerned the process used by the license renewal applicant to identify potential SAMAs, the determination and documentation of the risk profile used in the analysis process, the determination of the risk benefits, and the bases for the SAMA implementation costs. SNC submitted additional information by letters dated July 26, 2000 (SNC 2000b), and August 31, 2000 (SNC 2000c), clarifying its approach for SAMA identification, risk quantification and documentation, and SAMA implementation and benefit quantification. This response addressed the staff's concerns and reaffirmed that none of the remaining SAMAs would be cost-beneficial.

An assessment of SAMAs for HNP is presented below.

5.2.2 Estimate of Risk for HNP

SNC's estimates of offsite risk at HNP are summarized in Section 5.2.2.1. The summary is followed by a review of SNC's risk estimates in Section 5.2.2.2.

5.2.2.1 SNC's Risk Estimates

The SAMA analysis is based on two distinct analyses: (1) the HNP PSA, Revision 0 (an update of the HNP Probabilistic Risk Assessment (PRA)/IPE model), and (2) a Level 3 analysis developed specifically for the ER SAMA analyses. The HNP PSA is a conversion of the IPE from the "large event tree, small fault tree" approach to the "linked fault tree" approach. The new model incorporated new information on equipment performance, plant configuration changes, and refinements in PRA modeling techniques. It contains a Level 1 analysis to determine the CDF and a Level 2 analysis to determine containment performance during severe accidents. The Level 1 analysis includes only internal events. Although SNC did not include the results of the IPEEE, it did review the IPEEE as part of Phase I of its SAMA evaluation. The total CDF for internal events is 1.6E-5 per reactor year (ry) and the Large Early Release Frequency (LERF) is 2.7E-6/ry. The breakdown of CDF is provided in Table 5-3. As shown in this table, the current analyses show that Loss of Feedwater events are a dominant contributor to CDF, followed by Loss of Station Battery A and Loss of Offsite Power.

Table 5-3. HNP Core Damage Frequency Profile

Accident Category	PSA % Total CDF
Loss of Offsite Power	16.7
Loss of 600V AC Bus C	8.4
Loss of Feedwater	20.2
Loss of Station Battery A	18.0
Main Steam Isolation Valve Closure	7.3
Anticipated Transient Without Scram (ATWS)	4.3

The Level 3 analysis uses the MELCOR Accident Consequence Code System 2 (MACCS2) code, Version 1.12, to determine the offsite risk impacts on the surrounding environment and public. Inputs for the Level 3 analysis include the HNP core radionuclide inventory, the Level 2 release fractions, site meteorological data, projected population distribution for the year 2030, emergency response evacuation modeling, and economic data.

SNC estimates the dose to the population within 80 km (50 mi) of the HNP site from internal initiators to be 3.5 person-rem per year. Table 5-4 shows the distribution of containment performance contributions to the population dose. The current submittal indicates that early containment failure releases dominate. The early release category includes Sequence 2, a station blackout event; Sequence 4, a loss of containment heat removal/drywell failure event; and Sequence 11, an ATWS with drywell failure event. As noted by SNC, risk is dominated by Sequence 2 because it is estimated to result in a higher dose (1.9 person-rem) and because it has a relatively high estimate for its probability of occurrence (1.79×10^{-6}/yr).

Table 5-4. Containment Failure Profile

Contributor	Submittal % Contribution to Population Dose
Bypass	5.4
Early	91.2
Late	3.3
Intact (venting)	<0.1

5.2.2.2 Review of SNC's Risk Estimates

SNC's estimate of offsite risk at HNP is based on the HNP PSA and a separate Level 3 MACCS2 analysis. This review considered the following major elements:

- the Level 1 and 2 risk models that form the bases for the December 1992 IPE submittal (SNC 1992)

- the major modifications to the IPE model that have been incorporated in the HNP PSA

- the Level 3 analyses performed to translate fission product release frequencies from the Level 2 PRA model into offsite consequence measures.

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

The staff's review of the HNP IPE is described in an NRC safety evaluation dated July 18, 1995 (NRC 1995). In that review, the staff evaluated the methodology, models, data, and assumptions used to estimate the CDF and characterize containment performance and fission product releases. The staff concluded that SNC's analysis met the intent of Generic Letter 88-20 (NRC 1988); that is, the IPE was of adequate quality to be used to look for design or operational vulnerabilities. Although the staff reviewed certain aspects of the IPE in more detail than others, the review primarily focused on the licensee's ability to examine HNP for severe accident vulnerabilities and not specifically on the detailed findings or quantification estimates. Overall, the staff believed that the HNP IPE was of adequate quality to be used as a tool in searching for areas with high potential for risk reduction and to assess such risk reductions, especially when the risk models are used in conjunction with insights, such as those from risk importance, sensitivity, and uncertainty analyses.

As mentioned earlier, the HNP PSA is an update and conversion of the original IPE submitted to the NRC. It was reviewed by the SNC PSA engineering staff. Because the model was developed from the original IPE, SNC determined that all reviews from the original IPE were still applicable.

A comparison of risk profiles between the original IPE (which was reviewed by the NRC staff) and the current version indicated several changes. First, the overall CDF has decreased. As discussed below, this result is due to several factors. In addition, the dominance of certain events (e.g., Loss of Feedwater, Loss of Station Battery, etc.) has increased, while the importance of other events (e.g., Loss of Offsite Power) has decreased. Nevertheless, the results confirm that the overall risk for the plant is low.

One major group of changes in the model from the IPE to the PSA is the addition of more details to the support system models, especially the electrical systems. However, perhaps the greater impact on the results is due to the conversion of the risk model from the large event tree method to the linked fault tree method. The original IPE fault trees were quantified using very small truncation values to capture as much of the failure probabilities as possible in the event tree split fractions. The event trees were then quantified at much higher truncation values to speed up the quantification process. In the PSA, a single truncation value was used throughout the quantification process. The differences in the quantification methods largely account for the differences in the estimates for the overall CDF and LERF.

The revised CDF estimated for HNP is still comparable to values estimated for other BWR/3 and BWR/4 model plants. Figure 11.2 of NUREG-1560 (NRC 1997a) shows that the total CDFs for these plants range from 9E-8/ry to 8E-5/ry, with an average value of 2E-5/ry.

SNC submitted an IPEEE by letter dated January 26, 1996 (SNC 1996a), in response to Supplement 4 of Generic Letter 88-20. SNC did not identify 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. In a letter dated October 23, 2000, the staff concluded that the submittal met the intent of Supplement 4 to Generic Letter 88-20 (NRC 2000b). SNC chose not to include the results of its analysis in the estimate of CDF. In its response to an RAI on how plant-specific external event insights were considered, SNC stated that, based on its review of the HNP IPEEE and NUREG-1560 (NRC 1997a) during Phase I of the SAMA evaluation, it identified three SAMAs associated with external events. Two had already been implemented at HNP and one did not pass the initial screening criteria. The largest CDF contributor examined in the IPEEE was internal fires, which contributed 7.5 E-06/ry for HNP Unit 1 and 5.4 E-06/ry for HNP Unit 2. A staff review of the risk-dominant fire zones revealed that the CDF from a fire in a single zone was typically an order of magnitude less than the CDF calculated for internal events. Therefore, there is reasonable assurance that the risk associated with a fire would be bounded by the CDF calculated for internal events. The staff also reviewed the Fire Submittal Screening Review of HNP (an attachment to NRC 2000b) and did not identify additional alternatives that needed to be further evaluated by the applicant. The staff finds SNC's consideration of external events for the purpose of this SAMA review acceptable.

The HNP IPE model included Level 2 components. Hence, the conversion to the linked fault tree method impacted the Level 2 results. Differences in the Level 2 results were also impacted by factors such as: (1) a power uprate, and (2) a new version of the Modular Accident Analysis Program (MAAP) code, which was used to estimate release fractions and provide containment analysis details.

The process used by SNC to extend the containment performance (Level 2) portion of the PSA to the offsite consequence (Level 3) assessment was reviewed. This included consideration of the source terms used to characterize fission product releases for each containment release mode and the major inputs and assumptions used in the offsite consequence analyses. SNC used Version 3.0B BWR, Revision 10, of the MAAP code to analyze postulated accidents and develop radiological source terms for each of the 15 bins into which the Containment Event Tree endstates had been grouped. In reviewing the submittal, the staff noticed that the predicted timing for various events, and in particular for Sequence 2, which was a dominant contributor to plant risk, differed significantly from MAAP results presented in the IPE. In response to an RAI, SNC clarified that the IPE results were based on calculations using MAAP 3.0B BWR, Revision 8.01. Differences between results for Sequence 2 in the new submittal and the IPE were attributed to changes in MAAP system models (e.g., improved modeling of the automatic depressurization system, which prolongs operation of the reactor core isolation cooling system) and to changes to the MAAP input parameter file to reflect plant modifications (e.g., the power uprate, instrument setpoint modifications, etc.). Source terms calculated for this submittal were incorporated as input to the NRC-developed MACCS2 code.

SNC's point-estimate source term for selected sequences was reviewed and found to either be in reasonable agreement with or higher than the NUREG-1150 (NRC 1990) Peach Bottom estimates for the closest corresponding release scenarios.

The MACCS2 input used site-specific meteorological data processed from measurements taken hourly in 1997. These data were collected at the site meteorological tower. Hence, the meteorological data are applicable to the site. In addition, SNC performed calculations comparing meteorological data for the years 1995 through 1997. Results indicate that 1997 data were conservative for the 3-year period from 1995 through 1997.

The population distribution used as input to the MACCS2 analyses is based on the 1990 sector population data for HNP provided in NUREG/CR-6525 (SECPOP90; NRC 1997b). Transient populations were not considered because of the rural setting of HNP and the small assumed transient population within 80 km (50 mi) of the site. The site-specific growth rates for the period between 1990 and 2000, which were obtained from census information,[a] were used to estimate a constant growth rate applicable out to 2040. Population growth within a 80-km (50-mi) radius of the site was projected by using the SECPOP90 computer program.

(a) Personal communications on April 2, 1999, between M. Sik, Georgia Governor's Office of Planning and Budget, and J. B. Hovey, Tetra Tech NUS, Inc., Aiken, South Carolina; Subject: 1980 and 1990 Census Counts and 2000 and 2010 Population Projections, 1997 Estimates.

In the original submittal, SNC only projected the population growth out to the end of 2030. At the request of the NRC, SNC projected the population growth out to the end of the license renewal period (2034 for HNP Unit 1 and 2038 for HNP Unit 2), assuming the same constant growth rate. This resulted in a greater population than that used in the SAMA analysis (4 percent higher for 2034 and 8 percent higher for 2038, relative to 2030). Correspondingly, a SAMA analysis using this larger population would result in a 4 percent greater benefit for HNP Unit 1 and an 8 percent greater benefit for HNP Unit 2. However, this would not change the conclusions of the SAMA analyses.

The staff concludes that the above methods and assumptions for the population growth estimates are reasonable and acceptable for the purposes of the SAMA evaluation.

Evacuation modeling was based on a site-specific evacuation study performed by SNC in 1996 (SNC 1996b). SNC assumed that 95 percent of the population within the evacuation zone (extending out to 16 km [10 mi] from the plant) would start moving 45 minutes after declaration of a General Emergency at a radial speed of 2.5 m/s (8.2 ft/s). SNC also assumed that 5 percent of the population would not evacuate. This assumption is conservative relative to the NUREG-1150 study (NRC 1990), which assumed evacuation of 99.5 percent of the population within the emergency planning zone.

In response to an RAI regarding the validity of the evacuation assumption for future years, SNC noted that risk estimates for the HNP site are relatively insensitive to evacuation assumptions because of its rural siting (the 0-16 km [0-10 mi] population is 2 percent of the 0-80 km [0-50 mi] population). Furthermore, SNC observed that conservative assumptions were selected in its evacuation calculations. For example, the assumed evacuation times corresponded to the speed of the slowest subpopulation (special needs persons under adverse conditions), which is approximately half of the evacuation speed indicated for the general population (under adverse conditions).

Evacuation notification is assumed to take place at the times specified for declaring a General Emergency. In a response to an RAI, SNC provided the times at which a General Emergency would be declared. For Level 2 Sequences 4 and 5, these times are simultaneous to the predicted time for the core to be uncovered. For Sequence 2, a General Emergency is declared as soon as the operators realize that they have a station blackout with no possibility of obtaining offsite or onsite power to restore decay-heat-removal systems. In Sequence 11, an ATWS has occurred, the main steam isolation valves have closed and the standby liquid control system has failed to inject. A General Emergency is declared based on a transient occurring with failure of a core shutdown system and containment failure likely. In Sequence 15, there are no water-injection capabilities available. Core damage and vessel failure are unavoidable. A General Emergency is declared when two of the three fission product boundaries (fuel cladding, reactor vessel, and containment) have failed and the failure of the third boundary is likely. For these scenarios, the reported times seem reasonable. Hence, the staff concludes that the evacuation assumptions and analysis are reasonable and acceptable for the purposes of the SAMA evaluation.

Site-specific economic data requiring spatial distributions as input to MACCS2 were prepared by specifying the data for each of the 29 counties within 80 km (50 mi) of the plant. The values used in each of the 160 sectors surrounding the plant corresponded to the county that made up a majority of the land in that sector. When no single county represented a majority of the sector, conglomerate data (weighted by the fraction of each county in the sector) were developed. For the remaining economic data, generic data were provided. Agricultural production information was taken from the 1997 Agricultural Census (USDA 1998) and the Atkinson County [Georgia] Extension Service.

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

5.2.3 Potential Design Improvements

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

5.2.3.1 Process for Identifying Potential Design Improvements

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

- reviews of SAMA analyses submitted in support of original licensing and license renewal activities for other operating nuclear power plants and advanced light-water reactor plants
- reviews of other NRC and industry documentation discussing potential plant improvements
- review of the plant-specific insights from the HNP IPE and IPEEE.

Table 6 in Attachment F to the ER lists the 114 candidate improvements extracted from the above reviews.

SNC performed a qualitative screening of the initial list of SAMAs using the following criteria:

- The SAMA is not applicable to HNP due to design differences (not applicable to the BWR/4/Mk I design).

- The SAMA was related to the mitigation of recirculation pump seal failures or an interfacing system loss-of-coolant accident (ISLOCA). These types of events are not significant risk contributors for BWRs. (See NRC Information Notice 92-36 [NRC 1992] and its supplement [NRC 1994b] for information specifically related to ISLOCAs.)

- The SAMA has already been implemented at HNP (or the HNP design meets the intent of the SAMA).

Based on the qualitative screening, only 42 SAMAs were applicable to HNP and were considered of potential value in averting the risk of severe accidents.

5.2.3.2 Staff Evaluation

SNC's efforts to identify potential SAMAs focused primarily on areas associated with internal initiating events. The initial list of SAMAs generally addressed the accident categories that are dominant CDF contributors or issues that tend to have a large impact on a number of accident sequences at HNP. The preliminary review of SNC's SAMA identification process raised some concerns that plant-specific risk contributors were not fully considered. The staff requested additional plant-specific risk information (dominant minimal cut sets and importance measures) to determine if any significant SAMAs might have been overlooked. The SNC response to the RAI indicated that the insights from the HNP IPE, and not the newer HNP PSA, were used in the identification process. There are a few differences in the final results between the IPE and the PSA, but the list of SAMA candidates appears to address the major contributors to risk for both the IPE and the PSA. Although SNC did not take full advantage of the HNP PSA and the capabilities of the detailed model, it made a reasonable effort to search for potential SAMA candidates, using the knowledge and experience of its PRA personnel; reviewing insights from the IPE, IPEEE, and other plant-specific studies; and reviewing plant improvements in previous SAMA analyses. It should be noted that insights from the IPE have already led to the implementation of numerous potential SAMAs at HNP.

The list of 114 candidate SAMAs strongly focuses on hardware changes that tend to be expensive to implement (of the 114 SAMAs, only about 25 percent involve something other than hardware changes, and only two non-hardware SAMA candidates made it through all the screening to the final analysis). While hardware changes may often provide the greatest risk reduction, consideration should be given to other options that provide marginally smaller risk reductions with much smaller implementation costs. This is particularly true when the maximum attainable benefit is relatively small. For example, instead of adding redundant direct current (DC) control power for the plant service water (PSW) pumps, making procedural changes to provide better manual control may gain nearly as much benefit with a significantly smaller implementation cost.

This issue was raised in an RAI. In its response, SNC cited 26 SAMA candidates as examples of where actions other than hardware changes were considered. Of these 26 SAMA candidates, only 3 were eligible for screening; 10 were already implemented at HNP, 8 were associated with recirculation pump seal failures or ISLOCAs (both considered to be too insignificant with respect to BWR risk to pursue), 2 were combined with other SAMAs (hardware changes), and 3 were determined to not be applicable to HNP. Thus, of the 42 SAMA candidates that were applicable to HNP and were of potential value in averting the risk of severe accidents, only 3 (about 7 percent) were not hardware changes.

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

5.2.4 Risk-Reduction Potential of Design Improvements

SNC evaluated the risk-reduction potential of the 42 unique SAMA candidates that were applicable to HNP by first applying a bounding technique. Each SAMA was assumed to completely eliminate all risk. If the implementation costs were greater than the maximum benefit ($500,000; see Section 5.2.6), then the SAMA was screened from further consideration. If the SAMA could not be screened based on this analysis, then a more refined look at the costs and benefits was warranted.

Using this approach, all but 16 SAMAs were eliminated because the cost was expected to exceed the maximum potential benefit. For each of the 16 remaining SAMA candidates, a more detailed conceptual design was prepared along with a more detailed estimated cost. During this analysis, SNC determined that six of the SAMA candidates were adequately covered by existing plant design and procedures. In addition, the detailed estimation revealed that the cost of one of the candidates (SAMA 41) was greater than the $500,000 cost associated with the maximum potential risk benefit. SNC dropped these seven SAMA candidates from further consideration. The nine remaining SAMA candidates are listed in Table 5-5.

Table 5-5. Cost-Benefit Results for Potentially Cost-Effective SAMA Candidates

No.	SAMA	Result of Potential Enhancement	CDF Reduction (percent)	P-Rem Reduction (percent)	Total Benefits	Imple-mentation Costs	Net Benefit
9	Add redundant direct current (DC) power for plant service water (PSW) pumps C & D.	Would increase reliability of PSW by reducing frequency of loss of PSW	0.11	0.07	$500	$97,000	($96,500)
22	Provide reliable power to control building fans.	Would increase availability of control room ventilation upon a loss of power	0.0	0.0	$0	$101,000	($101,000)
25	Add a diesel building switchgear room high-temperature alarm.	Would improve diagnosis of a loss of switchgear room cooling	0.2	1.2	$2492	$100,000	($97,508)
46	Use the fire protection system as a backup source for containment spray.	Would provide redundant containment spray function without the cost of installing a new system	0.0	0.01	$0[a]	$25,000	($25,000)
60	Improve 4.16-kilovolt (kV) bus cross-tie ability.	Would improve alternating current (AC) power reliability	0.0	0.05	$61	$100,000	($99,939)
73	Use fire protection system as a backup source for diesel cooling.	Would provide a redundant and diverse source of cooling for diesel generators	0.17	1.01	$2098	$126,000	($123,902)
78	Provide DC power to the 120/240-V vital AC system from station battery instead of its own battery.	Would increase the reliability of the 120-Vac buses	0.0	0.0	$78	$106,360	($106,282)

(a) Although there would be a non-zero benefit for this SAMA, the value is so low that it is approximately zero.

Table 5-5. (contd)

No.	SAMA	Result of Potential Enhancement	CDF Reduction (percent)	P-Rem Reduction (percent)	Total Benefits	Imple-mentation Costs	Net Benefit
99	Implement internal flood prevention and mitigation enhancements.	Would reduce the consequences of internal flooding	0.03	0.0	$98	$325,000	($324,902)
105	Proceduralize intermittent operation of the high-pressure coolant injection (HPCI) system.	Would allow extended duration of HPCI availability	0.0	0.0	$0	$22,200	($22,200)

Note: All benefits and costs are on a per unit basis.

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5.2.6.1 SNC Evaluation

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5.2.6.2 Staff Evaluation

5.3 References