5.0 Environmental Impacts of Postulated Accidents

Environmental issues associated with postulated accidents are discussed in the *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* (GEIS), NUREG-1437, Volumes 1 and 2 (NRC 1996; 1999).[a] The GEIS includes 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 are 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 characteristic.

(2) 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 for Category 1, and therefore, additional plant-specific review of 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

Two classes of accidents are evaluated in the GEIS. These are design-basis accidents (DBAs) and severe accidents, as discussed below.

(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 Addendum 1.

5.1.1 Design-Basis Accidents

In order to receive NRC approval to operate a nuclear power facility, an applicant must submit a safety analysis report (SAR) as part of its application. The SAR presents the design criteria and design information for the proposed reactor and comprehensive data on the proposed site. The SAR also discusses various hypothetical accident situations and the safety features that are provided to prevent and mitigate accidents. The NRC staff reviews the application to determine whether the plant design meets the Commission's regulations and requirements and includes, in part, the nuclear plant design and its anticipated response to an accident.

DBAs are those accidents that both the licensee and the NRC staff evaluate to ensure that the plant can withstand normal and abnormal transients, and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. A number of these postulated accidents are not expected to occur during the life of the plant, but are evaluated to establish the design basis for the preventive and mitigative safety systems of the facility. The acceptance criteria for DBAs are described in 10 CFR Part 50 and 10 CFR Part 100.
The environmental impacts of DBAs are evaluated during the initial license process, and the ability of the plant to withstand these accidents is demonstrated to be acceptable before issuance of the operating license (OL). The results of these evaluations are found in license documentation such as the staff's Safety Evaluation Report (SER), the Final Environmental Statement (FES), the licensee's Updated Final Safety Analysis Report (UFSAR), and Section 5.1 of this supplemental environmental impact statement (SEIS). The licensee is required to maintain the acceptable design and performance criteria throughout the life of the plant including any extended-life operation. The consequences for these events are evaluated for the hypothetical maximum exposed individual; as such, changes in the plant environment will not affect these evaluations. Because of the requirements that continuous acceptability of the consequences and aging management programs be in effect for license renewal, the environmental impacts as calculated for DBAs should not differ significantly from initial licensing assessments over the life of the plant, including the license renewal period. Accordingly, the design of the plant relative to DBAs during the extended period is considered to remain acceptable and the environmental impacts of those accidents were not examined further in the GEIS.

The Commission has determined that the environmental impacts of DBAs are of SMALL significance for all plants because the plants were designed to successfully withstand these accidents. Therefore, for the purposes of license renewal, design-basis events are designated as a Category 1 issue in 10 CFR Part 51, Subpart A, Appendix B, Table B-1. The early resolution of the DBAs make them a part of the current licensing basis of the plant; the current licensing basis of the plant is to be maintained by the licensee under its current license and, therefore, under the provisions of 10 CFR 54.30, is not subject to review under license renewal. This issue, applicable to Surry Power Station, Units 1 and 2, is listed in Table 5-1.

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 Section
POSTULATED ACCIDENTS	
Design-basis accidents (DBAs)	5.3.2; 5.5.1

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 Virginia Electric and Power Company (VEPCo) stated in its Environmental Report (ER; VEPCo 2001a) that it is not aware of any new and significant information associated with the renewal of the Surry Units 1 and 2 OLs. The staff has not identified any significant new information during its independent review of the VEPCo 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 related to this issue beyond those discussed in the GEIS.

5.1.2 Severe Accidents

Severe nuclear accidents are those that are more severe than DBAs because they could result in substantial damage to the reactor core, whether or not there are serious offsite consequences. In the GEIS, the staff assessed the impacts of severe accidents during the license renewal period, using the results of existing analyses and site-specific information to conservatively predict the environmental impacts of severe accidents for each plant during the renewal period.

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 ground water, 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.

Therefore, the Commission has designated mitigation of severe accidents as a Category 2 issue in 10 CFR Part 51, Subpart A, Appendix B, Table B-1. This issue, applicable to Surry Units 1 and 2, 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

The staff has not identified any significant new information with regard to the consequences from severe accidents during its independent review of the VEPCo ER (VEPCo 2001a), 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 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 Surry Units 1 and 2. The results of its review are discussed in Section 5.2.

5.2 Severe Accident Mitigation Alternatives

10 CFR 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 (EIS) or related supplement or in an environmental assessment. The purpose of this consideration is to ensure that plant changes (i.e., hardware, procedures, and training) with the potential for improving severe accident safety performance are identified and evaluated. SAMAs have not been previously considered for Surry Power Station, Units 1 and 2; therefore, the following sections address those alternatives.

5.2.1 Introduction

VEPCo submitted an assessment of SAMAs for Surry Units 1 and 2 as part of the ER (VEPCo 2001a). The assessment was based on the Surry Probabilistic Risk Assessment (PRA), which is an updated version of the Surry Individual Plant Examination (IPE) for internal events (VEPCo 1991), the Surry Individual Plant Examination for External Events (IPEEE) (VEPCo 1994), and supplemental analyses of offsite consequences and economic impacts performed specifically for the SAMA analysis. VEPCo generated a list of 160 candidate SAMAs based on a review of previous SAMA analyses in support of original plant licensing and license renewal, NRC and industry reports discussing potential plant improvements, dominant risk contributors in the plant-specific risk study, and insights provided by VEPCo's PRA staff. VEPCo assessed the costs and benefits associated with each of the potential SAMAs and concluded that none of the candidate SAMAs evaluated were cost-beneficial for Surry Power Station.

Based on a review of the applicant's SAMA assessment, the NRC issued a request for additional information (RAI) to VEPCo by letter dated October 17, 2001 (NRC 2001). Key questions concerned the modifications to the Surry PRA made subsequent to the IPE, treatment of external events in the SAMA analysis, the use of the plant-specific risk study in the SAMA identification process, and the evaluation of costs and benefits for certain SAMAs. VEPCo submitted additional information by letter dated December 10, 2001 (VEPCo 2001b) and by e-mails dated January 15 and January 22, 2002 (NRC 2002) in response to the staff's RAIs. These responses addressed the staff's concerns and reaffirmed the conclusion that none of the SAMAs would be cost-beneficial.

The staff's assessment of SAMAs for Surry Power Station follows.

5.2.2 Estimate of Risk for Surry Power Station

VEPCo's estimates of offsite risk at Surry Power Station are summarized below. The summary is followed by the staff's review of VEPCo's risk estimates.

5.2.2.1 VEPCo's Risk Estimates

Two distinct analyses are combined to form the basis for the risk estimates used in the SAMA analysis: (1) the Surry Level 1 and 2 PRA models, which is an updated version of the IPE, and (2) a supplemental analysis of offsite consequences and economic impacts (essentially a Level 3 PRA model) developed specifically for the SAMA analysis. The Surry PRA Level 1 and 2 models were originally developed in response to the request for an IPE contained in Generic Letter 88-20 (NRC 1988). The Level 1 model was updated in 1994 before performing the IPEEE fire analysis, and again in 1997 to support implementation of the maintenance rule. In addition, before performing the SAMA analysis, a number of changes were made to the Level 2 model to reflect new experimental results, and to provide more consistency with the Level 2 model for VEPCo's North Anna Power Station.

The baseline core damage frequency (CDF) for the purpose of SAMA evaluation is approximately 3.8×10^{-5} per reactor-year, based on the risk assessment for internally initiated events. Although VEPCo did not include the contribution of risk from external events within the Surry Power Station risk estimates, it did account for the potential risk-reduction benefits associated with external events by doubling the estimated benefits for internal events. This is discussed further in Section 5.2.2.2. A breakdown of the CDF is provided in Table 5-3. As shown in this table, loss-of-coolant accidents (LOCAs) contribute about 58 percent, while transients

Table 5-3. Surry Power Station Core Damage Frequency

Initiating Event	Frequency (per reactor-year)
Loss-of-coolant accident (LOCA)	2.2×10^{-5}
Transients	9.3×10^{-6}
Loss of offsite power/station blackout (LOOP/SBO)	2.5×10^{-6}
Steam generator tube rupture (SGTR)	2.3×10^{-6}
Interfacing system LOCA (ISLOCA)	1.6×10^{-6}
Anticipated transient without scram (ATWS)	4.5×10^{-9}
Total CDF from internal events	3.8×10^{-5}

contribute about 25 percent of the total internal events CDF. Anticipated transients without scram (ATWS) are negligible contributors to CDF for Surry Power Station. The frequency associated with the largest releases (i.e., interfacing system LOCA [ISLOCA] and steam generator tube rupture [SGTR]) for Surry Power Station is estimated to be about 3.9×10^{-6} per reactor-year. The station blackout (SBO) contribution to the transients was not explicitly provided in the submittal; however, in response to an RAI, VEPCo provided the frequency and contribution to the total frequency (see Table 5-3). The CDFs cited here and used in the SAMA analysis are best-estimate values. The uncertainty analysis for the updated PRA indicates a 95 percent confidence-level (upper) CDF value of 1.16×10^{-4} per reactor-year, or about three times the best-estimate value. The impact of this uncertainty on the SAMA analysis is discussed in Section 5.2.6.2.

The offsite consequences and economic impact analyses use the MELCOR Accident Consequence Code System 2 (MACCS2), Version 1.12, to determine the offsite risk impacts on the surrounding environment and public. Inputs for this analysis include plant/ site-specific input values for core radionuclide inventory, source term and release fractions, meteorological data, projected population distribution, emergency response evacuation modeling, and economic data. The magnitude of the onsite impacts (in terms of clean-up and decontamination costs and occupational dose) is based on information provided in NUREG/BR-0184 (NRC 1997b).

VEPCo estimated the dose to the population within 80 km (50 mi) of the Surry Power Station from internal initiators to be about 0.18 person-Sv (18 person-rem) per year. Table 5-4 shows the contributions to population dose by containment release mode. SGTRs and ISLOCAs together account for approximately 95 percent of the population dose although they collectively comprise only about 10 percent of the total internal events CDF. This is due to the relatively

Table 5-4. Risk Profile for Surry Power Station

Containment Release Mode	Contribution to Release Frequency[a] (%)	Contribution to Population Dose[b] (%)
Containment intact	59	<0.1
Early containment failure	1	1
Late containment failure	30	4
Containment bypass - SGTR	6	65
Containment bypass - ISLOCA	4	30

(a) Total release frequency for internal events = 3.8×10^{-5} per reactor-year.
(b) Total population dose = 0.18 person-Sv (18 person-rem) per reactor-year.

high fission-product releases in these sequences. Early and late containment failure contribute about 5 percent of the population dose. About 60 percent of the core melt accidents at Surry Power Station do not result in containment failure and have only a minimal contribution to population dose.

5.2.2.2 Review of VEPCo's Risk Estimates

VEPCo's determination of offsite risk at Surry Power Station is based on the following three major elements of analysis:

- the Level 1 and 2 risk models for Surry Power Station that form the basis for the 1991 IPE submittal and the 1994 IPEEE submittal

- the major modifications to the risk model subsequent to the IPE that distinguish the current PRA from the IPE

- the MACCS2 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 VEPCo's risk estimates for the SAMA analysis, as summarized below.

The staff's review of the Surry IPE is described in a staff report dated December 16, 1993 (NRC 1993). 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 VEPCo'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, it primarily focused on the licensee's ability to examine Surry Power Station for severe accident vulnerabilities and not specifically on the detailed findings or quantification estimates. Overall, the staff believed that the Surry 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, sensitivity, and uncertainty analyses. It is important to note that some changes have been made to the Surry risk model since the original IPE was completed and reviewed by the NRC staff. These include both modifications to the models and changes due to plant modification, as discussed below.

A comparison of CDF profiles between the IPE and the updated PRA indicates that the estimate of the CDF for internal events has been reduced from 7.4×10^{-5} per reactor-year to 3.8×10^{-5} per reactor-year. The lower values in the updated PRA are attributed to plant and modeling improvements which have been implemented at Surry Power Station since the IPE was submitted.

The original Level 1 model documented in the 1991 Surry IPE submittal had a CDF of 7.4×10^{-5} per reactor-year (from internally initiated events, including internal flooding). A minor update to the Level 1 model was performed before the licensee completed the IPEEE fire analysis in December 1994.

A significant update to the Level 1 model occurred in 1997 to support implementation of the maintenance rule. A third update to the PRA model occurred in late 1997/early 1998. These updates were performed to incorporate significant plant modifications, correct model errors, and enhance the model with state-of-the-art improvements. Among the individual fault tree models changed or added were those involving auxiliary feedwater, the swing diesel, the station blackout diesel, the ATWS mitigating systems actuation circuitry, the component cooling water system, station service and switchyard buses, and various support systems for balance-of-plant components and backup mitigating functions. Modeling for the loss of emergency switchgear room (ESGR) and loss of 4160-V emergency bus initiating events were also modified, and the human error probability was modified to account for reduced time to hot leg recirculation during large LOCA events. The modified baseline CDF, as of the most recent model changes, is 3.8×10^{-5} per reactor-year.

A comprehensive peer review of the Level 1 and 2 PRA model used in the IPE was completed in August 1991. This review was conducted by a team composed of both VEPCo personnel and outside contractors. In addition, the updated Level 1 PRA model used as a basis for the SAMA analysis was reviewed as the pilot in the Westinghouse Owners Group peer certification effort.

The updated CDF value is lower than most of the original IPE values estimated for other pressurized water reactors (PWRs) with large dry containments. Figure 11.6 of NUREG-1560 (NRC 1997c) shows that the IPE-based total internal events CDF for Westinghouse three-loop plants range from 6×10^{-5} to 4×10^{-4} per reactor-year. However, many of these CDF estimates have similarly been reduced due to modeling and hardware changes subsequent to the respective IPE submittals. Thus, this observation may no longer be significant.

As noted in Table 5-4, SGTR and ISLOCA contribute 6 percent and 4 percent, respectively, to the total release frequency in internal events. Because of the large fission product releases for bypass sequences relative to other release modes, these sequences dominate the Surry Power Station risk profile. The conditional probability of early containment failure is approximately 1 percent, and about 30 percent of core damage sequences are expected to lead to late containment failure. Due to the sub-atmospheric design of the containment, containment isolation failures are relatively insignificant (about 0.3 percent of CDF). With the exception of the somewhat high CDF associated with bypass of the containment, and the lack of credit in the PRA for scrubbing releases from SGTRs, the results of the updated Surry PRA appear to be consistent with those of other IPEs for PWRs with large dry or subatmospheric containments insofar as the general CDF, containment response, and release and risk profiles are concerned.

VEPCo submitted an IPEEE by letter dated December 14, 1994 (VEPCo 1994). VEPCo did not identify any 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 the associated safety evaluation report (NRC 2000), the staff concluded that the IPEEE met the intent of Supplement 4 to Generic Letter 88-20 (NRC 1991).

Although VEPCo used probabilistic risk methods for the seismic and fire portions of the IPEEE, in their SAMA analysis they chose to capture the potential risk benefits associated with external events by doubling the calculated internal events benefits for each SAMA. In assessing the reasonableness of this assumption, the staff considered the relative contribution to the total risk from the various external events based on best available information. The Surry Power Station high winds and external flooding analyses show that the plant is adequately designed to protect against the effects of these natural events. Transportation and nearby facility accidents were not considered to be potential sources of damage at the plant because of the plant's rural location. Other external events were evaluated and found to be insignificant contributors to CDF. Even though VEPCo's doubling of CDF to account for the benefits of a SAMA in external events provides a reasonable numerical estimate of the potential impact, this approach may potentially fail to capture the benefits that could result from specific SAMAs aimed at particular external events. In response to an RAI, VEPCo reasoned that since no external event vulnerabilities in terms of containment bypass or isolation failure were identified in the IPEEE, the offsite consequences can be bounded by the use of an internal events profile. In addition, the CDF cited by VEPCo from external events – approximately 1.3 x 10^{-5} per reactor-year – is considerably lower than the CDF for internal events (3.8 x 10^{-5} per reactor-year). Therefore, the approach used by VEPCo is considered to be acceptable.

The Surry Power Station Level 2 IPE model (VEPCo 1991) that was reviewed by NRC in 1993 has been modified to make the model consistent with that for VEPCo's North Anna Power Station. Both plants' models were converted to large early release frequency (LERF) models shortly after the IPE/IPEEE process was completed. The models remained unchanged until the beginning of the SAMA analysis, at which time a unified source-term category (STC) grouping was implemented that essentially used the approach presented in the North Anna IPE. The general containment event tree (CET) was also modified to reflect recent experimental results in severe accident analysis research (e.g., the resolution of the direct containment heating issue). The revision in the Level 2 PRA model, as a result of the aforementioned changes, resulted in a reduction in the overall contribution to early containment failure. This has a relatively small impact on the overall risk of severe accidents at Surry Power Station since the contribution to risk from early containment failure was already small. The staff concludes that the use of the Surry Power Station Level 2 model provides a sufficiently detailed characterization of containment response to support a license renewal SAMA analysis.

The staff reviewed the process used by VEPCo to extend the containment performance (Level 2) portion of the PRA to an assessment of offsite consequences (essentially a Level 3 PRA). This included consideration of the source terms used to characterize fission product releases for each of 24 source-term categories and consideration of the major inputs and assumptions used in the offsite consequence analyses. VEPCo used the severe accident source terms presented in the Surry IPE as input to the NRC-developed MACCS2 code. For radionuclides not reported in the IPE, releases were set to zero. VEPCo's source terms were reviewed and found to be consistent with the source terms provided in other plants' submittals and are considered reasonable.

VEPCo used site-specific meteorological data processed from hourly measurements for one full year (1998) as input to the MACCS2 code. All data was collected at the Surry Power Station meteorology tower. Hourly meteorological data for two additional years (1996 and 1997) was also used for sensitivity comparison. The use of data from either 1996 or 1997 results in only a few percent change in the total benefit of the candidate SAMAs. Year-to-year weather variations are not significant in the SAMA analysis because (1) weather variations are diminished in the MACCS2 analyses due to its weather-sampling scheme, and (2) the same meteorological assumptions are used in estimating both the base-case consequences and the SAMA-case consequences.

The population distribution the applicant used as input to the MACCS2 analysis was initially prepared using the computer program SECPOP90 (NRC 1997a). The output from SECPOP90 is a file based on a reference database for the specified site. The SECPOP90-prepared population data was then modified and updated using the Surry Power Station UFSAR, Section 2.1.3, 50-mile population distribution for the year 2030 in place of the SECPOP90 1990 Census data. The methods and assumptions for estimating population are considered reasonable and acceptable for purposes of the SAMA evaluation.

VEPCo's emergency evacuation modeling was based on a single evacuation zone extending out 16 km (10 mi) from the plant. VEPCo assumed that the people within the evacuation zone would move at an average evacuation speed of 1.8 m/s (4 mph) with a 7200-second delay between the alarm and start of evacuation. The applicant's base-case analysis assumed 100 percent of the population within the emergency planning zone would participate in the evacuation. In contrast, in NUREG-1150 (NRC 1990a) the staff assumed evacuation of 99.5 percent of the population. VEPCo performed a sensitivity analysis in which only 95 percent of the population evacuates. The result was only about a 1-percent change in the total benefit of the candidate SAMAs. Additional sensitivity analyses were also performed in which MACCS2 parameters relating to the time and duration of release and evacuation delay times were increased and decreased by 50 percent. The result was about a 10-percent change in the total benefit of the candidate SAMAs. This change is small and would not alter the outcome of the SAMA analysis. Accordingly, the evacuation assumptions and analysis are deemed reasonable and acceptable for purposes of the SAMA evaluation.

Much of the site-specific economic data were provided by SECPOP90 (NRC 1997a) and used in the MACCS2 analyses. SECPOP90 contains a database extracted from U.S. Census Bureau CD-ROMs (1990 census data), the 1992 Census of Agriculture CD-ROM Series 1B, the 1994 U.S. Census County and City Data Book CD-ROM, the 1993 and 1994 Statistical Abstract of the United States, and other sources. These regional economic values were updated to 1999 using cost-of-living and other data from the U.S. Census Bureau and the Department of Agriculture. VEPCo performed a sensitivity analysis in which the farmland and non-farmland decontamination costs were increased by 25 percent. The result was about a 6 percent or less increase in the total benefit of the candidate SAMAs.

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

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 VEPCo are discussed in this section.

5.2.3.1 Process for Identifying Potential Design Improvements

VEPCo's process for identifying potential plant improvements consisted of the following elements:

- a review 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,

- a review of other NRC and industry reports discussing potential plant improvements, e.g., NUREG-1560 (NRC 1997c), and NUREG/CR-5575 (NRC 1990b)

- a review of plant-specific improvements identified in the Surry IPE and IPEEE

- a review of the top 100 cutsets of the updated Surry PRA, and survey of Surry PRA staff for additional insights.

VEPCo's initial list of 160 candidate improvements was extracted from the process and is reported in Table G.2-1 in Appendix G of the ER (VEPCo 2001a).

VEPCo performed a qualitative screening on the initial list of 160 SAMAs using the following criteria:

- The SAMA is not applicable to Surry Power Station either because (1) the enhancement is only for boiling water reactors, the Westinghouse AP600 design, or ice condenser containments, or (2) it is a plant-specific enhancement that does not apply at Surry Power Station, or

- The SAMA has already been implemented at Surry Power Station (or the Surry Power Station design meets the intent of the SAMA), or

- The SAMA is related to a reactor coolant pump (RCP) seal vulnerability at many PWRs, stemming from charging pump dependency on component cooling water (CCW). The Surry plants do not have this vulnerability because the charging pumps do not rely on

CCW. However, other RCP seal LOCA improvements are considered, such as installing improved RCP seals.

Based on the qualitative screening, 107 SAMAs were eliminated. Of these 107 SAMAs, 38 were eliminated because they had already been implemented at Surry Power Station (or the design met the intent of the SAMA). The 53 remaining SAMAs are listed in Table G.2-2 of Appendix G of the ER (VEPCo 2001a), and were subjected to a final screening and evaluation process. The final screening process involved identifying and eliminating those SAMAs whose cost exceeded their benefit by at least a factor of two. All of the 53 remaining SAMAs were eliminated in this final screening.

5.2.3.2 Staff Evaluation

The preliminary review of VEPCo's SAMA identification process raised several questions regarding the set of SAMAs identified. The staff requested clarification regarding the portion of risk represented by the top 100 cutsets, and whether an importance analysis was used to confirm the adequacy of the SAMA identification process, since a review of the importance ranking of basic events in the PRA has the potential to identify SAMAs that may not be apparent from a review of the top cutsets.

VEPCo chose to review the top 100 cutsets for identification of potential SAMAs because they contain the dominant contributors to risk. The applicant states that the top 100 cutsets examined account for the majority (about 60 percent) of the CDF for internal events and contain all of the ISLOCA and much of the SGTR contribution to offsite consequences. The cutsets appearing below the 100th cutset have an individual frequency of 4.8×10^{-8} per reactor-year or less, and a collective frequency of approximately 1.5×10^{-5} per reactor-year. VEPCo also noted that since none of the SAMAs identified from the top 100 cutsets were found to be cost-beneficial, it is not likely that SAMAs from the cutsets below the top 100 would be either.

VEPCo indicated that an importance analysis was not used in the initial SAMA identification process. However, an importance analysis was performed as part of the model update. The importance list contained 131 basic events with a risk reduction worth (RRW) above 1.005. VEPCo performed a limited review of the importance list and verified that the risk-significant basic events were contained in the top 100 cutsets.

The staff notes that SAMAs with the greatest risk reduction potential should be revealed through the cutset screening because the top cutsets include the majority of the CDF and the risk-significant sequences, and all elements of their contribution are examined. Further, since the individual frequency of cutsets below the cutoff is 4.8×10^{-8} per reactor-year or less, and the collective frequency of cutsets below the cutoff is about 1.5×10^{-5} per reactor-year, it is unlikely that consideration of additional cutsets or further importance analyses would identify additional SAMAs that offer similar or greater risk reduction potential than those identified through cutset screening. The staff concludes that the process used to identify candidate SAMAs is sufficient to identify potential plant improvements that can significantly reduce risk.

VEPCo's efforts to identify potential SAMAs focused primarily on areas associated with internal initiating events. This is reasonable since external events only contribute a small amount to the total CDF and the containment response to external events was found to be similar to that from internal events in the IPE. The list of 53 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 Surry Power Station. The potential SAMA candidates included a balance of hardware, procedure, and training enhancements, as in the following examples:

- for loss of offsite power sequences, SAMAs included providing a hardwired connection to alternate offsite power (SAMA 77), and a lower-cost alternative of developing procedures to repair or change out failed 4-kV breakers (SAMA 69),

- for sequences with loss of heating, ventilation, and air conditioning, SAMAs included providing a non-safety-related, redundant train of switchgear ventilation (SAMA 25), and a lower-cost alternative of developing procedures for opening doors and using fans to limit temperature increases (SAMA 26), the latter of which is already implemented at Surry Power Station, and

- for sequences involving loss of support systems, SAMAs included adding a third component cooling water pump (SAMA 15), and a lower-cost alternative of enhancing training and procedures for loss of component cooling water or service water (SAMA 21).

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.

The staff concludes that VEPCo used a systematic and comprehensive process for identifying potential plant improvements for Surry Power Station. While explicit treatment of external events in the SAMA identification process was limited, VEPCo doubled the estimated benefit for internal events to account for any unmodelled risk reduction that could be attributed to external events. Therefore, the staff concludes that this limited treatment of external events is acceptable.

5.2.4 Risk Reduction Potential of Design Improvements

VEPCo evaluated each of the 53 SAMAs remaining after the initial screening using a bounding technique. Thirty-three bounding analysis cases were developed to accomplish this effort. Table 5-5 lists the remaining SAMAs, the bounding analyses performed to estimate the risk reduction for each SAMA, the estimated risk reduction in terms of percent reduction in CDF and person-sievert (person-rem) dose, and the estimated total benefit (present value) of the averted risk. As discussed previously, VEPCo doubled the estimated benefit for internal events to account for any unmodelled risk reduction that could also occur in external events. The total benefit values reported in Table 5-5 incorporate this doubling. The determination of the benefits for the various SAMAs is discussed in Section 5.2.6.

The staff has reviewed VEPCo's bases for calculating the risk reduction for the various plant improvements and concludes that the rationale and assumptions for estimating risk reduction are reasonable and generally conservative (i.e., the estimated risk reduction is higher than what would actually be realized). Accordingly, the staff based its estimates of averted risk for the various SAMAs on VEPCo's risk-reduction estimates. The estimated risk reduction for several of the SAMAs was negligible or zero, and in one case was slightly negative. In these instances, the SAMA either affects sequences or phenomena that do not contribute to risk at Surry Power Station or represents an ineffective plant improvement. As such, a minimal impact on risk is not unreasonable in those cases.

5.2.5 Cost Impacts of Candidate Design Improvements

VEPCo estimated the costs of implementing each SAMA through the application of engineering judgment, estimates from other applicants' submittals, and site-specific cost estimates. The SAMA cost analyses were prepared by VEPCo staff experienced in estimating the cost of performing work at a nuclear plant. Cost estimates were made as order-of-magnitude approximations. The depth of analysis performed varied depending on the magnitude of the expected benefit. For most of the SAMAs considered, because the cost estimates were sufficiently greater than the benefits calculated, no detailed evaluation was required. In these cases, the applicant indicated that the implementation costs would exceed twice the benefit. Detailed cost estimating was only applied in those situations in which the benefit was significant and application of judgement would be questioned. Detailed cost estimates were developed for the eight SAMAs listed in Table 5-6.

VEPCo assumed the minimum cost of generating a new procedure, including its implementation, to be $30,000. If the SAMA involved a hardware modification, it was assumed that the cost would be at least $100,000.

Table 5-5. SAMA Cost/Benefit Screening Analysis

Analysis Case and Applicable SAMAs	Analysis Assumption	Percent Reduction		Total Benefit ($)
		CDF	Dose	
IMPROVEMENTS RELATED TO EX-VESSEL ACCIDENT MITIGATION/CONTAINMENT PHENOMENA				
Qualitative Assessment 39-Create a concrete crucible with heat-removal potential under the basemat to contain molten debris 40-Create a water-cooled rubble bed on the pedestal 47-Create a core melt source reduction system 55-Create another building, maintained at a vacuum to be connected to containment	Eliminate all offsite releases.	0.0	100.0	1.64 million
SCB[(a)] 42-Enhance fire-protection system and/or standby gas treatment system hardware and procedures 54-Provide a reactor vessel exterior cooling system	Set the frequencies for source-term categories 1 through 16, 19 and 20, to zero.	0.0	4.9	45,000
HYD 37-Create/enhance hydrogen igniters with independent power supply 38-Create a passive hydrogen ignition system 48-Provide containment inerting capability	Set the probability of late containment failure due to hydrogen burn to zero.	0.0	0.02	1,000
DEB 43-Create reactor cavity flooding system 44-Create other options for reactor cavity flooding 154-Enhance reactor coolant system depressurization ability	Modify the CET failure probabilities for debris cooling.	0.0	0.0	0
No analysis case 46-Provide core-debris control system	This failure mode was zero in the Surry Level 2 analysis, so no further calculation was required.	0.0	0.0	0

Table 5-5. (contd)

Analysis Case and Applicable SAMAs	Analysis Assumption	Percent Reduction		Total Benefit ($)
		CDF	Dose	
CSP 30-Install containment spray throttle valves 32-Install a redundant containment spray system 33-Enhance the existing containment spray system 49-Use fire-water spray pump for containment spray 50-Install a passive containment spray system	Replace event tree functional equations related to containment and recirculation sprays with an event that has an unavailability of zero.	0.0	0.00	0
IMPROVEMENTS RELATED TO RCP SEAL LOCAS				
SWP 9-Provide an additional service water (SW) pump	Add logic for a new pump to fault trees CW1 and CW2.	2.0	0.3	34,000
SLO 10-Create independent RCP seal injection system with dedicated diesel 11-Create independent RCP seal injection system without dedicated diesel 14-Install improved RCP seals	Change event tree functional equations to eliminate the RCP seal LOCA contribution.	4.0	0.3	63,000
CCP[(a)] 15-Add a third component cooling water (CCW) pump 21-Enhance training and procedures for loss of CCW or SW	Add logic for a new pump to fault tree CC1.	0.02	0.3	5,000
IMPROVEMENTS RELATED TO SECONDARY/SUPPORT SYSTEMS				
CWV 23-Alter circulating water valve power-supply arrangement	Revise SWN0IC1 fault tree at four gates to provide a redundant 480-V power supply.	-0.5	-0.08	-4,000

Table 5-5. (contd)

Analysis Case and Applicable SAMAs	Analysis Assumption	Percent Reduction CDF	Percent Reduction Dose	Total Benefit ($)
BCC 81-Alter electric power dependency to BC and CC service water valves	Replace the motor-operated isolation-valve basic events with air-operated valve basic events, and remove power dependencies for each of the motor-operated valves.	0.7	0.5	17,000
IMPROVEMENTS IN AC/DC POWER RELIABILITY AND AVAILABILITY				
BCH 61-Use fuel cells instead of lead-acid batteries 64-Provide alternate battery-charging capability	Set battery failure basic events to zero.	5.4	0.8	88,000
OSP 77-Provide a connection to alternate offsite power source	Reduce loss of offsite power frequency by a factor of 5.	5.5	1.5	105,000
OPR 70-Emphasize steps in recovery of offsite power after SBO	Reduce offsite power recovery basic events by 25 percent.	1.8	0.5	33,000
4 kV 69-Develop procedures to repair or change out failed 4-kV breakers	Reduce basic events for all 4-kV breaker failures by a factor of 4.	1.9	2.0	62,000
IMPROVEMENTS RELATED TO HEATING, VENTILATION, AND AIR CONDITIONING (HVAC)				
HVC 25-Provide a non-safety-related, redundant train of switchgear ventilation	Change the initiating events frequency of the loss of HVAC to zero, and eliminate conditional ESGR failure by setting unavailability to zero.	13.9	5.0	278,000

Table 5-5. (contd)

Analysis Case and Applicable SAMAs	Analysis Assumption	Percent Reduction		Total Benefit ($)
		CDF	Dose	
HVA				
27-Add a switchgear room high temperature alarm	Reduce operator error for failure to recover HVAC by a factor of 10.	0.02	0.00	<1,000
Improvements Related to Decay Heat Removal Capability				
DHR				
34-Install a containment vent large enough to remove ATWS decay heat	Replace event-tree functional equations related to containment heat removal with an event that has an unavailability of zero.	4.9	1.6	90,000
35-Install a filtered containment vent to remove decay heat		4.9	5.5	135,000
36-Install an unfiltered containment vent to remove decay heat		4.9	1.6	90,000
FWS				
111-Install accumulators for turbine-driven auxiliary feedwater (TDAFW) pump flow control valves 115-Provide portable generators to be hooked in to the TDAFW after battery depletion	Modify event-tree functional equations related to auxiliary feedwater (AFW) in an SBO to use a basic event whose unavailability is zero.	0.1	0.04	4,000
FDW				
122-Create passive secondary side coolers	Modify event-tree functional equations related to main feedwater or AFW to use a basic event whose unavailability is zero.	12.8	17.2	490,000
SGP				
123-Automate air bottle swap for steam generator power-operated relief valves	Set basic event REC-INAIR-LOCAL to zero.	0.0	0.03	<1,000
SLB				
158-Install secondary side guard pipes up to the main steam isolation valves	Set the main steam line break initiating event frequencies to zero.	0.0	0.0	0

Table 5-5. (contd)

Analysis Case and Applicable SAMAs	Analysis Assumption	Percent Reduction CDF	Percent Reduction Dose	Total Benefit ($)
CND 124-Utilize bypass around the main steam trip valves to use condenser dump after safety injection	Remove house event XHOS-NO-CND-DUMP from five fault trees and gates.	2.2	0.01	33,000
IMPROVEMENTS FOR COPING WITH/IDENTIFYING CONTAINMENT BYPASS				
SGI 86-Install improved instrumentation and control circuits to detect and respond to SGTR	Set human error probabilities for isolating the faulted steam generator to zero.	2.8	27	256,000
SGR 88-Increase secondary side-pressure such that a SGTR would not cause the relief valves to lift 89-Replace steam generators with new design	Set the frequency of Plant Damage State 25 to zero.	5.7	60	576,000
ISS 101-Add remotely operated firewater line that could be used to scrub ISLOCA releases	Transfer the entire frequency of CET endstate 23 (unscrubbed ISLOCA) to CET endstate 22 (scrubbed ISLOCA).	0.0	5.3	40,000
ISL 103-Add a check valve downstream of the low head safety injection pumps on cold leg injection line to reduce ISLOCA frequency	Reduce ISLOCA frequency to zero.	4.3	30	253,000

Table 5-5. (contd)

Analysis Case and Applicable SAMAs	Analysis Assumption	Percent Reduction		Total Benefit ($)
		CDF	Dose	
IMPROVEMENTS RELATED TO ECCS				
LHI 125-Provide capability for diesel-driven, low-pressure vessel makeup	Use unavailability of zero for all "late" low head safety injection and recirculation events in the event trees, and credit the fire protection connection to low head safety injection and recirculation in the fault trees.	5.0	0.01	76,000
HPI 126/127-Provide an additional high-pressure injection pump with independent diesel	Add new pump logic to all charging and high head safety injection fault trees.	3.5	2.1	89,000
IMPROVEMENTS RELATED TO REDUCING INITIATING EVENT FREQUENCY				
ATW 145/146-Install motor generator (MG) set trip breakers in control room	Set the frequency of ATWS initiating events to zero.	0.01	0.0	<1,000
LLO 159-Add digital large break LOCA protection	Reduce the large LOCA initiating event frequency by 25 percent.	3.3	0.01	25,000
RTB 82-Relocate transfer buses to different room	Add the entire fire CDF (1.9×10^{-6}) to STC 19 (SBO).	5.0	0.7	41,000
MGB 83-Install fast-acting MG breaker	Reduce the transient initiating event frequency by 25 percent.	0.1	0.04	3,000

(a) Requires both plant hardware and procedure modifications.

Table 5-6. Surry Power Station SAMAs with Detailed Cost Estimates

SAMA No.	Description	Cost ($)
24	Provide a non-safety-related, redundant train of switchgear ventilation	15-25 million
64	Provide a portable, diesel-driven battery charger and associated disconnects	1.5-3 million
77	Provide a hard-wired connection to alternate offsite power source (Gravel Neck Combustion Turbines Station) and associated switchgear and disconnects	2-5 million
81	Replace service-water isolation valves with air-operated, fail close design	0.9-1.5 million
86	Provide improved instrumentation and control circuits to detect and respond to SGTR	1.5-3 million
101	Add remotely operated firewater line that could be used to scrub ISLOCA releases	125,000
103	Add check valve in each cold leg injection path to reduce ISLOCA frequency	0.75-1.25 million
125	Add a line to permit low-pressure vessel makeup from firewater header	350,000-600,000

The staff requested additional justification for several of the detailed cost estimates provided by VEPCo, including SAMAs 64, 77, and 86. VEPCo provided this information by e-mail, dated January 22, 2002 (NRC 2002). The staff reviewed the bases for the applicant's cost estimates. For certain improvements, the staff also compared the quantitative or qualitative cost estimates provided in Table 4-6 of the ER to estimates developed elsewhere for similar improvements, including estimates developed as part of other applicants' analyses of SAMAs for operating reactors and advanced light-water reactors. Based on this audit, the detailed cost estimates were judged to reflect valid bases and assumptions, with the exception of some labor estimates, which appear high. However, even if such estimates were lowered by an order of magnitude, the cost of the alternative would not be altered to the extent that it would become cost-beneficial. The qualitative cost estimates in Table 4-6 of the ER were found to be consistent with previous estimates and reasonable for the SAMAs under consideration. The NRC staff concludes that the cost estimates are sufficient and appropriate for use in the SAMA evaluations.

5.2.6 Cost-Benefit Comparison

The cost-benefit comparison as evaluated by VEPCo and the NRC staff evaluation of the cost-benefit analysis are described in the following sections.

5.2.6.1 VEPCo Evaluation

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

Net Value = ($APE + $AOC + $AOE + $AOSC) - COE

where $APE = present value of averted public exposure ($)
$AOC = present value of averted offsite property damage costs ($)
$AOE = present value of averted occupational exposure ($)
$AOSC = present value of averted onsite costs ($)
COE = cost of enhancement ($).

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

Averted Public Exposure (APE) Costs

The APE costs were calculated using the following formula:

APE = Annual reduction in public exposure (Δperson-rem/reactor-year)
x monetary equivalent of unit dose ($2000 per person-rem)
x present value conversion factor (10.76, based on a 20-year period with a 7-percent discount rate).

As stated in NUREG/BR-0184 (NRC 1997b), it is important to note that the monetary value of the public health risk after discounting does not represent the expected reduction in public health risk due to a single accident. Rather, it is the present value of a stream of potential losses extending over the remaining lifetime (in this case, the renewal period) of the facility. Thus, it reflects the expected annual loss due to a single accident, the possibility that such an accident could occur at any time over the renewal period, and the effect of discounting these potential future losses to present value. For the purposes of determining the maximum attainable benefit, VEPCo calculated an APE of $392,000.

Averted Offsite Property Damage Costs (AOC)

The AOCs were calculated using the following formula:

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

VEPCo cited an annual offsite economic risk of $39,585 based on the Level 3 risk analysis. This value appears to be higher than values for other sites and those presented in NUREG/BR-0184 (NRC 1997b). This higher value is primarily due to the relatively high frequency of SGTRs in the Surry PRA (2.33×10^{-6} per reactor-year, including both SGTR initiators and induced ruptures), which contribute 75 percent of the total offsite economic risk. For the purposes of determining the maximum attainable benefit, VEPCo calculated an AOC of $426,000.

Averted Occupational Exposure (AOE) Costs

The AOE costs were calculated using the following formula:

AOE = Annual CDF reduction
x occupational exposure per core damage event
x monetary equivalent of unit dose
x present value conversion factor.

VEPCo derived the values for averted occupational exposure based on information provided in Section 5.7.3 of NUREG/BR-0184 (NRC 1997b). Best estimate values provided for immediate occupational dose [33 person-Sv (3300 person-rem)] and long-term occupational dose [200 person-Sv (20,000 person-rem over a 10-year cleanup period)] were used. The present value of these doses was calculated using the equations provided in NUREG/BR-0184 in conjunction with a monetary equivalent of unit dose of $2000 per person-rem, a real discount rate of 7 percent, and a time period of 20 years to represent the license renewal period. For the purposes of determining the maximum attainable benefit, VEPCo calculated an AOE of $14,400.

Averted Onsite Costs (AOSC)

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

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

ACC = Annual CDF reduction
x present value of cleanup costs per core damage event
x present value conversion factor.

The total cost of cleanup and decontamination subsequent to a severe accident is estimated in NUREG/BR-0184 (NRC 1997b) as $\$1.5 \times 10^9$ (undiscounted). This value was converted to present costs over a 10-year cleanup period and integrated over the term of the proposed license extension.

Averted power replacement costs (RPC) are calculated using the following formula:

RPC = Annual CDF reduction
x present value of replacement power for a single event
x factor to account for remaining service years for which replacement power is required
x reactor power scaling factor.

Each of the units at Surry Power Station has a gross electrical output of 855.4 MWe, which is lower than the reference rating in NUREG/BR-0184 (NRC 1997b). Thus, a scaling factor (855.4/910) of 0.94 could be applied to the corresponding formulae. However, a scaling factor of 1.0 was conservatively used. For the purposes of determining the maximum attainable benefit, VEPCo calculated an AOSC (combination of ACC and RPC) of $738,000.

Using the above equations, VEPCo estimated the total present dollar value equivalent associated with completely eliminating internally initiated severe accidents at Surry Power Station is $1.57 million for each unit. This value was then doubled to account for additional risk reduction associated with also eliminating external events. This results in a maximum attainable benefit of $3.2 million for eliminating all severe accident risk.

VEPCo Results

The total benefit associated with each of the 53 SAMAs remaining after the initial screening is provided in column 5 of Table 5-5. These values were determined based on the above equations for the various averted costs together with the estimated annual reductions in CDF and person-Sv (person-rem) dose (columns 3 and 4 of Table 5-5). The estimated benefits were then doubled to account for additional risk reduction in external events. The values for total benefit reported in Table 5-5 include this doubling.

In determining the net value of each SAMA, VEPCo applied an additional factor of 2 multiplier to account for uncertainties in the cost-benefit methodology. Specifically, for each SAMA, they compared the total benefit[a] (doubled to account for external events) to the estimated cost of the enhancement and screened out the SAMA only if the cost of the enhancement was at least twice the benefit. All 53 SAMAs were eliminated because the estimated costs are expected to exceed the total benefit by at least a factor of 2. The end result was that no SAMA candidates were found to be cost-beneficial.

VEPCo performed sensitivity analyses to evaluate the impact of parameter choices on the analysis results. The sensitivity analyses included the calculation of candidate SAMA benefits using a 3-percent discount rate as recommended in NUREG/BR-0184 (NRC 1997b). The sensitivity cases resulted in less than a factor of 2 increase in the benefit calculation, and, therefore, all SAMAs were still screened out. Thus, the conclusion that none of the candidate SAMAs would be cost-beneficial remains unchanged.

5.2.6.2 Staff Evaluation

The cost-benefit analysis performed by VEPCo was based primarily on NUREG/BR-0184 (NRC 1997b) and was executed appropriately. The risk profile for Surry Power Station is observed to be dominated by containment bypass events (primarily SGTRs). With the exception of six costly modifications that are not properly applicable to an existing plant (e.g., redesign of the reactor cavity to accommodate a water-cooled rubble bed), the analysis found a maximum benefit of $278,000 with most changes resulting in a benefit of less than about $100,000.

The staff questioned the evaluation of several SAMAs in an RAI (NRC 2001). One SAMA in particular, SAMA 70, appeared to be cost-beneficial. This alternative involves a change to procedures for recovery of offsite power after a station blackout. According to Table 4-6 of the ER (VEPCo 2001a), a benefit of $33,000 was calculated. VEPCo estimated the minimum cost

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(a) The benefit can be due to a reduction in CDF and/or a reduction in person-Sv (person-rem) dose resulting from the alternative being implemented.

of a procedure change to be $30,000. Because this amount is less than the estimated benefit, the SAMA appears to be cost-beneficial. However, in their RAI response (NRC 2002), VEPCo indicated that the benefit was calculated assuming a 25 percent reduction in the offsite power nonrecovery terms, and that this is very optimistic because training for offsite power recovery is already given, and failure to recover offsite power is more likely attributed to actual failures of the grid and not to personnel error. Operator training has no impact on these types of failure. VEPCo indicated that the benefit in this area is actually quite small and would realistically be 1 or 2 percent as opposed to the 25 percent presented in the SAMA analysis. Based on this assessment, the total benefit would be at least an order of magnitude less than that provided in Table 4-6 of the ER. VEPCo further stated that it would not be practical to eliminate or trade off any of the current training material given the heavily loaded training schedule. Based on the rationale, the staff agrees that this SAMA does not appear to be warranted.

The staff believes that the costs of the 53 candidate SAMAs assessed would be considerably higher than the associated benefits. This conclusion is upheld despite a number of uncertainties and nonquantifiable factors in the calculations, noted as follows:

- External events were accounted for in the analysis by doubling the risk-benefits found considering internal events only. This was justified on the basis of the fact that the externally initiated CDF (1.3×10^{-5} per reactor-year) at Surry Power Station is less than the internally initiated CDF (3.8×10^{-5} per reactor-year), and the observation that there are no particular containment vulnerabilities in the external event risk profile.

- Uncertainty in the internal events CDF was not explicitly included in the calculations, which employed best-estimate values. The 95-percent confidence level for the internal events CDF is approximately three times the best estimate, and the results of the analysis show that no SAMA is found to be cost-beneficial within a factor of 3 or 4. Therefore, consideration of CDF uncertainty is not expected to alter the conclusions of the analysis.

- Risk reduction and cost estimates were generally found to be conservative. As such, uncertainty in the costs of any of the contemplated changes would not likely have the effect of making them cost-beneficial.

- A number of sensitivity risk-benefit calculations were performed with respect to the discount rate (as low as 3 percent) and various MACCS2 parameters, including evacuation time and completeness, meteorological data, source-term energy, and sheltering. The results of these calculations showed that none of the risk benefits were increased by more than a factor of 2. Because this is less than the margin between cost and benefit for most of the SAMAs considered, the staff concludes that uncertainties in these parameters would not alter the conclusions.

5.2.7 Conclusions

VEPCo compiled a list of 160 SAMA candidates based on the SAMA analyses submitted in support of licensing activities for other nuclear power plants, NRC and industry reports discussing potential plant improvements, and the plant-specific insights from the VEPCo IPE, IPEEE, and PRA model. Candidate SAMAs were identified by a thorough and systematic process that included examination of the Surry IPE and IPEEE, the top cutsets from the updated Surry PRA, and review of SAMA analyses for other operating nuclear power plants and other NRC and industry documentation. While few SAMAs were identified with a view towards external events, the IPEEE revealed no containment vulnerabilities particular to external events, and the staff judges that the process could be effectively carried out by considering primarily internal events. A qualitative screening removed SAMA candidates that did not apply to Surry Power Station for various reasons. A total of 107 SAMA candidates were either eliminated or combined with other potential improvements during the initial screening process, leaving only 53 SAMA candidates subject to the final screening process.

Using guidance in NUREG/BR-0184 (NRC 1997b), the updated Surry PRA model, and a Level 3 analysis developed specifically for SAMA evaluation, VEPCo estimated the total benefits for each of the 53 remaining SAMAs based on consideration of internal events, and then doubled the benefits for each SAMA to account for additional risk reduction in external events. In determining the net value of each SAMA, VEPCo applied an additional factor of 2 multiplier to account for uncertainties in the cost-benefit methodology. Specifically, for each SAMA, they compared the total benefit (which had been doubled to account for external events) to the estimated cost of the enhancement, and screened out the SAMA only if the cost of the enhancement was at least twice the benefit. All 53 SAMAs were eliminated because the estimated costs are expected to exceed the total benefit by at least a factor of 2. The end result was that no SAMA candidates were found to be cost-beneficial.

The staff reviewed the VEPCo analysis and concluded that the methods used and the implementation of those methods were sound. Based on its review, the staff concurs that none of the candidate SAMAs are cost beneficial. This conclusion is consistent with the low residual level of risk indicated in the Surry PRA and the fact that VEPCo has already implemented many plant improvements identified from the IPE and IPEEE process at the Surry Power Station.

5.3 References

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

10 CFR Part 54. Code of Federal Regulations, Title 10, *Energy,* Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”

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