

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

(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.

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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. 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: "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, 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

1 Severe Accidents: Based on information in the GEIS, the Commission found: “The probability
2 weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to
3 groundwater, and societal and economic impacts from severe accidents are small for all plants.
4 However, alternatives to mitigate severe accidents must be considered for all plants that have
5 not considered such alternatives.”
6

7 The staff has not identified any significant new information with regard to the consequences
8 from severe accidents during its independent review of the SNC ER, the staff’s site visit, the
9 scoping process, or its evaluation of other available information. Therefore, the staff concludes
10 that there are no impacts of severe accidents beyond those discussed in the GEIS. However, in
11 accordance with 10 CFR 51.53(c)(3)(ii)(L), the staff has reviewed severe accident mitigation
12 alternatives (SAMAs) for HNP. The results of its review are discussed in Section 5.2.
13

14 **5.2 Severe Accident Mitigation Alternatives**

15
16 Title 10 of the Code of Federal Regulations, Part 51.53(c)(3)(ii)(L) requires that license renewal
17 applicants consider alternatives to mitigate severe accidents if the staff has not previously
18 evaluated SAMAs for the applicant’s plant in an environmental impact statement or related
19 supplement or in an environmental assessment. The purpose of this consideration is to ensure
20 that plant design changes with the potential for improving severe accident safety performance
21 are identified and evaluated. SAMAs have not been previously considered for HNP; therefore,
22 the following sections address those alternatives.
23

24 **5.2.1 Introduction**

25
26 SNC submitted an assessment of SAMAs for HNP as part of the ER (SNC 2000a). This
27 assessment was based on the *Hatch 1 Probabilistic Safety Assessment* (PSA), Revision 0 (an
28 updated version of the Individual Plant Examination [IPE, SNC 1992]) for core damage
29 frequency (CDF) estimation and containment performance, and a separate Level 3 model for
30 the ER SAMA risk determination. In identifying and evaluating potential SAMAs, SNC
31 considered the insights from the HNP IPE and Individual Plant Examination for External Events
32 (IPEEE, SNC 1996a) as well as several recent SAMA analyses for other plants (Limerick, Watts
33 Bar, and Comanche Peak) and other industry documentation, such as NUREG-1560 (NRC
34 1997a), NUREG-1462 (NRC 1994), and the GEIS (NRC 1996; 1999), that discuss potential
35 plant improvements. SNC identified and evaluated 114 SAMA candidates. As discussed
36 below, this list was reduced to 42 unique SAMA candidates because the remainder were either
37 not applicable to boiling-water reactors (BWRs), related to phenomena that are not risk-
38 significant in BWRs, or similar to other SAMAs being considered. Other SAMAs were excluded

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1 because they had already been implemented at HNP to address insights and recommendations
2 from the HNP PSA and IPE. The study concluded that none of the remaining SAMAs was cost
3 beneficial.

4
5 Based on a review of the SAMA assessment, the NRC issued a request for additional
6 information (RAI) to SNC by letter dated May 30, 2000 (NRC 2000a). Major issues concerned
7 the process used by the license renewal applicant to identify potential SAMAs, the
8 determination and documentation of the risk profile used in the analysis process, the
9 determination of the risk benefits, and the bases for the SAMA implementation costs. SNC
10 submitted additional information by letters dated July 26, 2000 (SNC 2000b), and August 31,
11 2000 (SNC 2000c), clarifying its approach for SAMA identification, risk quantification and
12 documentation, and SAMA implementation and benefit quantification. This response addressed
13 the staff's concerns and reaffirmed that none of the remaining SAMAs would be cost-beneficial.

14
15 An assessment of SAMAs for HNP is presented below.

16 17 **5.2.2 Estimate of Risk for HNP**

18
19 SNC's estimates of offsite risk at HNP are summarized below. The summary is followed by a
20 review of SNC's risk estimates.

21 22 **5.2.2.1 SNC's Risk Estimates**

23
24 The SAMA analysis is based on two distinct analyses: 1) the HNP PSA, Revision 0 (an update
25 of the HNP Probabilistic Risk Assessment (PRA)/IPE model), and 2) a Level 3 analysis
26 developed specifically for the ER SAMA analyses. The HNP PSA is a conversion of the IPE
27 from the "large event tree, small fault tree" approach to the "linked fault tree" approach. The
28 new model incorporated new information on equipment performance, plant configuration
29 changes, and refinements in PRA modeling techniques. It contains a Level 1 analysis to
30 determine the CDF and a Level 2 analysis to determine containment performance during severe
31 accidents. The Level 1 analysis includes only internal events. Although SNC did not include
32 the results of the IPEEE, it did review the IPEEE as part of Phase I of its SAMA evaluation. The
33 total CDF for internal events is only 1.6E-5 per reactor year (ry) and the Large Early Release
34 Frequency (LERF) is 2.7E-6/ry. The breakdown of CDF is provided in Table 5-3. As shown in
35 this table, the current analyses show that Loss of Feedwater events are a dominant contributor
36 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 ($1.79 \times 10^{-6}/\text{yr}$).

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

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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:

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

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

- C 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.

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

10
11 The revised CDF estimated for HNP is still comparable to values estimated for other BWR3/4
12 plants. Figure 11.2 of NUREG-1560 (NRC 1997a) shows that the total CDFs for these plants
13 range from $9E-8/ry$ to $8E-5/ry$, with an average value of $2E-5/ry$.

14
15 SNC submitted an IPEEE by letter dated January 26, 1996 (SNC 1996a), in response to
16 Supplement 4 of Generic Letter 88-20. SNC did not identify any fundamental weaknesses or
17 vulnerabilities to severe accident risk in regard to the external events related to seismic, fire,
18 high winds, floods, transportation and nearby facility accidents, and other external hazards. In a
19 letter dated October 23, 2000, the staff concluded that the submittal met the intent of
20 Supplement 4 to Generic Letter 88-20 (NRC 2000b). SNC chose not to include the results of its
21 analysis in the estimate of CDF. In its response to an RAI on how plant-specific external event
22 insights were considered, SNC stated that, based on its review of the HNP IPEEE and
23 NUREG-1560 (NRC 1997a) during Phase I of the SAMA evaluation, it identified three SAMAs
24 associated with external events. Two had already been implemented at HNP and one did not
25 pass the initial screening criteria. The largest CDF contributor examined in the IPEEE was
26 internal fires, which contributed $7.5 E-06/ry$ for HNP Unit 1 and $5.4 E-06/ry$ for HNP Unit 2. A
27 staff review of the risk dominant fire zones revealed that the CDF from a fire in a single zone
28 was typically an order of magnitude less than the CDF calculated for internal events. Therefore,
29 there is reasonable assurance that the risk associated with a fire would be bounded by the CDF
30 calculated for internal events. The staff also reviewed the Fire Submittal Screening Review of
31 HNP (an attachment to NRC 2000b) and did not identify any additional alternatives that needed
32 to be further evaluated by the applicant. The staff finds SNC's consideration of external events
33 for the purpose of this SAMA review acceptable.

34
35 The HNP IPE model included Level 2 components. Hence, the conversion to the linked fault
36 tree method impacted the Level 2 results. Differences in the Level 2 results were also impacted
37 by factors such as: (1) a power uprate, and (2) a new version of the Modular Accident Analysis
38 Program (MAAP) code, which was used to estimate release fractions and provide containment
39 analysis details.

40

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1 The process used by SNC to extend the containment performance (Level 2) portion of the PSA
2 to the offsite consequence (Level 3) assessment was reviewed. This included consideration of
3 the source terms used to characterize fission product releases for each containment release
4 mode and the major inputs and assumptions used in the offsite consequence analyses. SNC
5 used Version 3.0B BWR, Revision 10, of the MAAP code to analyze postulated accidents and
6 develop radiological source terms for each of the 15 bins into which the Containment Event
7 Tree endstates had been grouped. In reviewing the submittal, the staff noticed that the
8 predicted timing for various events, and in particular for Sequence 2, which was a dominant
9 contributor to plant risk, differed significantly from MAAP results presented in the IPE. In
10 response to an RAI, SNC clarified that the IPE results were based on calculations using MAAP
11 3.0B BWR, Revision 8.01. Differences between results for Sequence 2 in the new submittal
12 and the IPE were attributed to changes in MAAP system models (e.g., improved modeling of the
13 automatic depressurization system, which prolongs operation of the reactor core isolation
14 cooling system) and to changes to the MAAP input parameter file to reflect plant modifications
15 (e.g., the power uprate, instrument setpoint modifications, etc.). Source terms calculated for
16 this submittal were incorporated as input to the NRC-developed MACCS2 code.

17
18 SNC's point estimate source term for selected sequences was reviewed and found to either be
19 in reasonable agreement with or higher than the NUREG-1150 (NRC 1990) Peach Bottom
20 estimates for the closest corresponding release scenarios.

21
22 The MACCS2 input used site-specific meteorological data processed from measurements taken
23 hourly in 1997. These data were collected at the site meteorological tower. Hence, the
24 meteorological data are applicable to the site. In addition, SNC performed calculations
25 comparing meteorological data for the years 1995 through 1997. Results indicate that 1997
26 data were conservative for the 3-year period from 1995 through 1997.

27
28 The population distribution used as input to the MACCS2 analyses is based on the 1990 sector
29 population data for HNP provided in NUREG/CR-6525 (SECPOP90; NRC 1997b). Transient
30 populations were not considered because of the rural setting of HNP and the small assumed
31 transient population within 80 km (50 mi) of the site. The site-specific growth rates for the
32 period between 1990 and 2000, which were obtained from census information^(a), were used to
33 estimate a constant growth rate applicable out to 2040. Population growth within a 80-km
34 (50-mi) radius of the site was projected by using the SECPOP90 computer program.

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

1 In the original submittal, SNC only projected the population growth out to the end of 2030. At
2 the request of the NRC, SNC projected the population growth out to the end of the license
3 renewal period (2034 for HNP Unit 1 and 2038 for HNP Unit 2), assuming the same constant
4 growth rate. This resulted in a greater population than that used in the SAMA analysis
5 (4 percent higher for 2034 and 8 percent higher for 2038, relative to 2030). Correspondingly, a
6 SAMA analysis using this larger population would result in a 4 percent greater benefit for HNP
7 Unit 1 and an 8 percent greater benefit for HNP Unit 2. However, this would not change the
8 conclusions of the SAMA analyses.

9
10 The staff concludes that the above methods and assumptions for the population growth
11 estimates are reasonable and acceptable for the purposes of the SAMA evaluation.

12
13 Evacuation modeling was based on a site-specific evacuation study performed by SNC in 1996
14 (SNC 1996b). SNC assumed that 95 percent of the people within the evacuation zone
15 (extending out to 16 km [10 mi] from the plant) would start moving 45 minutes after declaration
16 of a general emergency at a radial speed of 2.5 m/s (8.2 ft/s). SNC also assumed that
17 5 percent of the population would not evacuate. This assumption is conservative relative to the
18 NUREG-1150 study (NRC 1990), which assumed evacuation of 99.5 percent of the population
19 within the emergency planning zone.

20
21 In response to an RAI regarding the validity of the evacuation assumption for future years, SNC
22 noted that risk estimates for the HNP site are relatively insensitive to evacuation assumptions
23 because of its rural siting (the 0-16 km [0-10 mi] population is 2 percent of the 0-80 km [0-50 mi]
24 population). Furthermore, SNC observed that conservative assumptions were selected in its
25 evacuation calculations. For example, the assumed evacuation times corresponded to the
26 speed of the slowest subpopulation (special needs persons under adverse conditions), which is
27 approximately half of the evacuation speed indicated for the general population (under adverse
28 conditions).

29
30 Evacuation notification is assumed to take place at the times specified for declaring a general
31 emergency. In a response to an RAI, SNC provided the times at which a general emergency
32 would be declared. For Level 2 Sequences 4 and 5, these times are simultaneous to the
33 predicted time for the core to be uncovered. For Sequence 2, a general emergency is declared
34 as soon as the operators realize that they have a station blackout with no possibility of obtaining
35 offsite or onsite power to restore decay-heat-removal systems. In Sequence 11, an ATWS has
36 occurred, the main steam isolation valves have closed and the standby liquid control system
37 has failed to inject. A general emergency is declared based on a transient occurring with failure
38 of a core shutdown system and containment failure likely. In Sequence 15, there are no water
39 injection capabilities available. Core damage and vessel failure are unavoidable. A general
40 emergency is declared when two of the three fission product boundaries (fuel cladding, reactor
41 vessel, and containment) have failed and the failure of the third boundary is likely. For these

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1 scenarios, the reported times seem reasonable. Hence, the staff concludes that the evacuation
2 assumptions and analysis are reasonable and acceptable for the purposes of the SAMA
3 evaluation.

4
5 Site-specific economic data requiring spatial distributions as input to MACCS2 were prepared by
6 specifying the data for each of the 29 counties within 80 km (50 mi) of the plant. The values
7 used in each of the 160 sectors surrounding the plant corresponded to the county that made up
8 a majority of the land in that sector. When no single county represented a majority of the sector,
9 conglomerate data (weighted by the fraction of each county in the sector) were developed. For
10 the remaining economic data, generic data were provided. Agricultural production information
11 was taken from the 1997 Agricultural Census (USDA 1998) and the Atkinson County [Georgia]
12 Extension Service.

13
14 The staff concludes that the methodology used by SNC to estimate the CDF and offsite
15 consequences for HNP provides an acceptable basis from which to proceed with an
16 assessment of risk reduction potential for candidate SAMAs. Accordingly, the staff based its
17 assessment of offsite risk on the CDF and offsite doses reported by SNC.

18 19 **5.2.3 Potential Design Improvements**

20
21 The process for identifying potential plant improvements, an evaluation of that process, and the
22 improvements evaluated in detail by SNC are discussed in this section.

23 24 **5.2.3.1 Process for Identifying Potential Design Improvements**

25
26 SNC's process for identifying potential plant improvements (SAMAs) consisted of the following
27 elements:

- 28
29 C reviews of SAMA analyses submitted in support of original licensing and license renewal
30 activities for other operating nuclear power plants and advanced light water reactor plants
31
- 32 C reviews of other NRC and industry documentation discussing potential plant improvements
33
- 34 C review of the plant-specific insights from the HNP IPE and IPEEE.
35

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

38
39 SNC performed a qualitative screening of the initial list of SAMAs using the following criteria:
40

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

3
4 C The SAMA was related to the mitigation of an interfacing system loss of coolant accident
5 (ISLOCA). NRC Information Notice 92-36 and its supplement were cited as characterizing
6 the risk contributions of ISLOCA for BWRs as being very small.

7
8 C The SAMA has already been implemented at HNP (or the HNP design meets the intent of
9 the SAMA).

10
11 Based on the qualitative screening, only 42 SAMAs were applicable to HNP and were
12 considered of potential value in averting the risk of severe accidents.

13
14 **5.2.3.2 Staff Evaluation**

15
16 SNC's efforts to identify potential SAMAs focused primarily on areas associated with internal
17 initiating events. The initial list of SAMAs generally addressed the accident categories that are
18 dominant CDF contributors or issues that tend to have a large impact on a number of accident
19 sequences at HNP. The preliminary review of SNC's SAMA identification process raised some
20 concerns that plant-specific risk contributors were not fully considered. The staff requested
21 additional plant-specific risk information (dominant minimal cut sets and importance measures)
22 to determine if any significant SAMAs might have been overlooked. The SNC response to the
23 RAI indicated that the insights from the HNP IPE, and not the newer HNP PSA, were used in
24 the identification process. There are a few differences in the final results between the IPE and
25 the PSA, but the list of SAMA candidates appears to address the major contributors to risk for
26 both the IPE and the PSA. Although SNC did not take full advantage of the HNP PSA and the
27 capabilities of the detailed model, it made a reasonable effort to search for potential SAMA
28 candidates, using the knowledge and experience of its PRA personnel; reviewing insights from
29 the IPE, IPEEE, and other plant-specific studies; and reviewing plant improvements in previous
30 SAMA analyses. It should be noted that insights from the IPE have already led to the
31 implementation of numerous potential SAMAs at HNP.

32
33 The list of 114 candidate SAMAs strongly focuses on hardware changes that tend to be
34 expensive to implement (of the 114 SAMAs, only about 25 percent involve something other than
35 hardware changes, and only two non-hardware SAMA candidates made it through all the
36 screening to the final analysis). While hardware changes may often provide the greatest risk
37 reduction, consideration should be given to other options that provide marginally smaller risk
38 reductions with much smaller implementation costs. This is particularly true when the maximum
39 attainable benefit is relatively small. For example, instead of adding redundant direct current
40 (DC) control power for the PSW pumps, making procedural changes to provide better manual
41 control may gain nearly as much benefit with a significantly smaller implementation cost.

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

9
10 The NRC notes that the set of SAMAs submitted is not all inclusive, since additional, possibly
11 even less expensive, design alternatives can always be postulated. However, the staff
12 concludes that the benefits of any additional modifications are unlikely to exceed the benefits of
13 the modifications evaluated and that the alternative improvements would not likely cost less
14 than the least expensive alternatives evaluated, when the subsidiary costs associated with
15 maintenance, procedures, and training are considered. On this basis, the NRC concludes that
16 the set of potential SAMA alternatives identified by SNC is acceptable.

17 18 **5.2.4 Risk Reduction Potential of Design Improvements**

19
20 SNC evaluated the risk reduction potential of the 42 unique SAMA candidates that were
21 applicable to HNP by first applying a bounding technique. Each SAMA was assumed to
22 completely eliminate all risk. If the implementation costs were greater than the maximum
23 benefit (\$500,000, see Section 5.2.6), then the SAMA was screened from further consideration.
24 If the SAMA could not be screened based on this analysis, then a more refined look at the costs
25 and benefits was warranted.

26
27 Using this approach, all but 16 SAMAs were eliminated because the cost was expected to
28 exceed the maximum potential benefit. For each of the 16 remaining SAMA candidates, a more
29 detailed conceptual design was prepared along with a more detailed estimated cost. During this
30 analysis, SNC determined that six of the SAMA candidates were adequately covered by existing
31 plant design and procedures. In addition, the detailed estimation revealed that the cost of one
32 of the candidates (SAMA 41) was greater than the \$500,000 cost associated with the maximum
33 potential risk benefit. SNC dropped these seven SAMA candidates from further consideration.
34 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	Implementation 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 pumps C & D	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	\$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	\$2,492	\$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	\$2,098	\$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	Implementation 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.

1 For each of these SAMAs, a risk reduction analysis was performed. The specific impacts on the
2 CDF and LERF models were identified, the appropriate model elements were changed to reflect
3 the plant or procedure enhancement, and the models were requantified. Table 5-5 shows the
4 percent reductions in the CDF and person-rem public exposure for each SAMA.

5
6 The evaluation of the SAMA risk-reduction potentials did not consider uncertainties. The HNP
7 PSA used in the risk-reduction evaluation does not lend itself to propagating uncertainty;
8 therefore, an uncertainty analysis was not performed. The uncertainties in the PSA, risk-
9 reduction estimates, and costs all contribute to uncertainties in the value-impact analyses for
10 each SAMA. Factors of 3 to 5 are common for the Level 1 PSA alone. Even larger
11 uncertainties are common for the Level 2 and Level 3 analyses. However, the margins between
12 the costs and the benefits for the SAMAs presented in Table 5-5 are so large that even if the
13 risk reduction benefits were a factor of 10 greater, all of the SAMAs would still be eliminated.

14
15 The NRC staff concludes that the risk-impact analyses performed for the final nine SAMA
16 candidates were conducted according to accepted PRA practices and are acceptable and
17 appropriate for the SAMA analysis.

18 19 **5.2.5 Cost Impacts of Candidate Design Improvements**

20
21 SNC developed a preliminary cost estimate for each of the 42 unique SAMA candidates as part
22 of a cost-screening analysis. The screening criterion was established at a cost of \$500,000
23 based on the analysis of the maximum potential benefit. Thus, if a SAMA cost more than
24 \$500,000, there was no potential for being cost-beneficial, even if it eliminated all risk.

25
26 The preliminary cost estimates were developed to determine which SAMA candidates would
27 clearly cost more than \$500,000 and could readily be dismissed. The cost estimates were
28 based on the total costs associated with performing engineering, procurement, and
29 construction. The cost history for similar modifications at the plant or at other plants was
30 considered in developing the estimates.

31
32 Using the \$500,000 screening value, 26 candidate SAMAs were eliminated. For the
33 16 remaining SAMA candidates, a more detailed conceptual design was prepared along with a
34 more detailed cost estimate based on the same set of cost elements considered before plus
35 training costs. During the detailed analysis, SNC determined that six of the candidate SAMAs
36 were adequately covered by existing plant design and procedures. SNC found that another
37 candidate SAMA was more expensive than the \$500,000 cutoff value. SNC eliminated these
38 seven candidate SAMAs from further consideration. Table 5-5 shows the cost-benefit analysis
39 results for the nine remaining SAMA candidates.

40

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1 The cost estimates are reasonable and in most cases are probably conservative (low) in that
2 they do not consider the cost of replacement power during extended outages to implement the
3 modifications and do not include contingency costs associated with unforeseen implementation
4 obstacles. Where applicable, costs were determined on a dual-unit basis (rather than doubling
5 a single-unit estimate) to give a more accurate overall cost estimate.
6

7 The staff concludes that the cost estimates are sufficient and appropriate for use in the SAMA
8 evaluations.
9

10 **5.2.6 Cost-Benefit Comparison**

11
12 The staff's evaluation of SNC's cost-benefit analysis is described in the following sections.
13

14 **5.2.6.1 SNC Evaluation**

15
16 The methodology used by SNC was based primarily on NRC's guidance for performing cost-
17 benefit analysis, i.e., *Regulatory Analysis Technical Evaluation Handbook*, NUREG/BR-0184
18 (NRC 1997c). The guidance involves determining the net value for each SAMA according to the
19 following formula:
20

$$21 \text{Net Value} = (\text{APE} + \text{AOC} + \text{AOE} + \text{AOSC}) - \text{COE}$$

22
23 where APE = present value of averted public exposure (\$)
24 AOC = present value of averted offsite property damage costs (\$)
25 AOE = present value of averted occupational exposure (\$)
26 AOSC = present value of averted onsite costs (\$)
27 COE = cost of enhancement (\$)
28

29 If the net value of a SAMA is negative, the cost of implementing the SAMA is larger than the
30 benefit associated with the SAMA and it is not considered cost-beneficial. The same analytical
31 approach was used by SNC for the initial screening of the SAMAs. However, for the screening
32 process SNC calculated the maximum averted costs assuming that all severe accident costs
33 were eliminated. SNC's derivation of each of the associated costs is summarized below.
34

35 **Averted Public Exposure (APE)**

36
37 SNC called this cost the Offsite Exposure Cost. Averted public exposure costs were calculated
38 using the following formula:
39

1 APE = Annual reduction in public exposure () person-rem/ry)
 2 x monetary equivalent of unit dose (\$2,000 per person-rem)
 3 x present value conversion factor (10.76, based on a 20-year period with a 7 percent
 4 discount rate)
 5

6 As stated in NUREG/BR-0184 (NRC 1997c), it is important to note that the monetary value of
 7 the public health risk after discounting does not represent the expected reduction in public
 8 health risk due to a single accident. Rather, it is the present value of a stream of potential
 9 losses extending over the remaining lifetime (in this case, the renewal period) of the facility.
 10 Thus, it reflects the expected annual loss due to a single accident, the possibility that such an
 11 accident could occur at any time over the renewal period, and the effect of discounting these
 12 potential future losses to present value. For the purposes of initial screening (severe accident
 13 costs eliminated), SNC calculated an APE of \$72,565.
 14

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

16
 17 SNC called this the Offsite Economic Cost. Averted offsite property damage costs were
 18 calculated using the following formula:
 19

20 AOC = Annual CDF reduction
 21 x offsite economic costs associated with a severe accident (on a per event basis)
 22 x present value conversion factor
 23

24 For the purposes of initial screening (severe accident costs eliminated), SNC cited an annual
 25 offsite economic risk of \$9,262 based on the Level 3 risk analysis. This results in a discounted
 26 value of \$99,659.
 27

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

29
 30 SNC calls this the Onsite Exposure Cost. Averted occupational exposure costs were calculated
 31 using the following formula:
 32

33 AOE = Annual CDF reduction
 34 x occupational exposure per core damage event
 35 x monetary equivalent of unit dose
 36 x present value conversion factor
 37

38 SNC derived the values for averted occupational exposure from information provided in
 39 Section 5.7.3 of the regulatory analysis handbook (NRC 1997c). Best estimate values provided
 40 for immediate occupational dose (3,300 person-rem) and long-term occupational dose
 41 (20,000 person-rem over a 10-year cleanup period) were used. The present value of these

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1 doses was calculated using the equations provided in the handbook in conjunction with a
2 monetary equivalent of unit dose of \$2,000 per person-rem, a real discount rate of 7 percent,
3 and a time period of 20 years to represent the license renewal period. For the purposes of
4 initial screening (severe accident costs eliminated), SNC calculated an AOE of \$6,237.

6 **Averted Onsite Costs (AOSC)**

7
8 Averted onsite costs include averted cleanup and decontamination costs and averted power
9 replacement costs. Repair and refurbishment costs are considered for recoverable accidents
10 only and not for severe accidents. SNC derived the values for AOSC based on information
11 provided in Section 5.7.6 of the regulatory analysis handbook (NRC 1997c).

12
13 SNC divided this cost element into two parts, the Onsite Cleanup and Decontamination Cost
14 (also commonly referred to as averted cleanup and decontamination costs [ACC]) and the
15 Replacement Power Cost.

16
17 Averted cleanup and decontamination costs are calculated using the following formula:

$$\begin{aligned} 18 \text{ ACC} &= \text{Annual CDF reduction} \\ 19 &\quad \times \text{present value of cleanup costs per core damage event} \\ 20 &\quad \times \text{present value conversion factor} \end{aligned}$$

21
22
23 The total cost of cleanup and decontamination subsequent to a severe accident is estimated in
24 the regulatory analysis handbook as \$1.1E+9 (undiscounted). This value was converted to
25 present costs over a 10-year cleanup period and integrated over the term of the proposed
26 license extension. For the purposes of initial screening (severe accident costs eliminated), SNC
27 calculated an ACC of \$193,973.

28
29 Long-term replacement power costs (RPC) are calculated using the following formula:

$$\begin{aligned} 30 \text{ RPC} &= \text{Annual CDF reduction} \\ 31 &\quad \times \text{present value of replacement power for a single event} \\ 32 &\quad \times \text{factor to account for remaining service years for which replacement power is required} \\ 33 &\quad \times \text{reactor power scaling factor} \end{aligned}$$

34
35
36 For the purposes of initial screening (severe accident costs eliminated), SNC calculated an RPC
37 of \$120,041. The total averted cost for the screening process is \$492,476, which SNC rounded
38 up to \$500,000.

40 **SNC Results**

1 The cost-benefit results for the individual analysis of the final nine SAMA candidates are
2 presented in Table 5-5. All of the SAMAs have significantly large negative net values. SNC
3 concluded that implementation of any of these SAMAs is not justified because the costs of
4 implementation greatly exceed the benefits. As such SNC has decided not to pursue any of
5 these SAMAs further.

6 7 **5.2.6.2 Staff Evaluation**

8
9 The cost-benefit analysis conducted by SNC was based primarily on the NRC's *Regulatory*
10 *Analysis Technical Evaluation Handbook* (NRC 1997c). No deviations were found. The staff
11 concludes that the cost of implementing any of the nine SAMAs would far exceed the estimated
12 benefit, with a margin of about a factor of 20. Use of a 3 percent discount rate in place of the
13 7 percent discount rate used in SNC's analysis would increase net values, but would not lead to
14 the identification of any cost-beneficial SAMAs. Similarly, implementing any of the SAMAs in
15 the near term instead of waiting until the start of the license renewal period (thereby extending
16 the period in the value-impact analysis) would not increase the net benefit sufficiently to make
17 any of the SAMA candidates cost-beneficial.

18 19 **5.2.7 Conclusions**

20
21 SNC compiled a list of 114 SAMA candidates using as resources SAMA analyses submitted in
22 support of licensing activities for other nuclear power plants, NRC and industry documents
23 discussing potential plant improvements, and the plant-specific insights from the HNP IPE and
24 IPEEE. A qualitative screening removed those SAMA candidates that (1) did not apply to HNP
25 due to design differences, (2) were related to the mitigation of recirculation pump seal failures or
26 ISLOCAs (not significant risk contributors for BWRs), or (3) had already been implemented at
27 HNP. Only 42 SAMA candidates survived this screening process.

28
29 Using the HNP PSA and a Level 3 analysis developed specifically for SAMA evaluations, a
30 maximum obtainable benefit of about \$500,000 was calculated. This value was used as a
31 second screening that eliminated the SAMA candidates whose cost to implement would exceed
32 the maximum obtainable benefit. This process left only 16 SAMA candidates for further
33 analysis.

34
35 For each of these 16 SAMA candidates, a more detailed conceptual design and cost estimate
36 were developed. In doing so, SNC determined that six SAMA candidates were adequately
37 covered by existing plant design and procedures and that another would cost more than
38 \$500,000 to implement. SNC eliminated these seven SAMA candidates from further
39 consideration. The final nine SAMA candidates were processed through a detailed cost-benefit
40 analysis as shown in Table 5-5.

41

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1 The cost-benefit analyses showed that none of the final nine SAMA candidates were cost-
2 beneficial and the negative net benefit margins were large. SNC concluded that there was no
3 justification to implement any of the SAMA candidates and decided not to pursue any of the
4 SAMA candidates any further.

5
6 The staff reviewed the SNC analysis and concluded that the methods used and the
7 implementation of those methods were sound. While there is at least one area of weakness in
8 the analysis (a lack of explicit treatment of uncertainties), the conservative treatment of SAMA
9 benefits and costs, the resulting large negative net benefits and the inherently small baseline
10 risks, support the preliminary conclusion that the SAMA evaluations performed by SNC are
11 reasonable and sufficient for the license renewal submittal.

12
13 Based on its review of SNC's SAMA analyses, it is the staff's preliminary conclusion that none
14 of the candidate SAMAs are cost-beneficial. This conclusion is consistent with the low residual
15 level of risk indicated in the HNP PSA and the fact that HNP has already implemented many
16 plant improvements identified by the IPE and IPEEE.

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