Appendix G

NRC Staff Evaluation of Severe Accident Mitigation Alternatives for the R.E. Ginna Nuclear Power Plant in Support of License Renewal Application

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G.1 Introduction

Rochester Gas and Electric (RG&E) submitted an assessment of severe accident mitigation alternatives (SAMAs) for the R.E. Ginna (Ginna) Nuclear Power Plant as part of the Ginna Environmental Report (ER) (RG&E 2002). This assessment was based on the most recent Ginna probabilistic safety assessment (PSA) available at that time, a plant-specific offsite consequence analysis performed using the MELCOR Accident Consequence Code System 2 (MACCS2) code, and insights from the Ginna Individual Plant Examination for External Events (IPEEE) (RG&E 1997a, 1998a, 1998b, 1998c). In identifying and evaluating potential SAMAs, RG&E considered SAMA analyses performed for other operating plants that have submitted license renewal applications, as well as industry and U.S. Nuclear Regulatory Commission (NRC) documents that discuss potential plant improvements, such as NUREG-1560 (NRC 1997a) and NUREG-1742 (NRC 2002a). RG&E also identified SAMAs that were dominant contributors to core damage frequency (CDF) and large early release frequency (LERF) based on the plant-specific PSA. RG&E assessed the costs and benefits associated with each of the potential SAMAs and concluded that two of the candidate SAMAs evaluated are potentially cost beneficial for Ginna.

Based on a review of the SAMA assessment, the NRC issued a request for additional information (RAI) to RG&E by letter dated December 26, 2002 (NRC 2002a). Key questions concerned (1) dominant risk contributors at Ginna and the SAMAs that address these contributors, (2) the impact on dose consequences if all release categories were considered rather than just large early release categories, (3) the potential impact of uncertainties on the study results, and (4) detailed information on several specific candidate SAMAs. RG&E submitted additional information on January 31, 2003, and February 28, 2003, in response to the RAI (RG&E 2003a, 2003b). The February 28, 2003, response included a completely revised SAMA analysis (Section 4.14 and Appendix E of the ER) based on an updated version of the PSA. In these responses, RG&E provided tables containing the results of importance analyses, revised results based on the removal of scrubbing of fission product releases, and an assessment of the impacts of uncertainties. RG&E's responses addressed the staff's concerns and reaffirmed that only two SAMAs would be cost beneficial.

An assessment of SAMAs for Ginna is presented as follows.

G.2 Estimate of Risk for Ginna

RG&E's estimates of offsite risk at Ginna are summarized in Section G.2.1. The summary is followed by the staff's review of RG&E's risk estimates in Section G.2.2.

G.2.1 RG&E's Risk Estimates

Two distinct analyses are combined to form the basis for the risk estimates used in the SAMA analysis: (1) the Ginna Level 1 and 2 PSA model, which is an updated version of the Individual Plant Examination (IPE) (RG&E 1994, 1997b, 1997c), and (2) a supplemental analysis of offsite consequences and economic impacts (essentially a Level 3 PSA model) developed specifically for the SAMA analysis. The Level 1 and 2 PSA used as the basis for the SAMA analysis is the most recent PSA model of record, and is referred to as Version 4.2. The scope of the Ginna PSA does not include full consideration of seismic events. However, the dominant fire and internal flooding sequences are included in the PSA.

The baseline CDF for the purpose of the SAMA evaluation is approximately 4×10^{-5} per year. The CDF is based on the risk assessment for internally initiated events at power and at shutdown, and the dominant external events, specifically, fire and internal flooding at power. RG&E did not include the contribution of risk from seismic events within the Ginna risk estimates. It is RG&E's position that due to the recent and extensive evaluations and modifications performed as part of IPEEE and Seismic Qualification Utility Group (SQUG) activities, seismic events have been adequately addressed and need not be explicitly treated in the SAMA analysis (additional discussion provided in Section G.2.2).

The breakdown of CDF by initiating event/accident type is provided in Table G-1. Internal events at power contribute about 33 percent of the total CDF and are composed of (1) steam generator tube ruptures (15 percent of the total), (2) loss of coolant accidents (LOCAs) less than 5 cm (2 in.) (6 percent of the total), (3) station blackout (SBO) (5 percent of the total), (4) LOCAs greater than 5 cm (2 in.) (2 percent of the total), and (5) interfacing system LOCAs and anticipated transient without scram (ATWS) (each about 1 percent of the total) (RG&E 2003b). Shutdown events represent about 17 percent of the total CDF (RG&E 2003b). External event initiators represent about 50 percent of the total CDF and are composed of fire initiators (28 percent of the total CDF) and floods (22 percent of the total CDF) (RG&E 2003b).

The Level 2 PSA model has also been updated since the IPE. As described in the RAI responses (RG&E 2003b), results from the previous detailed Level 2 analysis were converted to the simplified LERF methodology described in NUREG/CR-6595 (NRC 1999a). In the updated

Table G-1. R.E. Ginna Nuclear Power Plant Core Damage Frequency (Revision 4.2 of PSA)

Contributor	CDF (per year)	Percent of Total CDF
Internal Events – At Power		
Transients	1.0×10^{-6}	3
Station Blackout (SBO)	2.1×10^{-6}	5
Anticipated transient without scram (ATWS)	2.0×10^{-7}	1
Steam generator tube rupture (SGTR)	6.0×10^{-6}	15
Loss of coolant accidents (LOCAs) <2 inches	2.6×10^{-6}	6
LOCAs >2 inches	7.0×10^{-7}	2
Interfacing system LOCA (ISLOCA)	2.5×10^{-7}	1
Internal Events – Shutdown	6.8×10^{-6}	17
CDF from internal events	$\mathbf{2.0 \times 10^{-5}}$	**50**
External Events		
Fire	1.1×10^{-5}	28
Flood	8.8×10^{-6}	22
CDF from external events	$\mathbf{2.0 \times 10^{-5}}$	**50**
Total CDF	$\mathbf{4.0 \times 10^{-5}}$	**100**

analysis, the 25 source term categories (STCs) used in the IPE were rebinned into 11 release category bins, each of which was assigned a representative source term based on the original MAAP analyses performed for the IPE. The conditional probabilities and release characteristics associated with each release category were provided in response to an RAI (RG&E 2003b). An explanation of the binning process and a mapping of the STCs to release category bins was also provided (RG&E 2003c).

The offsite consequences and economic impact analyses use the MELCOR MACCS2 code, Version 1.12, to determine the offsite risk impacts on the surrounding environment and public. Inputs for this analysis include plant-specific and site-specific input values for core radionuclide inventory, source term and release characteristics, site meteorological data, projected population distribution (within a 80-km [50-mi] radius) for the year 2030, emergency response evacuation modeling, and economic data.

In the ER, RG&E estimated the dose to the population within 80 km (50 mi) of the Ginna site to be approximately 0.163 person-sievert (Sv) (16.300 person-rem) per year (RG&E 2003b). The breakdown of the total population dose by containment release mode is summarized in

Table G-2. Bypass events (steam generator tube rupture [SGTR] and interfacing system loss-of-coolant accident [ISLOCA]) and late containment failures dominate the population dose risk at Ginna.

Table G-2. Breakdown of Population Dose by Containment Release Mode

Containment Release Mode	Population Dose Person-Sv Per Year	Population Dose Person-Rem Per Year	Percent Contribution
SGTR[(a)]	0.063	6.3	39
ISLOCAs	0.044	4.4	27
Early containment failure	0.020	2.0	12
Late containment failure[(b)]	0.030	3.0	19
No containment failure	0.006	0.6	3
Total	**0.163**	**16.300**	**100**

(a) Includes thermally induced SGTR.
(b) Includes contribution from shutdown events.

G.2.2 Staff's Review of RG&E Risk Estimates

RG&E's determination of offsite risk at Ginna is based on the following three major elements of analysis:

- the Level 1 and 2 risk models that form the bases for the 1994 IPE and 1997 IPEEE submittals (RG&E 1994, 1997a, 1997b, 1997c, 1998a. 1998b, 1998c)

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

- the MACCS2 analyses performed to translate fission product release frequencies from the level 2 PSA model into offsite consequence measures.

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

The staff's review of the Ginna IPE is described in an NRC report dated September 16, 1997 (NRC 1997b). 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 RG&E's analyses 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. The staff's review primarily focused on the licensee's ability to examine Ginna for severe accident vulnerabilities and not specifically on the detailed findings or quantification estimates. Overall, the staff believed that the Ginna 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.

In the IPE, RG&E identified five vulnerabilities as follows:

1. Relays for steam generator low-level actuation of auxiliary feedwater (AFW). The relays for this signal must be energized to actuate the AFW; however, they are currently powered by a non-safety bus that is unavailable upon a loss of offsite power.

2. ISLOCA through penetration 111. A LOCA outside containment through penetration 111 fails all residual heat removal (RHR) due to the low elevation of the RHR pump pits.

3. Standby AFW system out-of-service activities. Currently, both trains of this system can be taken out of service for up to 7 days; however, it is credited for providing steam generator cooling water for certain LOCAs outside containment.

4. Charging pump suction. Upon loss of dc control power or instrument air, the charging pump suction line fails to open the volume control tank, which may be empty because its supply source will have been eliminated as a result of the loss of power or air.

5. Intermediate building ventilation. The preferred AFW pumps are located in the basement of the intermediate building, which is ventilated via either building exhaust fans or natural circulation from a fire door opening; however, only one train of the exhaust fans is powered by the emergency diesel generators.

In an RAI, the staff questioned the current status of these vulnerabilities and whether any unresolved vulnerabilities were included in the SAMA evaluation. In response to the RAI, RG&E stated that items 1 and 3 had been resolved through plant modifications. Items 2 and 4, although considered by RG&E to be adequately addressed based on further review under the IPE program, are covered by SAMAs 3, 4, and 5. RG&E indicated that item 5 was originally identified as a result of overly conservative assumptions in the PSA model, and based on a more realistic assessment, it was reduced to a no-action status (RG&E 2003a). The staff inquired further about the conservative assumptions contained in the model. During a telephone conversation, RG&E explained that there are two methods of accomplishing ventilation within the intermediate building: (1) natural circulation via Fire Door F36 and (2) forced ventilation by the intermediate building exhaust fans (NRC 2003). Because only one train of the exhaust fans are diesel generator-backed, the three AFW pumps rely on the passive cooling in an SBO event in which the diesel generator is inoperable. A reanalysis of the building's ventilation determined that no active cooling is required for AFW; therefore, this item is no longer an item of concern.

The IPE also identified an issue associated with the dc electrical configuration that could result in a common mode failure of the pressurizer power-operated relief valves (PORVs). This was corrected during a subsequent outage.

A comparison of internal events risk profiles between the IPE and the PSA used in the SAMA analysis indicates a decrease of approximately 3.7 x 10^{-5} per year in the total CDF (about a factor of two). The reduction is attributed to plant and modeling improvements that have been implemented at Ginna since the IPE was submitted. A summary listing of those changes that resulted in the greatest impact on the total CDF was provided in response to an RAI (RG&E 2003b), and include:

- Relocated the service water (SW) piping that ran through the two battery rooms. This change eliminates the potential loss of both battery rooms due to failure to isolate SW line breaks in this area, which was the largest contributing CDF sequence.

- Modified procedures to avoid situations in which both trains of standby auxiliary feedwater (SAFW) could be taken out of service at the same time, thereby improving the ability to provide steam generator cooling in the event of a high-energy line break in the intermediate or turbine building.

- Revised the "Alternate Shutdown for Control Complex Fire" procedure to also apply to relay room floods. Previously, the procedure only addressed fire.

- Developed a new procedure to instruct plant personnel to manually close the Bus 18 breakers to prevent a SBO condition in the event of a worst-case fire.

- Updated generic data sources for initiating events, including the use of WCAP-15210 (WEC 1999) and NUREG/CR-5750 (NRC 1999b).

- Added plant-specific data for component failure rates, test and maintenance unavailabilities, and initiating event frequencies, and refined the Bayesian updating process.

- Increased frequencies for loss of offsite power to include all severe weather events, and included ISLOCAs whose frequencies previously fell below the threshold level for detailed analyses.

- Updated the human reliability analysis to provide detailed evaluations of more events in lieu of screening values.

- Removed conservatism for common cause failures that can induce initiators such as loss of service water, component cooling water, and instrument/service air.

- Added fires, internal floods, and shutdown risk models to the fault trees to enable their solution and risk ranking. Removed loss of spent fuel pool cooling and fuel-handling accidents and analyzed separately, because they do not lead to core damage.

The modeling changes from the IPE version to the current PSA are significant. Some contributors such as transients (previously a 25 percent contribution to internal events CDF) were significantly reduced. For example, the use of updated event frequencies significantly decreased the CDF from large LOCA, and plant changes such as a modification to the service water piping in battery rooms eliminated the largest contributor to CDF. Given the magnitude of the plant and model changes, the overall reduction in CDF appears to be reasonable.

The IPE CDF value for Ginna is comparable to most of the original IPE values estimated for other pressurized water reactors (PWRs) with a large dry containment. Figure 11.6 of NUREG-1560 shows that the IPE-based total internal events CDF for two-loop Westinghouse plants ranges from 5×10^{-5} to 1.2×10^{-4} per reactor-year (NRC 1997a). The internal events CDF based on the latest PSA (approximately 1.3×10^{-5} per year for events at power) is lower than the IPE values for other two-loop plants. However, it is recognized that other plants in addition to Ginna have reduced the values for CDF subsequent to the IPE submittals through modeling and hardware changes.

The staff considered the peer review performed for the Ginna PSA, and the potential impact of the peer review findings on the SAMA evaluation. In response to an RAI (RG&E 2003b), RG&E described the recent peer review of the Ginna PSA model. In preparation for a Westinghouse Owners Group peer review, an assessment of the Ginna PSA was performed by RG&E, the findings of which resulted in Revision 4.1. Revision 4.1 of the PSA model was reviewed by the Westinghouse Owners Group in May 2002. As a result of the peer review, RG&E updated the PSA to correct the most significant findings and observations. The updated model is referred to as Revision 4.2. According to RG&E, a few of the peer review comments were not incorporated into the current version of the PSA; however, those comments were evaluated and judged to have minimal impact of the plant CDF and no impact on the SAMA analysis (RG&E 2003c). Two high-level peer review items that were not addressed in the PSA but that could impact the SAMA analysis relate to the use of fission product scrubbing factors in the determination of source terms for bypass events. RG&E explicitly addressed these comments in the SAMA analysis by removing credit for scrubbing.

Ginna has two reactor coolant pumps (RCPs), each equipped with qualified high-temperature O-rings. The staff questioned RG&E regarding the model used to evaluate RCP seal LOCAs during loss-of-seal cooling events (NRC 2002a, 2003). The model used in Revision 4.2 is a composite based on (1) the original Westinghouse RCP Seal LOCA model developed in WCAP-10541 (WEC 1986), (2) the RCP Seal LOCA model employed by the NRC in NUREG-1150 (NRC 1990), (3) the Rhodes-based Brookhaven National Laboratory model, and (4) the most recent Westinghouse RCP Seal LOCA model described in WCAP-15603 (RG&E 2003c). RG&E noted that if the Rhodes model was used, the CDF would be higher by less than1 percent (RG&E 2003c). Based on RG&E's response, which supports use of the current model, the staff concludes that no new SAMA candidates would have evolved from application of the Rhodes model.

RG&E submitted an IPEEE in January 1997 (RG&E 1997c) in response to Supplement 4 of Generic Letter 88-20. This was followed by a submittal that included the fire analysis (RG&E 1998a). RG&E did not identify any vulnerabilities to severe accident risk in regard to the external events related to seismic, fire, or other external events. The Ginna hurricane, tornado, and high winds analyses show that the plant is adequately designed or procedures exist to cope with the effects of these natural events. Additionally, the Ginna IPEEE demonstrated that transportation and nearby facility accidents were not considered to be significant vulnerabilities at the plant. However, a number of areas were identified for improvement in both the seismic and fire areas as discussed below. In a letter dated December 21, 2000, the staff concluded that the submittal met the intent of Supplement 4 to Generic Letter 88-20, and that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities (NRC 2000). A strength noted in the IPEEE submittal was that Ginna is an Systematic Evaluation Program (SEP) plant and was subjected to a detailed review for SEP, much of which is applicable to IPEEE.

The Ginna IPEEE does not provide the means to determine the numerical estimates of the CDF contributions from seismic initiators. The seismic portion of the IPEEE consisted of a reduced-scope seismic evaluation using the methodology for Seismic Margins Assessment, described in Electric Power Research Institute NP-6041 (EPRI 1988). Since initial plant licensing, Ginna has undergone a number of programs addressing seismic design issues, one of which was the SEP. Under this and other programs, RG&E conducted extensive reevaluations of, and made upgrades to, structures, systems, and equipment at Ginna, using a 0.2g Regulatory Guide 1.60 spectrum as seismic input (NRC 1973). These efforts have extended seismic capacity of Ginna beyond the original seismic design basis.

During the IPEEE seismic analysis, RG&E identified five vulnerabilities:

- The house heating boiler, which is located near the service water pumps in the screenhouse, was not anchored. It could shift and damage the attached natural gas line.

- There are several locations where block wall failures could result in the release of combustibles: an oxygen line in the auxiliary building, a hydrogen line and valve station in the intermediate building, and hydrogen cylinders in the turbine building.

- There are two fire suppression systems that could be actuated by block wall failures: (1) the manual deluge system in the relay room and (2) both a manual deluge system and a pre-action sprinkler system on elevation 253 in the intermediate building.

- Block walls are used as fire barriers throughout the plant. The walls whose failure could impact the fire protection of safety-related equipment are those separating the service building from the intermediate building (column line 3), and those separating the turbine building from intermediate building (column line F).

- The two reactor coolant pump oil collecting tanks in the containment basement were not reviewed during the seismic walkdown because the containment was inaccessible.

These issues were later resolved as a part of the Ginna's IPEEE Fire Analysis by either design evaluations or design changes (RG&E 1998a).

Additionally, seismic issues were identified for 52 items of equipment (NRC 2002b). Fourteen of these were resolved as part of the closeout of unresolved safety issue (USI) A-46 (NRC 1987). In response to an RAI, RG&E indicated that the remaining 38 items have been resolved, and outlined the resolution of all 38 items, a majority of which were resolved by plant modification (RG&E 2003c). Typical modifications included installation of restraints, hangers, anchorages, and modifications of anchorages.

RG&E noted that one item still remains open: seismically induced flooding resulting from the failure of the Reactor Makeup Water Tank (RMWT) and the Monitor Tank (RG&E 2003a). In response to a staff inquiry regarding why this vulnerability was not addressed in the SAMA analysis, RG&E indicated that a modification to address this contributor is planned for implementation in 2005 (NRC 2003). Various design options are being evaluated, including installation of leak-tight, removable curb around the RHR sub-basement entrance to a level that would neither pose a flooding danger to the safety injection pumps nor allow the RMWT and Monitor Tank contents to enter the sub-basement (RG&E 2003c). This item has been entered into the Plant Change Request (PCR) system and is being tracked in the Commitment and Action Tracking System as item 10602 (RG&E 2003a).

The Ginna IPEEE fire assessment used a PSA approach to systematically and successively evaluate fire hazards and their associated risks. The analysis was performed in three phases. The first two phases, consisting of qualitative and quantitative screening steps, used methods that are consistent with the Fire-Induced Vulnerability Evaluation methodology, which was approved for use in NUREG-1407 for screening. The third phase was a detailed fire PSA, which was performed for fire areas and fire zones that were not screened. A quantification for fire events in the IPEEE indicated that the contribution to plant CDF from fire was about 3×10^{-5} per year.

Based on the analysis, RG&E concluded that there were no fire-induced vulnerabilities. However, several plant and procedural modifications were identified as a result of the analysis. The following modification was implemented and was credited in the analysis:

- Fuses will be installed on control circuits routed in the screen house associated with the functioning of 4160 VAC circuit breakers. The fuses will be designed to open if grounding occurs during a fire, thus permitting the protective function of the circuit breakers to remain intact.

Several other modifications were identified by the licensee at the time of the IPEEE submittal, specifically:

- an operating procedure enhancement for performing local recovery of the pressurizer heaters if control of the heaters is lost from the control room (the pressurizer heaters are one means of providing long-term reactor coolant system [RCS] circulation)

- insertion of a warning in the alternate shutdown procedure ER-FIRE-1 to indicate that, in the event of a spurious opening of motor-operated valve (MOV) 857B (which fails RHR shutdown cooling), this valve can be closed locally

- installation of additional sealed containers for transient combustibles storage in the auxiliary building basement

- spurious opening of MOVs 850A and 850B due to hot shorts can lead to draining of the refueling water storage tank (RWST) volume into the containment sump

- installation of a local pressure gauge to permit RWST level measurement in the event of fire-induced damage to level instrumentation.

In response to NRC questions on the IPEEE submittal, RG&E performed a detailed update of the fire risk study that included explicitly modeling operator actions and fire suppression systems. As a result, the above modifications were no longer risk significant and were dismissed. The results of the update were documented in RG&E's response to an RAI (RG&E 1999). The staff reviewed the response and concluded that the licensee's submittals met the intent of the IPEEE process.

Since the time of the IPEEE, further changes to the fire and internal flood analyses have been made. In response to an RAI, RG&E delineated the significant changes made to these analyses since the submission of the IPEEE. The changes include:

- The installed fire suppression systems have been explicitly modeled in the fault trees.

- Several human error events have been added, and a few were deleted to reflect more detailed modeling of specific fire events.

- The model has been revised to reflect a December 2000 plant modification to the service water piping in battery rooms, which eliminated the largest contributing CDF sequence.

- Several human error events for floods have been subjected to detailed human error analysis to yield more accurate values for their probabilities.

- Several flooding initiator frequencies have been revised as well as some new ones added to model certain zone-specific floods in greater detail.

Based on the current PSA, the contribution to the total CDF from fires is comparable to the CDF contribution from internal events (approximately 1×10^{-5} per year). As such, in an RAI the staff inquired whether specific SAMAs were considered that might reduce the risk due to fire (NRC 2002a). In response, RG&E stated that six of the eight candidate SAMAs (SAMA numbers 1, 2, 3, 4, 6, and 7) address elements of internal fire (RG&E 2003a).

Because RG&E included contributions from fire and floods in its base case evaluation, and due to the extensive efforts made during the IPEEE and SQUG processes to address seismic issues, the staff finds RG&E's consideration of external events to be acceptable.

Given that RG&E incorporated all relevant and significant comments from the Westinghouse Owners Group peer review and revised the SAMA analysis accordingly, that RG&E satisfactorily addressed staff questions regarding the PSA (RG&E 2003a, 2003b, 2003c), and that the CDF falls within the range of contemporary CDFs for Westinghouse plants with large dry containments, the staff concludes that the Level 1 and 2 PSA is of sufficient quality to support the SAMA evaluation.

The staff reviewed the process used by RG&E to extend the containment performance (Level 2) portion of the PSA to an assessment of offsite consequences (essentially a Level 3 PSA). This process included consideration of the source terms used to characterize fission product releases for the applicable containment release category and the major input assumptions used in the offsite consequence analyses. The MACCS2 code was used to estimate offsite consequences. Plant-specific input to the code includes the Ginna reactor core radionuclide inventory, emergency evacuation modeling, release category source terms, site-specific meteorological data, and projected population distribution within a 80-km (50-mi) radius for the year 2030. This information is provided in Appendix E of the Ginna ER (RG&E 2002).

RG&E used source term release fractions for 11 different release classes defined for Ginna. Tables 3 and 4 of the RAI responses provide a breakout of the source terms by release category (RG&E 2003b). The frequencies of the various release classes are based on an updated version of the IPE, developed consistent with the methodology described in NUREG/CR-6595. In the updated analysis, the 25 STCs used in the IPE were rebinned into 11 release category bins, each of which was assigned a representative source term based on the original MAAP analyses performed for the IPE. The binning and assignment of source terms appears to have been performed in a consistent manner; that is, the release category bins generally contain STCs with similar release characteristics and timing and are assigned a source term consistent with these characteristics. A sensitivity study was performed for a 10 percent increase in the quantity of fission products released. (The core inventory was increased by 10 percent while maintaining the release fractions.) This resulted in a 7 percent increase in the population dose. RG&E used the 10 percent larger source term as input into MACCS2 for the base case. The staff concludes that the assignment of source terms is acceptable for use in the SAMA analysis.

The applicant used site-specific meteorological data processed from hourly measurements as input to the MACCS2 code. Annual data from 1992 through 1994 were input into the MACCS2 code for the base case. The results showed that the total dose and cost results for the most severe release category (ISLOCA) are within 12 percent of the average. The data from 1992 yielded results above the average for all release cases and, therefore, was selected and used as the input. Where data blocks were missing in the source files, supplementary information was derived from meteorological data obtained from the National Oceanic and Atmospheric Administration from the Greater Rochester International Airport, approximately 24 km (15 mi) west of Ginna. The staff notes that previous SAMA analyses results have shown little sensitivity to year-to-year differences in meteorological data and considers use of the 1992 data in the base case to be reasonable.

The population distribution the applicant used as input to the MACCS2 analysis was estimated for the year 2030, based on the NRC geographic information system for 1990 (NRC 1997c), and the population growth rates were based on the 2000 county-level census data. A sensitivity study was performed by increasing the projected population for 2030 by 10 percent. This resulted in a greater than 20 percent increase for both offsite dose and economic costs. Due to this significant increase, RG&E used the 2030 population plus 10 percent in the base case analysis. The staff considers the methods and assumptions for estimating population reasonable and acceptable for purposes of the SAMA evaluation.

The emergency evacuation model was modeled as a single evacuation zone extending 16 km (10 mi) from the plant. It was assumed that 95 percent of the population would move at an average speed of approximately 1.8 meters per second (6.0 ft per second) with a delayed start time of 2 hrs (7200 s). This assumption is conservative relative to the NUREG-1150 study (NRC 1990), which assumed evacuation of 99.5 percent of the population within the emergency planning zone. The evacuation assumptions and analysis are deemed reasonable and acceptable for the purposes of the SAMA evaluation.

Much of the site-specific economic data were provided by specifying the data for each of the 13 counties surrounding the plant, to a distance of 50 miles. The SECPOP90 site input file was manually updated to the 2000 timeframe (NRC 1997c). The agricultural economic data were updated using available data from the 1997 Census of Agriculture supplemented by other data available through other federal agencies (USDA 1999). These included per value of farm and non-farm wealth, and fraction of farm wealth from improvements (e.g., buildings).

The staff concludes that the methodology used by RG&E to estimate the offsite consequences for Ginna, which includes the frequency-weighted contribution from all release categories, provides an acceptable basis from which to proceed with an assessment of risk reduction potential for candidate SAMAs. Accordingly, the staff based its assessment of offsite risk on the CDF and offsite doses reported by RG&E.

G.3 Potential Plant Improvements

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

G.3.1 Process for Identifying Potential Plant Improvements

In the Ginna ER (RG&E 2003b), only eight candidate SAMAs were identified. However, a much broader set of SAMAs was considered by RG&E to arrive at these eight SAMAs. RG&E

elaborated on its process for identifying potential SAMAs in response to RAIs (RG&E 2003a, 2003b, 2003c). The process consisted of the following elements:

- review of SAMA analyses performed for other operating plants that have submitted license renewal applications, particularly Fort Calhoun Station

- review of other NRC and industry documentation discussing potential plant improvements (e.g., NUREG-1560) (NRC 1997a)

- review of potential improvements identified in the plant-specific risk analyses (IPE, IPEEE, and subsequent PSA revisions)

- a review of the Fussel-Vesely (F-V) and risk achievement worth (RAW) importance measures, and the dominant CDF and LERF cut sets for Revision 4.2

- insights provided by RG&E plant staff.

Based on this process, 192 candidate SAMAs considered by previous applicants, plus several plant-specific SAMAs based on the Ginna PSA were identified (RG&E 2003c). RG&E performed a qualitative screening of the initial list of SAMAs and eliminated SAMAs from further consideration using the following criteria:

- The SAMA modifies features not applicable to Ginna.

- The SAMA would involve major plant design and/or structural changes that would clearly be well in excess (greater than two times) of the maximum attainable benefit (MAB).

- The SAMA would provide only minimal risk reduction based on review of F-V and RAW.

This qualitative screening process reduced the list to approximately 20 candidate SAMAs (RG&E 2003c). These SAMAs were further defined and then reviewed based on the following considerations:

- ability to implement the change at Ginna (i.e., are there any design challenges or physical limitations)

- the risk reduction that would realistically be achieved

- whether implementation of the change would increase vulnerabilities in other areas.

This culminated in eight plant-specific candidate SAMAs. These eight SAMAs were further evaluated, and two SAMAs were found to be potentially cost beneficial, as described below in Sections G.4 and G.6. RG&E considered the impact of uncertainties on the results of the SAMA analysis (RG&E 2003a). No additional SAMAs were judged to be cost beneficial (RG&E 2003b).

G.3.2 Review of RG&E's Process

The preliminary review of the Ginna ER raised concerns regarding the process used to identify potential SAMAs, and the completeness of the set of SAMAs considered. This was satisfactorily resolved though the additional information provided by the applicant, as described above. The staff also requested information regarding whether an importance analysis was used to confirm the adequacy of the SAMA identification process, and the portion of risk represented by the dominant risk contributors. In response to the RAI, RG&E provided a tabular listing of the contributors with the greatest potential for reducing risk as demonstrated by F-V and RAW assigned to the event. This approach inherently considers the top 95 percent of the CDF and LERF cut sets. RG&E also reviewed the dominant 50 CDF and LERF cut sets, which accounts for the top 45 percent of the CDF cut sets and 75 percent of the LERF cut sets (RG&E 2003b). Based on this, the staff concludes that RG&E's efforts to identify potential SAMAs included consideration of areas that presented the greatest potential for reducing risk. The list of eight 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 Ginna.

In the original ER submittal, the estimated MAB was $992,000 (RG&E 2002). During the screening process, SAMAs whose cost exceeded two times the MAB were removed from further consideration. The SAMA analysis was subsequently revised to address peer review comments, and that portion of the ER was resubmitted. As a result, the MAB increased to $1.93 million. RG&E concluded that the increase in MAB did not result in the identification of any additional SAMAs. The staff agrees with this conclusion because the initial screening removed SAMAs that are estimated to cost $2 million or more.

The staff questioned RG&E whether it considered some of the cost beneficial SAMAs identified at previous plants, specifically, the use of a portable generator to power steam generator level instrumentation, and improvements to the reactor protection system logic to reduce the likelihood of failure of two 125 VAC instrument buses causing the spurious opening of the PORVs (NRC 2003). In a telephone conversation, RG&E stated that such vulnerabilities did not exist at Ginna due to design differences, or that sufficient battery capacity existed. Ginna is a 4-hour coping plant but has 8-hour capacity batteries (NRC 2003). Based on a review of the response, the staff agrees with this conclusion.

The staff notes that the set of SAMAs submitted is not all inclusive, since 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 RG&E used a systematic and comprehensive process for identifying potential plant improvements for Ginna, and that the set of potential plant improvements identified by RG&E is reasonably comprehensive and, therefore, is acceptable. This search included reviewing insights from the IPE, IPEEE, and other plant-specific studies; reviewing plant improvements in previous SAMA analyses; and using the knowledge and experience of its PRA personnel.

G.4 Risk Reduction Potential of Plant Improvements

RG&E estimated the risk-reduction potential of the eight remaining SAMA candidates that were applicable to Ginna. RG&E used model requantification to determine the potential benefits. The CDF and LERF reductions were estimated using the current version of the Ginna PSA (Revision 4.2). The changes made to the PSA model to quantify the impact of each SAMA are detailed in Section E.3 of Appendix E to the Ginna ER (RG&E 2003b). Table G-3 provides a summary of the assumptions used to estimate the risk reduction, the risk reduction in terms of percent reduction in CDF and population dose, the total benefit (present value) of the averted risk, and the estimated implementation cost for each of the eight SAMAs. The determination of the benefits for the various SAMAs is discussed in Section G.6.

In response to an RAI, RG&E considered the uncertainties associated with the calculated CDF. This matter is discussed further in Section G.6.2.

The staff has reviewed the bases used by RG&E 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 risk reduction estimates provided by RG&E.

G.5 Cost Impacts of Candidate Plant Improvements

RG&E estimated the costs of implementing the eight candidate SAMAs through the application of engineering judgment and site-specific cost estimates. The cost estimates (presented in Section E.3 of Appendix E to the Ginna ER) conservatively did not include the cost of

Table G-3. SAMA Cost/Benefit Screening Analysis

SAMA	Assumptions	Percent Risk Reduction: CDF	Percent Risk Reduction: Population Dose	Total Benefit ($)	Estimated Cost ($)
1. Obtain a skid-mounted, 480-V diesel generator that could be directly connected to one train of the safeguards buses in the event of a failure of the two existing diesel generators.[(a)]	The addition of a skid-mounted, 480-V diesel generator with the same failure rate as the existing diesel generators and a 0.01 operator failure probability to start and align the diesel generator can supply the safeguards bus to reduce SBO and induced SBO sequences.	24.8	43.5	944,000	400,000
2. Obtain a third fire water source that is independent of the existing suction source for the motor- and diesel-driven fire pumps to be used in the event of a total loss of the screen house due to a fire or flood or loss of all service water suction due to environmental causes.	The addition of a diesel-driven pump of comparable size to the existing motor- and diesel-driven fire pumps can be connected to the existing fire system water piping and used for fire suppression or as a source of suction to the AFW pumps. The failure rate of the new pump is assumed to be the same as the existing diesel-driven fire pump. A failure rate of 0.1 is assumed for the operator action to connect the pump to the AFW system and 0.01 for the operator action to align the pump to supply the fire system during fire events.	1.8	3.3	70,000	200,000
3. Add a standby charging pump powered from a protected AC source and located in the intermediate or turbine building or SAFW pump building.	Conditions where charging pump A is out of service or directly failed, large floods that disable all three charging pumps and a charging pump room fire can be mitigated by an additional charging pump that autostarts on low flow or pressure. This pump is assumed to be powered from Bus 14.	11.2	2.5	107,000	1,100,000

Table G-3. (contd)

SAMA	Assumptions	Percent Risk Reduction		Total Benefit ($)	Estimated Cost ($)
		CDF	Population Dose		
4. Modify procedures to allow charging pump B or C to be manually aligned to Bus 14. This alignment could be used to mitigate fires requiring entry into procedure “Alternative Shutdown for Control Complex Fire” or fires disabling train B, where the A charging pump is out of service or fails to run.[(a)]	Manually aligning the B or C pump to Bus 14 can reduce all cut sets in which charging pump A is out of service or failed directly. A failure rate of 8.21×10^{-3} is used for aligning and starting the pump.	9.1	1.7	83,000	20,000
5. Add redundant check valves in the two RHR injection lines to the RCS to prevent a LOCA in the auxiliary building which could not be isolated.	The ISLOCA frequency is reduced reflecting the new configuration where failure of the additional check valve, the current check valve and the MOV, or both check valves and an inadvertent opening of the MOV, or a spurious safety injection signal would result in an ISLOCA. This was applied to the two lines through Penetration 111. It was also assumed that for this penetration LERF is a third of CDF because a third of the Penetration 111 piping that would be exposed to RCS pressure is inside containment.	0.2	7.7	45,000	1,000,000
6. Modify motor-driven AFW pump cooling system to be independent of service water (SW).	All cut sets that involve a loss of all AFW due to a failure of the SW suction source or a global failure of the screen house equipment due to fire or flooding will no longer lead to core damage due to the availability of the motor-driven pumps.	1.8	< 1	13,000	200,000

Table G-3. (contd)

SAMA	Assumptions	Percent Risk Reduction CDF	Percent Risk Reduction Population Dose	Total Benefit ($)	Estimated Cost ($)
7. Modify air-operated valve (AOV) 112C to fail close and AOV 112B to fail open on loss of instrument air. This change would allow the RWST to become the suction source for charging instead of the volume control tank (VCT).	All cut sets that contain the operator action to switch over the charging suction source from the VCT to the RWST can be reduced by setting this action to false (success).	2.0	< 1	14,000	50,000
8. Reconfigure the PORV so they transfer automatically from instrument air to N2 on low pressure and convert N2 supply line AOV to DC powered MOV.	The nitrogen system is available to support the power-operated relief valves with a failure probability of 4.76×10^{-3} (the failure rate of the components in the nitrogen system). Nitrogen support system failures were not included. This is conservative in that including these failures would increase the failure probability of the nitrogen system.	1.6	< 1	24,000	400,000

(a) SAMAs judged to be cost beneficial.

replacement power during extended outages required to implement the modifications, nor did they include recurring maintenance and surveillance costs or contingency costs associated with unforeseen implementation obstacles. Cost estimates typically included procedures, training, and documentation, in addition to any hardware.

The staff reviewed the bases for the applicant's cost estimates. For certain improvements, the staff also compared the cost estimates to estimates developed elsewhere for similar improvements, including estimates developed as part of other licensees' analyses of SAMAs for operating reactors and advanced light-water reactors. Six of the eight SAMAs were screened from further consideration on the basis that the expected implementation cost would be much greater than the estimated risk reduction benefit. This is reasonable for these six SAMAs given the relatively small estimated benefit (a maximum benefit of about $107,000 among the six SAMAs), and the sizeable costs typically associated with hardware modifications. It is noted that one SAMA (SAMA 7) involves a minimal hardware modification to two valve operators. However, the estimated benefit for this SAMA ($14,000) is small in comparison to the implementation costs ($50,000), and the actual costs are likely to be higher when all cost factors are included. The staff concludes that the cost estimates are sufficient and appropriate for use in the SAMA evaluation.

G.6 Cost/Benefit Comparison

RG&E's cost/benefit analysis and the staff's review are described in the following sections.

G.6.1 RG&E Evaluation

The methodology used by RG&E was based primarily on NRC's guidance for performing cost/benefit analysis (NRC 1997d). 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 costs ($)
AOSC = present value of averted onsite costs ($)
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. RG&E'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 1997d), 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 initial screening, RG&E calculated an APE of approximately $350,000 for the 20-year license renewal period, which assumes elimination of all severe accidents.

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.

For the purposes of initial screening, which assumes all severe accidents are eliminated, RG&E calculated an annual offsite economic risk of about $87,000 based on the Level 3 risk analysis. This results in a discounted value of approximately $932,000 for the 20-year license renewal period.

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.

RG&E derived the values for averted occupational exposure from information provided in Section 5.7.3 of the regulatory analysis handbook (NRC 1997d). Best estimate values provided for immediate occupational dose (3300 person-rem) and long-term occupational dose (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 the handbook 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 initial screening, which assumes all severe accidents are eliminated, RG&E calculated an AOE of approximately $15,000 for the 20-year license renewal period.

Averted Onsite Costs (AOSC)

Averted onsite costs (AOSC) 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. RG&E derived the values for AOSC based on information provided in Section 5.7.6 of the regulatory analysis handbook (NRC 1997d).

RG&E divided this cost element into two parts: (1) the onsite cleanup and decontamination Cost, also commonly referred to as averted cleanup and decontamination costs, and (2) the replacement power cost.

Averted cleanup and decontamination costs (ACC) were 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 the regulatory analysis handbook to be $\$1.5 \times 10^9$ (undiscounted). This value was converted to present costs over a 10-year cleanup period and integrated over the term of the proposed license extension.

Long-term replacement power costs (RPC) were 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

RG&E based its calculations on the value of 490 MWe, and scaled down from the 910 MWe reference plant in NUREG/BR-0184 (NRC 1997d). Therefore, RG&E applied a power scaling factor of 490 MWe/910 MWe to determine the replacement power costs. For the purposes of initial screening, which assumes all severe accidents are eliminated, RG&E calculated an RPC of approximately $169,000 for the 20-year license renewal period.

For the purposes of initial screening, which assumes all severe accidents are eliminated, RG&E calculated an AOSC of approximately $631,000 for the 20-year license renewal period.

Using the above equations, RG&E estimated the total present dollar value equivalent associated with completely eliminating severe accidents at Ginna to be about $1.93 million.

RG&E's Results

If the implementation costs were greater than the MAB, then the SAMA was screened from further consideration. A more refined look at the costs and benefits was performed for the remaining SAMAs. If the expected cost for those SAMAs exceeded the calculated benefit, the SAMA was considered not to be cost beneficial. The cost/benefit results for the individual analysis of the eight SAMA candidates are presented in Table G-3. As a result, two of the eight SAMAs were considered to be potentially cost beneficial:

- SAMA 1: Obtain a skid-mounted, 480-V diesel generator that could be directly connected to one train of the safeguards buses in the event of a failure of the two existing diesel generators.

- SAMA 4: Modify procedures to allow charging pump B or C to be manually aligned to Bus 14. This alignment could be used to mitigate fires requiring entry into procedure "Alternative Shutdown for Control Complex Fire" or fires disabling train B, where the A charging pump is out of service or fails to run.

RG&E performed sensitivity analyses to evaluate the impact of parameter choices on the analysis results (RG&E 2002, 2003a, 2003b). As discussed in Section 5.2.2.2, sensitivity cases that assumed a 10 percent increase in the projected population and a 10 percent increase in fission product releases were adopted in the baseline analysis. In addition, RG&E considered the impact on SAMA results if (1) a 3 percent discount rate (rather than 7 percent in the base case) as recommended in NUREG/BR-0184 (NRC 1997d) was used, and (2) if the 95th percentile values of the CDF were utilized in the cost/benefit analysis instead of the mean CDF. These analyses did not result in a positive net benefit for any additional SAMAs.

RG&E stated in the Ginna ER that the two potentially cost beneficial SAMAs identified above do not relate to adequately managing the effects of aging, and therefore, are not required to be implemented pursuant to 10 CFR Part 54 (RG&E 2003b). However, RG&E stated that it will consider implementation of these SAMAs through its current plant change process.

G.6.2 Review of RG&E's Cost/Benefit Evaluation

The cost/benefit analysis performed by RG&E was based primarily on NUREG/BR-0184 (NRC 1997d) and was executed consistent with this guidance.

In response to an RAI, RG&E considered the uncertainties associated with the calculated CDF (Table G-4). If the 95th percentile values of the CDF were used in the cost/benefit analysis instead of the mean CDF value used in the baseline analysis, the estimated benefits of the SAMAs would increase by about a factor of two. Increasing the benefit by this factor would have no impact on the conclusion of the SAMA evaluation; that is, even if the non-viable SAMAs (those qualitatively screened out) were increased by a factor of two, the resulting cost benefit would remain negative (RG&E 2003b).

Table G-4. Uncertainty in the Calculated Core Damage Frequency for R.E. Ginna Nuclear Power Plant

Percentile	CDF (per year)
5th	2.05×10^{-5}
50th	3.52×10^{-5}
mean	4.00×10^{-5}
95th	9.00×10^{-5}

In addition, RG&E performed sensitivity analyses that addressed assumptions made in other parts of the cost/benefit analysis, including variations in discount rate, weather, population, and source terms. These were either adopted in the base case (e.g., population and source terms) or are bounded by the CDF uncertainty assessment.

The staff concludes that, with the exception of the two cost beneficial SAMAs, the costs of the SAMAs would be higher than the associated benefits. This conclusion is supported by uncertainty assessment and sensitivity analysis and upheld despite a number of additional uncertainties and non-quantifiable factors in the calculations, summarized as follows:

- Uncertainty in the internal events CDF was not initially included in the calculations, which employed mean values to determine the benefits. The 95th percent confidence level for internal events CDF is approximately 2.25 times the best estimate CDF. Even upon considering the benefits at the 95th percentile value, no SAMAs were judged to be cost beneficial. Therefore, consideration of CDF uncertainty is not expected to alter the conclusions of the analysis.

- Seismic events were not included in the Ginna risk profile. However, seismic vulnerabilities were addressed during the IPEEE and SQUG evaluations. Fire and flood events have been included within the scope of the SAMA evaluation. An increase in the benefits by a factor of two had no impact on the results of the evaluation.

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

G.7 Conclusions

RG&E evaluated approximately 200 SAMA candidates using the SAMA analyses as submitted in support of licensing activities for other nuclear power plants, NRC and industry documents discussing potential plant improvements, and the plant-specific insights from the Ginna IPE, IPEEE, and current PSA model. A qualitative screening removed SAMA candidates that (1) were not applicable at Ginna due to design differences, (2) had already been implemented at Ginna, (3) were prohibitively expensive, or (4) did not provide a significant safety benefit. Upon conclusion of this screening, eight SAMA candidates were retained for further evaluation.

Using guidance in NUREG/BR-0184 (NRC 1997d), the current PSA model, and a Level 3 analysis developed specifically for SAMA evaluation, a maximum attainable benefit of about $1.93 million was calculated, representing the total present-dollar-value equivalent associated with completely eliminating severe accidents at Ginna. For the remaining eight SAMA candidates, a more detailed conceptual design and cost estimate were developed as shown in Table 5-5. The cost-benefit analyses showed that two of the eight SAMA candidates were potentially cost beneficial. Upon completion of a 3 percent discount rate sensitivity study, no additional SAMA candidates were determined to be cost beneficial. RG&E also considered the benefits at the 95th percentile CDF value, and found that no additional SAMAs were cost beneficial.

The staff reviewed the RG&E analysis and concluded that the methods used and the implementation of those methods were sound. The treatment of SAMA benefits and costs, the generally large negative net benefits, and the inherently small baseline risks support the general conclusion that the SAMA evaluations performed by RG&E are reasonable and sufficient for the license renewal submittal. The unavailability of a seismic PSA model precluded a quantitative evaluation of SAMAs specifically aimed at reducing risk of this initiator; however, significant improvements have been realized as a result of the IPEEE and SQUG processes at Ginna that would minimize the likelihood of identifying cost beneficial enhancements in this area. It is noted that one item still remains open: seismically induced flooding resulting from the failure of the RMWT and the Monitor Tank. However, RG&E is addressing this item through the PCR process and plans to implement a modification in 2005.

Although two SAMA candidates appear to be cost beneficial, they do not relate to adequately managing the effects of aging during the period of extended operation. Therefore, they need not be implemented as part of the license renewal pursuant to 10 CFR Part 54.

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