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VICE PRESIDENT
NUCLEAR OPERATIONS

February 28, 2003

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Mr. Robert G. Schaaf (Mail Stop O-12 D-3)
Office of Nuclear Reactor Regulation
Washington, D.C. 20555-0001

Subject: Response to December 26, 2002 "Request for Additional Information Regarding Severe Accident Mitigation Alternatives"
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Reference:
(1) Letter, Robert G. Schaaf (NRC) to Robert C. Mecredy (RG&E), December 26, 2002, "Request for Additional Information Regarding Severe Accident Mitigation Alternatives for the R. E. Ginna Nuclear Power Plant"
(2) Letter, Robert C. Mecredy (RG&E) to Robert G. Schaaf (NRC), "Response to December 26, 2002 "Request for Additional Information Regarding Severe Accident Mitigation Alternatives", dated January 31, 2003

Dear Mr. Schaaf:

This letter supplements our Reference (2) responses to the NRC's Reference (1) request.

In Attachment 1 to this letter, RG&E provides responses to Reference (1) requests for additional information (RAIs) 1b, 1c, 1d, 2a, 2c, 3, 5b, 6, and 7a-h. Also, in our Reference (2) response we stated that we would consider additional fire considerations (RAI 4c) and shutdown-related evolutions (RAI 8d) using Revision 4.2 of the PSA in our SAMA reanalyses. These evaluations were completed, and no additional fire or shutdown SAMAs were identified.

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Because of the extensive changes resulting from the revised SAMA analysis using PSA Rev. 4.2, we are providing in Attachment 2 a completely revised Ginna License Renewal Environmental Report, Section 4.14 and Appendix E. These sections will also be submitted subseqently in accordance with 10CFR54.21(b).

This submittal completes RG&E response to the NRC's Reference (1) request, and provides the NRC a clear and comprehensive assessment of the current status of our SAMA analysis.

Very truly yours,

Robert C. Mecredy

Attachments

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RESPONSE TO NRC'S REQUEST FOR ADDITIONAL INFORMATION REGARDING SEVERE ACCIDENT MITIGATION ALTERNATIVES (SAMAS) FOR THE R. E. GINNA NUCLEAR POWER PLANT (GINNA)

1. Although the process used by Rochester Gas and Electric Corporation (RG&E) to identify and screen potential SAMAs is described in general terms in the environmental report (ER), additional details are needed to understand how RG&E arrived at the final set of eight candidate SAMAs and to conclude that the full set of SAMAs evaluated by RG&E address the major risk contributors for Ginna. For example, RG&E states in the ER that it identified potential SAMAs from the Ginna Station Probabalistic Safety Assessment (PSA) and SAMA analyses submitted for other nuclear plant license renewals (Section 4.14.1), and that it focused on the dominant risk sequences identified by the model as well as the results of other risk-importance studies to further focus the evaluation (Section 4.14.3). However, few specifics are given. RG&E provides the Ginna risk profile and the importance analyses in Sections 1.2 and 1.3, but little information is provided on how the risk profile and importance analyses were used to identify or screen potential SAMAs. Additionally, the NRC staff notes that shutdown and fuel handling/spent fuel pool (SFP) cooling events are important contributors to core damage frequency (CDF) and large early release frequency (LERF), yet none of the SAMAs mentioned in the ER appear to address these contributors. In this regard, please provide the following additional information:

b. A description of how many sequences and cut sets were considered in the SAMA identification process and what percentage of the total CDF they represent;

c. A listing (more detailed than in Section E.1.3) of equipment failures and human actions that have the greatest potential for reducing risk at Ginna based on importance analyses and cut set screening;

d. A description of how many SAMAs were considered before arriving at the final set of eight candidates, and the process used to eliminate candidate SAMAs from further review or consideration;

Response to RAI 1b

The cutset review performed for the SAMA analysis is generally discussed in the response to RAI 1a provided in the January 31, 2003 submittal. The Ginna Station PSA Revision 4.2 was used to identify potential areas of risk benefit. This process involved use of the Fussell-Vesely (F-V) and Risk Achievement Worth (RAW) importance measures, as well as inspection of dominant core damage frequency (CDF) and large early release frequency (LERF) cutsets. In assessing potential SAMA modifications, two dimensional F-V /RAW plots were used to visually display the CDF and LERF importances associated with initiating events, human actions, specific systems and components. All items shown to have both high F-V and RAW values were reviewed. This approach inherently considers the top 95% of the CDF and LERF cutsets. In addition, the dominant fifty (50) CDF and LERF cutsets were investigated to establish the potential for economic risk-informed changes. This latter assessment included approximately the top 45% of the CDF cutsets and the top 75% of the LERF cutsets.

Response to RAI 1c

RG&E provided a partial response to this RAI in the previous submittal. This response supersedes that provided in the January 31, 2003 submittal. Table 1 presents the importance analysis results for PSA Revision 4.2. The PSA model changes are fully described in the response to RAI 2a.

TABLE 1
LEVEL 1 CDF IMPORTANCE ANALYSIS RESULTS

EIN	DESCRIPTION	PROB	F-V	RAW
Initiating Events				
FI0CR3-1	Fire in Zone CR-3 (Control Room Fire Scenario 1 or 2)	1.7E-03	1.58E-01	93.93
TX000RHR	Loss of RHR During Shutdown	4.06E-04	1.51E-01	374.32
FL000TB3	Steam Flooding Event in Turbine Building	6.32E-03	8.05E-02	13.65
LI0SGTRA	Steam Generator Tube Rupture in SG A	2.44E-03	7.50E-02	31.65
ACLOPSHTDN	Loss of Offsite Power During 24-hour Period when Shutdown	4.19E-04	7.49E-02	179.09
LI0SGTRB	Steam Generator Tube Rupture in SG B	2.44E-03	7.47E-02	31.51
Human Errors				
FSHFDCR-3-X	Fire brigade fail to manually suppress fire in Control Room	1.85E-02	1.90E-01	11.086
XXHFGSGTRE	Operators fail to respond to signals indicating SGTR (Early)	2.00E-03	8.55E-02	43.62
RHHFDREC04	Operators fail to recover RHR system before onset of boiling (4-12 hours)	1.00E-02	2.22E-02	3.2
RCHFPCDTR2	Operator fails to cooldown to RHR after SI fails – SGTR	9.66E-03	3.05E-02	4.13
SWHFDSTART	Operator fails to start SW pump	6.44E-03	1.86E-02	3.87
IFHFDAFWSW	Operators fail to locally align SW to TDAFW and SAFW suction following CR evacuation for floods or fires (ER-FIRE)	1.39E-02	1.82E-02	2.29
XXHFGNOAFW	Operators fail to diagnose a loss of all AFW	3.55E-04	2.19E-02	62.67
IFHFDTBISL	Failure to isolate large TB flood	1.88E-03	9.18E-03	5.87
CVHFDPMPST	Operators fail to manually load charging pump	6.44E-03	8.12E-03	2.25

TABLE 1 LEVEL 1 CDF IMPORTANCE ANALYSIS RESULTS				
EIN	**DESCRIPTION**	**PROB**	**F-V**	**RAW**
FSHFDTSCLT	Operators fail to use the TSC Battery Charger for long-term loss of AC Train B	7.29E-03	8.00E-03	2.09
AXHFPSAFWX	Operator fails to align and start SAFW pumps C & D	2.58E-03	7.60E-03	3.94
RRHFDRECRC-M	Operator fails to correctly shift the RHR system to recirculation and isolate CS-MBLOCA	5.25E-03	8.10E-03	2.53
RCHFPCOOLD	Operators fail to correctly shift the RHR after ARV sticks open or overfill occurs during SGTR	9.66E-03	1.98E-02	3.02
RCHFPCDDPR	Operators fail to cool down and depressurize to prevent SG overfill during SGTR	6.44E-03	1.61E-02	3.48
RHHFDREC24	Operators fail to recover RHR system before onset of boiling (12 – 24 hours)	5.00E-03	1.11E-02	3.21
MSHFPISOLR	Operators fail to isolate a ruptured steam generator	6.44E-03	8.40E-03	2.30
Test and Maintenance Activities				
CVTMCHPMPA	Test or maintenance renders charging pump A unavailable	7.04E-02	9.03E-02	2.192
DGTM00001B	Diesel Generator KDG01B unavailable due to testing or maintenance	1.74E-02	3.86E-02	3.177
DGTM00001A	Diesel Generator KDG01A unavailable due to testing or maintenance	1.55E-02	3.00E-02	2.903
AFTM0TDAFW	TDAFW pump out-of-service for maintenance	1.34E-02	2.57E-02	2.889
Systems				
DG			3.20E-01	1919.596
RCS			2.92E-01	8351851
Fire Protection			2.84E-01	25145.02
RHR			2.21E-01	647980.3
Offsite Power			1.82E-01	10344.21
CVCS			1.71E-01	25145.02
MS			1.40E-01	164.3292
CCW			1.09E-01	13090.92
AFW			9.71E-02	941888.8
SW			7.94E-02	8364.804
Motor-Operated Valves				
738A	MOV 738A fails to open	3.90E-03	1.30E-02	4.315

TABLE 1 LEVEL 1 CDF IMPORTANCE ANALYSIS RESULTS				
EIN	DESCRIPTION	PROB	F-V	RAW
738B	MOV 738B fails to open	3.90E-03	1.28E-02	4.257
Air-Operated Valves				
430	PORV PCV-430 fail to reseat after steam relief	5.00E-03	6.57E-03	2.306
PAF03	Failure of TDAFW pump train components	9.68E-03	1.72E-02	2.762
Major Electrical Component				
KDG01B/run	Diesel Generator B fails to run	4.22E-02	8.01E-02	2.815
KDG01A/run	Diesel Generator A fails to run	4.22E-02	7.47E-02	2.693
KDG01B/start	Diesel Generator B fails to start	1.01E-02	4.10E-02	5.013
KDG01A/start	Diesel Generator A fails to start	1.01E-02	3.96E-02	4.880
IBPDPCBCB	120 VAC Instrument Bus C (IBPDPCBCB) bus faults	2.09E-05	1.57E-02	745.85

Response to 1d

During the process of revising the SAMA analysis to incorporate Revision 4.2 of the PSA, RG&E performed further reviews to identify potential SAMAs consistent with the SAMA identification approach describe in response to RAI1a (provided in the January 31, 2003 submittal) and RAI 1b. RG&E identified one additional concept, adding spray shields to the CCW pumps to potentially mitigate risk associated with the operability of these pumps during a flooding transient. The evaluation of this concept included a cutset review and further design considerations. The results of the cutset review indicated that the failure rate was too low to generate a significant benefit, and further design considerations revealed that installation could not be done in a way not to impede routine maintenance. Therefore, RG&E determined that this concept would not be an effective modification and did not warrant detailed evaluation.

2. **In Section 1.1 of Appendix E to the ER, RG&E states that Revision 4.1 of the Ginna PSA was used for the SAMA analysis, and a brief description of the major changes to the preceding models is given. To gain a better understanding of how the PSA model has evolved and the impacts of the changes made to the model, please provide the following:**

   a. **A description of the major differences when comparing the Revision 4.1 PSA to the individual plant examination (IPE) model, which had an internal event CDF of 5.02E-05/y, including the plant and/or modeling changes that have resulted in the new CDF and LERF. According to Table E.1-2, the new internal-event CDF (not including shutdown and the "Fuel handling accident/Spent Fuel Pool") is 23 percent of the total CDF or 9.15E-06/y. Explain the principal reasons for this fivefold decrease in the full-power, internal events CDF, relative to the IPE results.**

c. A short description defining all the plant damage states (PDSs), and the accident sequences that dominate the PDSs (for the version of the model that was used for the SAMA analysis).

Response to RAI 2a:
The following information expands on that provided Section E.1.1 of the License Renewal Environmental Report and addresses the major revisions to the Ginna Station PSA Final Report.

Revision 1 - This was submitted in response to the NRC's original review of RG&E's Probablistic Safety Assessment (PSA) and is referred to by NRC in this question as the "IPE". This submittal estimated a total core damage frequency (CDF) of 5.02E-05/yr (excluding internal flooding). The CDF contribution was broken out as follows:

- Loss of Coolant Accidents (LOCAs) - 59%
- Steam Generator Tube Ruptures (SGTRs) - 16%
- Station Blackout - 12%
- Transients - 9%
- Interfacing System LOCAs - 2%
- Anticipated Transients Without Scram - 2%

The PSA as contained within Revision 1 addressed internal events (i.e., Level 1 issues) only. Revision 1 was submitted to the NRC on January 15, 1997 and was accepted by the NRC via letter dated September 16, 1997.

Revision 2 - This revision incorporated the revised Level 2 evaluation submitted in response to the NRC's review of the original RG&E PSA submitted in response to GL-88-20. This revision (primarily Section 10 of the Final Report) was submitted on August 30, 1997 and, although not reviewed in detail by the NRC, was accepted by the NRC via letter dated September 16, 1997.

Revision 3 - This revision was released in January 2000 and included a complete re-evaluation of internal events along with incorporating flooding and fire events, and a re-evaluation of Level 2. The major changes are summarized below:

1. Significant upgrades were made with respect to adding plant-specific data. Specifically, Revisions 0 and 1 primarily contained data from 1980-1988, while Revision 3 added selected data from 1994 (inception of Maintenance Rule data tracking) through 1998. This affected test and maintenance unavailabilities and initiating event frequencies (e.g., loss of offsite power)
2. Modeling conservatisms were removed, for example:
   a. The diesel generator (DG) fault tree model included detailed models of the supporting systems (e.g., fuel oil system). However, the generic and plant specific failure data for the DGs already included these support functions. Consequently, the overall DG failure rates were excessive (i.e., greater than twice the actual historical values). This was also true of the common cause assessment for the DGs (e.g., evaluating the fuel oil pumps separate from the overall DGs). The models were revised to remove these conservatisms while maintaining the necessary dependencies.
   b. Redundant (i.e., non-minimal) RCP seal LOCA cutsets were identified in the final results. This was primarily due to loss of CCW initiating events. This was corrected by adding "tags" to the model to track these events completely through quantification.

3. Generic data sources for initiating events were updated. This included the use of WCAP-15210, *Transient Initiating Event Operating History Database for U.S. Westinghouse NSSS Plants (1987-1997)*, and NUREG-CR-5750, *Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995*. In several instances, the use of these documents resulted in significantly lower initiating event frequencies (e.g., large LOCA frequency dropped from 1.80E-04/yr to 5.0E-06/yr).
4. The human reliability analysis was updated to provide detailed evaluations of more events. Previously, many events utilized screening values that were now updated based on reviews of timelines, operator cues, and procedural steps.
5. Fires were added to the fault tree models to enable their solution and risk ranking. The Fire Individual Plant Examination of External Events (IPEEE) analysis was submitted to the NRC by letter dated June 30, 1998. In response to NRC questions related to this submittal, RG&E revised the fire analysis (e.g., specifically modeled the fire water suppression systems). This revised analysis calculated a CDF due to fires of 3.3E-05/yr. This was submitted to the NRC on July 30, 1999 and accepted by the NRC on December 21, 2000. As a result of some of the changes described above, when the fire study was added to Revision 3, the fire CDF was determined to be 2.91E-05/yr.
6. Internal floods were added to the fault tree models to enable their solution and risk ranking. A revised Flood PSA was submitted to the NRC on March 1, 1999 in response to NRC questions on the original submittal. The Flood PSA calculated a CDF of 3.375E-05/yr. NRC accepted this analysis on December 21, 2000. As a result of some of the changes described above, when the flood study was added to Revision 3, the flood CDF was determined to be 3.16E-05/yr.
7. A shutdown risk model was added to the fault trees with a calculated CDF of 9.95E-06/yr.
8. The Level 2 analysis was replaced with a simplified LERF analysis consistent with NUREG/CR-6595. The original Level 2 analysis calculated a LERF of 1.11E-05/yr which was distributed as follows: SGTR (79%), ISLOCA (10%), LOCAs (11%). The simplified technique calculated a LERF of 3.567E-06/yr and included fires, floods, and internal events. Fires and floods dominated LERF due to their potential to directly affect equipment required to mitigate core damage and provide containment cooling and isolation. SGTRs continued to dominate internal event LERF. The original Level 2 analysis was relocated to Appendix I of the Final Report.

The result of changes 1 through 4 with respect to the Level 1 PSA contained in Revision 1 was a calculated internal event CDF of 2.55E-05, distributed as follows:

- Loss of Coolant Accidents (LOCAs) - 37%
- Steam Generator Tube Ruptures (SGTRs) - 27%
- Station Blackout - 14%
- Transients - 21%
- Interfacing System LOCAs - 1%
- Anticipated Transients Without Scram - < 1%

Revision 4 - This revision was issued in February 2002 and was primarily performed to reflect a December 2000 plant modification that eliminated the largest contributing CDF sequence (i.e., service water piping within battery rooms). Other changes included:

1. Data updates similar to Revision 3 were performed (i.e., adding plant-specific data from 1994 through 2000 for component failure rates and 1999-2000 for test and maintenance unavailabilities and initiating event frequencies). Included within the

data analysis was treatment of the plant-specific data between 1980-1988 and 1994-2000 as two separate pools which were only combined if it was statistically valid to do so. In this manner, many equipment failure rates dropped significantly once the older history was excluded (i.e., the new data more accurately reflected equipment reliability following issuance of the Maintenance Rule).

2. Hydrogen and other exothermic explosions were added to the model.
3. A parametric uncertainty analysis was performed.

The result of these changes was a calculated internal event CDF of 1.28E-05/yr, distributed as follows:

- Loss of Coolant Accidents (LOCAs) - 24%
- Steam Generator Tube Ruptures (SGTRs) - 30%
- Station Blackout - 11%
- Transients - 25%
- Interfacing System LOCAs - 10%
- Anticipated Transients Without Scram - < 1%

The calculated frequencies for the remaining events were as follows:

- Fires - 1.89E-05/yr
- Floods - 8.78E-06/yr (drop mainly as a result of plant modification described above)
- Shutdown - 7.88E-06/yr

Revision 4.1 - In preparation for the May 2002 Westinghouse Owner's Group (WOG) Peer Review, an assessment of the Ginna Station PSA was performed by an outside contractor. The findings from this assessment resulted in Revision 4.1 of the PSA that was completed in April 2002. The result of these changes was a calculated internal event CDF of 1.00E-05/yr, distributed as follows:

- Loss of Coolant Accidents (LOCAs) - 24%
- Steam Generator Tube Ruptures (SGTRs) - 31%
- Station Blackout - 11%
- Transients - 25%
- Interfacing System LOCAs - 8%
- Anticipated Transients Without Scram - 1%

The calculated frequencies for the remaining events were as follows:

- Fires - 1.28E-05/yr
- Floods - 7.28E-06/yr
- Shutdown - 8.45E-06/yr

Revision 4.2 - As a result of the WOG Peer Review in May 2002, the PSA was updated in November 2002 to correct the majority of the most significant Findings and Observations. The key model changes include:

1. The Bayesian updating process for plant-specific data, including initiators, has been refined and common-cause failures that can induce initiators such as loss of service

water, component cooling water, and instrument/service air have been re-evaluated to remove conservatism.

2. The methodology for incorporating the effect of dependency among post-initiator human actions during diagnostic and post-diagnostic phases has been strengthened, while median values from the screening process used in quantifying pre-initiator human errors have been replaced by mean values.
3. Inconsistencies within the event trees for transients (steamline break scenarios), steam generator tube rupture and anticipated transient without scram have been resolved.
4. Frequencies for losses of offsite power have been increased to include all severe weather events, even those that would not occur at Ginna Station (e.g., hurricanes, salt spray).
5. Frequencies of loss of spent fuel pool cooling and fuel handling accidents have been removed from the core damage frequency and analyzed separately. Similarly, frequencies for late containment releases, such as loss of spent fuel pool cooling and steam generator tube rupture with successful isolation of the failed generator, have been removed from the large early release frequency and analyzed separately.
6. Interfacing system loss-of-coolant accidents whose screening frequencies previously fell below the threshold level for detailed analyses have received detailed analyses and been included in the core damage and large early release frequencies.

The result of these changes was a calculated internal event CDF of 1.28E-05/yr, distributed as follows:

- Loss of Coolant Accidents (LOCAs) - 25%
- Steam Generator Tube Ruptures (SGTRs) - 50%
- Station Blackout - 18%
- Transients - 3%
- Interfacing System LOCAs - 2%
- Anticipated Transients Without Scram - 2%

The calculated frequencies for the remaining events were as follows:

- Fires - 1.14E-05/yr
- Floods - 8.78E-06/yr
- Shutdown - 6.81E-06/yr

A combined CDF of 3.98E-05 was used in the SAMA analysis cost benefit calculations consistent with the PSA Revision 4.2 results. As can be seen above, the majority of the reduction in risk since Revision 1 is due to improved data analysis and removal of model conservatisms. This was specifically reviewed by the WOG Peer Review Team and determined to be acceptable.

Response to RAI 2c:

The PSA information regarding plant damage states (PDS) is retained in Appendix I of Reference 1. The RG&E PSA described in Appendix I of Reference 1 identified 58 potential plant damage states. Of these states, ten PDSs were found to be dominant and represent over 90% of the total CDF. The methodology utilized for the SAMA evaluation maps the plant accident scenarios into eleven Release Category bins with unique fission product release characteristics. Of the eleven Release Category bins, one represents an intact containment; two represent late containment failures; one represents releases from an SGTR with an intact

secondary side; and the remaining states are parsed among the contributors to the large early release. Brief descriptions of the Release Category bins used in the SAMA analysis are provided in the response to RAI 3 along with a mapping to the PSA PDSs. Shutdown has been added to the RG&E PSA; therefore, a shutdown release class has also been defined (See Table 5 of RAI response 3).

Reference
1. R. E. Ginna Nuclear Power Plant Probabilistic Safety Assessment, Final Report, Rev. 4.0 February, 2002.

3. **In Section 2.1.2 of Appendix E to the ER, RG&E states that the source terms (STs) were obtained from the latest Level 2 Ginna Station PSA model analysis. Please provide more detailed information (e.g., a tabular list) on the release categories used in the SAMA analyses, including the definition, fractional releases, timing of releases, frequency, containment matrix (relationship between PDSs and release categories), and the associated conditional consequences. Confirm whether the STs are the same as in the IPE and, if not, explain how/why they are different.**

Response to RAI 3:
The source terms used in the SAMA analysis are the same as those contained in Appendix I to the RG&E PSA Final Report (see further discussion in response to RAI 2c) (Reference 1). This information was archival in nature and was generated to support the RG&E response to NRC Generic Letter (GL) 88-20. The modeling approach used for the SAMA evaluation is based on an extension to NUREG-6595 (Reference 2) to consider intact and late containment failure states. This methodology was used in the initial SAMA submittal and has been updated to address comments associated with the RG&E PSA peer review (see response to RAI 6) and used in the revision of the SAMA analysis (Attachment 2). A review of the current PSA results indicates that the dominant PDSs remain the same as those identified in the original PSA (Table I-4 of Reference 1)

The latest full Level 2 source term analysis performed for the Ginna Station was PSA Revision 2. The details of the source term releases used in the SAMA analysis are contained in Appendix I of Reference 1. In the analysis described in that appendix, the MAAP 3.0B computer code was used to establish radionuclide releases for 25 representative severe accident scenarios. For the SAMA analysis, the source term categories (STCs) described in Appendix I were reviewed and release characteristics were binned into 11 bounding categories based on the following:

a. Size of the containment release
b. Release energetics (e.g., high or low pressure containment failure)
c. Timing of the release
d. Release pathway (e.g., ISLOCA, loss of isolation or bypass through steam generator)
e. Mode of steam generator release (for steam generator sequences)

The resulting 11 bins and the inter-relationship of the PSA STCs to the current release class end-states and PSA PDSs are described in Table 2. MAAP-predicted fractional fission product releases for the release categories are presented in Table 3. The timing of releases is shown in Table 4, and a summary of offsite economic impact of the releases is provided in Table 5. Release class frequencies presented in Tables 4 and 5 are based on the revised SAMA analysis, which utilized the RG&E PSA Revision 4.2 (Reference 3).

Table 2
Summary of Source Term Release Bins for R E Ginna
SAMA Evaluations

No.	Release Category Bin	IPE Source Term Category (STC) End State	Description	Plant Damage State
1	RC_ISLOCA Inter-system LOCA (ISLOCA)	22	This release category represents a containment bypass scenario via an interfacing system LOCA outside of containment. A medium break LOCA with no injection or containment heat removal is the representative ISLOCA. AFW and accumulators are assumed to be available. Core damage occurs within 1 hour and vessel failure is predicted to occur in about an hour and a half. Source terms do not credit any scrubbing due to overlying water pools or settling in the Auxiliary Building.	PDS 55
2	RC_LOCI Core Damage with Loss of Containment Isolation (LOCI)	19	This category represents a vessel failure scenario with a containment isolation failure. It is conservatively assumed that no injection, fan coolers or containment sprays are available. The source term results for this case are represented by a large early containment failure case with no fan coolers available.	PDS 3
3	RC_SGTR WET Core damage following an unmitigated SGTR (SG contains secondary inventory)	24	This is an unisolated steam generator tube rupture (SGTR) with feedwater being applied to the affected SG. No safety injection is assumed and the affected SG ARV sticks open at 20 minutes. With no injection or AFW, the primary and the broken loop secondary inventory eventually boils away. As AFW is functional, it is assumed that AFW can be reestablished late in the sequence. AFW is restored to the broken loop when primary temperature reaches about 1800 °F.	PDS 57
4	RC_SGTR DRY Core damage following an unmitigated SGTR (SG dry)	25	This is an unisolated steam generator tube rupture (SGTR) without feedwater being applied to the affected SG. No safety injection is assumed and the affected SG ARV sticks open at 20 minutes. With no injection or AFW, the primary and the broken loop secondary inventory eventually boils away. AFW is not restored to the broken loop SG and it remains dry.	PDS 56
5	RC_SGTR ARV Cycle Core damage following an unmitigated SGTR (SG releases at high pressure, relief valves cycling)	23	This STC end state is an isolated SGTR without feedwater being applied to the affected SG. No safety injection is assumed, and the feedwater to the affected SG is terminated at 30 minutes. Majority of fission product release occurs due to affected SG ARV cycling during core damage. After vessel failure, the primary and secondary pressures equalize, thereby limiting the releases to containment leakage.	PDS 56

Table 2
Summary of Source Term Release Bins for R E Ginna
SAMA Evaluations

No.	Release Category Bin	IPE Source Term Category (STC) End State	Description	Plant Damage State
6	RC_LATE_GLOBAL Late Failure Global Core damage event followed by a late global containment failure	10	This STC end state describes a scenario with vessel failure and a late global containment failure assumed to occur with containment heat removal available, but with dry core-concrete interaction also occurring. The representative scenario is a station blackout with an induced hot leg rupture. This allows for all molten core debris to remain in the cavity following vessel failure and dry core-concrete interaction to occur. At 10 hours, the fan coolers are assumed to be functional and restored. This results in a reduction of containment steam concentration leading to a hydrogen burn at 11.7 hours. The hydrogen burn is assumed to fail the containment. The restoration of fan coolers also puts water in the cavity, cools the core debris, and terminates core-concrete attack.	PDS 45
7	RC_LATE_SMALL Late Failure Small Core damage event followed by a late small containment failure (leak)	13	This STC end state is a vessel failure scenario with a leak-before-break containment failure assumed to occur without containment heat removal available and with core-concrete interaction. A station blackout with an induced hot leg rupture and no power recovery is chosen as the representative sequence. All core debris enters the cavity following vessel failure, and core-concrete attack is assumed to occur quickly. At 16.7 hours the containment pressure reaches the failure pressure due to the mass and energy release into the containment with no containment heat removal. Gradual containment depressurization to about 40 psia occurs at around 48 hours into the accident. With no containment heat removal available, core-concrete attack continues to the end of the accident.	PDS 46
8	RC_TISGTR Core damage event followed by a thermally induced SGTR and dry SG release	25	This is a core damage event followed by an induced steam generator tube rupture (SGTR) without feedwater being applied to the affected SG. With no safety injection, the affected SG ARV sticks open at 20 minutes and with no injection and AFW to the broken loop SG, the primary and the broken loop secondary inventory eventually boils away. AFW is not restored to the broken loop SG and it remains dry. The majority of the fission product releases occur via the tube rupture and out the stuck open ARV of the broken SG.	PDS 56

Table 2
Summary of Source Term Release Bins for R E Ginna
SAMA Evaluations

No.	Release Category Bin	IPE Source Term Category (STC) End State	Description	Plant Damage State
9	RC_HPRCS Early containment failure due to rapid containment pressurization following a high pressure reactor vessel failure	16	This STC end state is a vessel failure scenario with an early global containment failure assumed to occur without containment heat removal available and with core-concrete interaction. The representative scenario is a station blackout with an induced hot leg rupture and no power recovery.	PDS 46
10	RC_LPRCS Early containment failure due to containment challenges associated with a low pressure reactor vessel failure	7	This end state describes a scenario with vessel failure followed by an early leak containment failure. There is no containment heat removal and no core-concrete interaction. The representative sequence was chosen as a medium LOCA without recirculation. Containment failure is assumed to occur two seconds after vessel failure.	PDS 10
11	RC_INTACT Intact Containment Core damage in the presence of an intact containment (containment heat removal available)	1	This STC end state describes a scenario with vessel failure, but without containment failure due to the long-term availability of containment heat removal. The representative sequence is a small break LOCA without the recirculation mode of injection available. Vessel failure occurs with core debris entering the cavity at the time of vessel breach. About 15% of the debris is assumed to be retained in the vessel. With fan coolers available and with RWST inventory in the containment, water is available to cover the core debris in the cavity. The debris is assumed to be in a coolable configuration with only about 0.015 ft of concrete attack calculated to occur before the debris mixture is quenched. Containment failure does not occur, and fission product releases occur due to leakage.	PDS 30

Table 3
MAAP Predicted Fractional Fission Product Releases[a]

Sequence	STC End State	Plume No.	NG	I	Cs	Te	Ba	Sr	Mo	La	Ce
1. RC_ISLOCA	22	1	0 992	0 669	1	0.137[b]	0 00506	0 00509	0.00122	0 019	0 0242
2 RC_LOCI	19	1	0.82	0.0291	0.0583	0.0369	0.00155	0.00311	0.0000507	0.00318	0.0067
3. RC_SGTR WET	24	1	0 99	0.0406	0.0805	-	-	-	-	-	-
		2	-	-	-	0.00399	0 0000835	0 0000115	0.000049	0 000677	0 000677
4. RC_SGTR DRY	25	1	0.992	0.274	0.539	-	-	-	-	-	-
		2	-	-	-	0.0458	0.0005	0 0000728	0.000082	0.00101	0.00101
5. RC_SGTR ARV Cycle	23	1	0 327	0 00495	0.0098	0.000204	0.00001	0.00001	0.00001	0 00001	0 00001
6. RC_LATE_ GLOBAL	10	1	0.885	0.0186	0.0402	0.0298[b]	0.0000504	0.0000635	0.000308	0.000237	0.000378
7. RC_LATE_ SMALL	13	1	0.904	0 000937	0 003057	0.03566	0.0000228	0 0000341	0.00001	0.000197	0 000265
8.RC_TISGTR	25	1	0.992	0.274	0.539	-	-	-	-	-	-
		2	-	-	-	0.0458	0.0005	0.0000728	0.000082	0.00101	0.00101
9. RC_HPRCS	16	1	0.999	0.024	0.0659	0 0835	0.0000784	0.0000689	0 00001	0.00039	0 000528
10. RC_LPRCS	7	1	0 956	0.000382	0.000818	0 0000912	0 00001	0 00001	0.00001	0.0000891	0.0000891
11. RC_INTACT[c]	1	1	0 00062	0 000102	0.000141	0.0000039	0.0000039	0.0000039	0.0000039	0.0000039	0 0000039
		2	0.00099	0.0000164	0 00002	0.0000062	0.0000062	0.0000062	0 0000062	0.0000062	0.0000062

a. Data contained in this table is compiled from Appendix I of Revision 4 of the Ginna PSA (Reference 1)

b. Values for Sb used when Sb releases were greater than Te releases. This is consistent with MACCS grouping.

c. Cs and I contributions for the intact releases have been increased an order of magnitude for conservatism and to facilitate modeling.

Table 4
Summary of Source Term Release Category Bins for R. E. Ginna SAMA Evaluations

No	Release Category Bin	IPE Source Term Category (STC)	Frequency of Release[a] (per year)	Timing of initial release (hrs)
1	Inter-system LOCA (ISLOCA) – RC_ISLOCA	22	2.5E-07	4
2	Core Damage with Loss of Containment Isolation (LOCI) – RC_LOCI	19	4.17E-07	4
3	Core damage following an unmitigated SGTR (SG contains inventory) – RC_SGTR WET	24	4 63E-06	2
4	Core damage following an unmitigated SGTR (SG dry) – RC_SGTR DRY	25	0.0	2
5	Core damage following an unmitigated SGTR (SG releases at high pressure, relief valves cycling) – RC_SGTR ARV Cycle	23	1.32E-06	2
6	Core damage event followed by a late global containment failure – RC_LATE_GLOBAL	10	1 05E-06	12
7	Core damage event followed by a late small containment failure (leak) – RC_LATE_SMALL	13	1.05E-06	16.7
8	Core damage event followed by a thermally induced SGTR and dry SG release – RC_TISGTR	25	2 4E-08	2
9	Early containment failure due to rapid containment pressurization following a high pressure reactor vessel failure – RC_HPRCS	16	4.13E-07	5
10	Early containment failure due to containment challenges associated with a low pressure reactor vessel failure – RC_LPRCS	7	7.95E-09	4
11	Core damage in the presence of an intact containment (containment heat removal available) – RC_INTACT	1	2.38E-05	4

a. Release frequencies based on Revision 4.2 of Ginna PSA.

Table 5
Summary of Offsite Economic Impact of Releases
(Maximum Bounding Condition Extrapolated to 2030)

Release Category	Frequency of Release (yr^{-1})	Offsite Dose (person-rem/event)	Offsite Dose Risk (person-rem/yr)	Offsite Economic Costs ($/event)	Offsite Economic Risk ($/yr)
RC_ISLOCA	2.50E-07	1.76E+07	4.40	2.27E+10	5,680
RC_LOCI	4.17E-07	3 38E+06	1.41	1.11E+10	4,630
RC_SGTR WET	4 63E-06	1.15E+06	5 32	9.43E+09	43,700
RC_SGTR DRY	0 0	4.62 E+06	0 00	1.82E+10	0
RC_SGTR ARV Cycle	1 32E-06	6.89E+05	0.91	5.62E+09	7,420
RC_LATE_GLOBAL	1.05E-06[a]	9.39E+05	0 99	9.41E+09	9,880
RC_LATE_SMALL	1.05E-06[a]	4.51E+05	0 47	2.19E+09	2,300
RC_TISGTR	2.40E-08	4.72E+06	0.11	1.90E+10	456
RC_HPRCS	4.13E-07	1.36E+06	0 56	1.06E+10	4,380
RC_LPRCS	7.95E-09	1.93E+05	0 002	8.07E+08	6
RC_INTACT	2.38E-05	2.27E+04	0.54	2.82E+07	671
Total "at power"	3.3E-05		14.71		79,123
Shutdown[b]	6.81E-06	2 26E+05	1.54	1.10E+09	7,490
Total (including shutdown)	3.98E-05		16.26		86,613

a. Late failures were equally divided between "global" and "small" containment failures.
b. Event impacts conservatively estimated as one half of late small leak containment failures.

References

1. R. E. Ginna Nuclear Power Plant Probabilistic Safety Assessment, Final Report, Rev. 4 February, 2002.
2. NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events", January, 1999.
3. R. E. Ginna Nuclear Power Plant Probabilistic Safety Assessment, Final Report, Rev. 4.2, December, 2002.

5. **The SAMA analysis did not include an assessment of the impact that PSA uncertainties would have on the conclusions of the study. Some license renewal applicants have opted to double the estimated benefits (for internal events) to accommodate any contributions for other initiators (e.g., seismic) when sound reasons exist to support such a numerical adjustment, and to incorporate additional margin in the SAMA screening criteria to address uncertainties in other parts of the analysis. Please provide the following information to address these concerns:**

   b. **An assessment of the impact on the SAMA screening process if the risk reduction estimates are increased to account for uncertainties in the risk assessment and the additional benefits associated with seismic events.**

Response to RAI 5b:
The RG&E PSA model fully and conservatively considers internal and external floods and fire. Therefore, the analyses performed with this model are conservative. In addition, seismic is a negligible contributor given the plant location. Therefore, use of an additional factor of two for further bounding uncertainty is not necessary. Regardless, the impact of increasing the benefits by a factor of two will not impact conclusions of this report with regard to the SAMA benefit, as even if the non-viable SAMAs were increased by a factor of two, the resulting cost benefit remains negative.

6. **In Section 1.1 of Appendix E to the ER, RG&E states that an industry peer review was performed in May 2002, and that the findings of the peer review will be incorporated into future revisions of the model. RG&E also states that, while the peer review findings could not be incorporated into the model in time to support the ER submittal, it did account for anticipated model impacts in the analysis of the candidate SAMAs. Please provide details regarding the major findings of the peer review and the potential impact of these findings on the identification and dispositioning of potential SAMAs. Also, describe how the peer review findings were considered or accounted for in the SAMA evaluation.**

Response to RAI 6:
As a result of the WOG peer review recommendations, RG&E updated the PSA and a revised SAMA analysis was performed (results provided in Attachment 2). In the update process most of the significant peer review findings were directly incorporated into Revision 4.2 of the PSA. Highlights of the peer review are provided in response to RAI 2a under subsection Revision 4.2.

A few of the peer review comments were not directly incorporated into the PSA. Of those, most were assessed to have minimal impact on the plant CDF and/or were assessed to have no

impact on the SAMA analysis. Two high-level peer review items not addressed directly in the PSA and that could impact the SAMA analysis relate to the use of scrubbing factors in the determination of LERF. These items were explicitly addressed in the revised SAMA assessment by setting the scrubbing factors contained within the LERF cutset file to unity (no scrubbing). This approach maximized the LERF and dose impact for both the calculation of the maximum attainable benefit and in the evaluation of the individual SAMA candidates.

7. **During the staff's review of the SAMA analysis, numerous inconsistencies and apparent errors were noted, as summarized below. Please reconcile these differences.**

   a. **Table 4.14-2 of the ER, indicates that SAMA 1 reduces CDF by 14.8 percent and has an estimated benefit of $813K. SAMA 5 reduces CDF by 3.3 percent and has an estimated benefit of $844K. Both of these benefits are close to the maximum attainable benefit (MAB) of $992K, and appear to be too high given their relatively small impact on total CDF. Please explain this apparent inconsistency.**

   b. **Table E.1-1 indicates that the interfacing-systems loss-of-coolant accident (ISLOCA) CDF is 8E-7/y (two percent of the total CDF) whereas Table E.1-3 indicates its contribution to LERF is 6E-9/y (2.09E-06/y times 0.3 percent), and Table E.2-4 shows the ISLOCA Release Category contribution to be 4E-9/y. This suggests that there is greater than a two order-of-magnitude difference between the ISLOCA CDF and the ISLOCA LERF and release to the public. Most ISLOCAs in other PSAs are unattenuated containment by-pass events with a conditional large early release probability of 1.0. Please explain the attenuation and mitigation features of ISLOCA events that justify the apparent conditional large early release probability of <1 percent.**

   c. **Table E.1-2 indicates the CDF for fuel handling accident/SFP cooling is 1.3E-6/y (3.37 percent of the CDF) for a fully off-loaded core, whereas Table E.1-3 indicates the LERF for SFP cooling is 4.7E-7/y. Thus, the probability of a large early release, given a spent fuel pool cooling accident, is about 36 percent. Please explain why all of these core damage events do not result in a large early release.**

   d. **In Table E.2-4, the sum of all frequencies is 4.03E-5/y versus the stated CDF of 3.97E-5/y. Although the difference is only 6E-7/y, and may be due to rounding, this is larger than many of the frequency entries in the table. Please explain the reasons for the difference in these values.**

   e. **In Table E.2-4, the sum of all Release Categories that would appear to be LERF contributors is 1.67E-6/y, which is less than the stated LERF of 2.09E-6/y. Please explain the reasons for the difference in these values. Also identify which Release Categories are considered to contribute to LERF.**

   f. **Table E.2-4 reports the frequency of steam generator tube rupture (SGTR) (WET) as 1.02E-6/y, but Table E.1-3 indicates it is 7.5E-7/y. Please explain**

the reasons for the difference in these values. Also, explain why all SGTR events are assumed to be wet.

g. **SAMA 1 - The reduction in CDF is said to be 5.88E-6/y, but from Table E.1-2, SBO is only 2.43 percent of the total CDF, or 9.6E-7/y. Also, SAMA 1 indicates a reduction in population dose by 4.39 person-rem per year, but according to the text and Table E.2-4, the total population dose for all events is 4.09 person-rem per year. Please address these inconsistencies.**

h. **SAMA 5 indicates a reduction in population dose by 17.6 person-rem per year, but according to the text and Table E.2-4, the total population dose for all events is 4.09 person-rem per year. Please address this inconsistency.**

Response to RAI 7a:
Revision 4.2 of the RG&E PSA has been completed and has been integrated into the SAMA evaluation process. Specific assumptions in earlier revisions had credited scrubbing in the determination of the large early release frequency (LERF) of selected events. This particularly affected the treatment of ISLOCA and influenced the results presented in Table 4.14-2 of the ER. The ER SAMA assessments that overlapped with this assumption did not credit this additional scrubbing in the reduction in LERF frequency or the dose assessment. The revised SAMA analysis (Attachment 2) has removed these assumptions from the calculations and, hence, the SAMA evaluations and the maximum attainable benefit (MAB) are evaluated consistently.

With respect to the relative benefits, SAMA 1 is currently estimated to have a CDF benefit of about 25% reduction. This arises from eliminating the full range of SBO and fire/flood induced SBO core damage states. The revised benefit associated with implementation of SAMA 1 is conservatively estimated to be about $944K or approximately 50% of the revised MAB (approximately $1.93M).

Note that as a result of PSA modeling changes, the CDF due to the large ISLOCA from penetration 111 has been reduced from 1.576E-06 per year to 3.56E-07 per year. The percentage reduction associated with implementing SAMA 5 is currently estimated as 0.2% with an averted cost of approximately $45K, which results in a negative net benefit of $955K (approximately 2% of the revised MAB).

Response to RAI 7b:
Revision 4.1 of the RG&E model used to define the base case benefit in the ER credited a 99% scrubbing efficiency for ISLOCAs that release radionuclides through the auxiliary building. This scrubbing capability was used to avert early fatalities and, therefore, remove the event from consideration as a LERF. Scrubbing was attributed to availability of the auxiliary building filter system. Scrubbing during ISLOCAs is expected due to chemical reactions and deposition of particulate fission products in the ISLOCA piping and within the auxiliary building. Additional scrubbing may be available if the break becomes submerged. Furthermore, operation of the auxiliary building filter and ventilation system will also provide some additional level of fission product scrubbing. Revision 4.2 PSA used in the revised SAMA analysis does not credit scrubbing of ISLOCA releases in either the determination of MAB or the evaluation of associated SAMA benefits. Consequently, the ISLOCA CDF and LERF contribution are consistent. This approach has brought the ISLOCA CDF and LERF contribution into agreement, as well as

eliminating the inconsistency noted in response to 7a. Responses to RAI 7a and 7h provide additional detail for the treatment of ISLOCA

Response to RAI 7c:
Spent fuel pool releases are not considered large early releases. Please see the response to 8b included in the January 31, 2003 submittal for additional discussion.

Response to RAI 7d:
The noted difference was due to roundoff. The revised CDF is 3.977E-05 per year. The value is rounded up to 3.98E-05 in the revised SAMA analysis (Attachment 2). Note that a roundoff effect is noted in response to RAI 7i provided in the January 31, 2003 submittal. Where the sum of the accident types in the RAI 7i response is individually taken to be 3.97E-05 and is shown as 3.95E-05 due to roundoff error; however, it should be noted that throughout the revised analysis, RG&E used a CDF of 3.98E-05.

Response to RAI 7e:
The Revision 4.2 LERF value adjusted for the SAMA evaluation is calculated to be 5.74E-06 per year (see revised Appendix E provided in Attachment 2). The contributors to LERF and the respective frequencies are given below. Note that in response to 8c provided in the January 31, 2003 submittal, the Level 2 LERF frequency was given as 3.788E-06 per year. That assessment included consideration of the scrubbing factors discussed in response to RAI 6. As noted above the LERF assessment used in the SAMA evaluations was revised to remove the scrubbing factors.

Summary of Scenarios Contributing to LERF (per Year)	
	Frequency (per year)
1. ISLOCA	2.50E-07
2. SGTR that proceeds to core damage in the presence of an unisolated secondary side (SGTR_Ruptured)	4.63E-06
3. Core damage events that take place in the presence of an unisolated containment (LOCI)	4.17E-07
4. High pressure core damage events that fail containment shortly after vessel breach	4.13E-07
5. High pressure core damage sequences that result in thermally induced SGTR (TI_SGTR)	2.40E-08
6. Low pressure core damage events that fail containment shortly after vessel breach	7.95E-09
Total LERF	5.74E-06

Response to RAI 7f:
Revision 4.2 of the PSA calculates a SGTR CDF of 5.95E-06 per year. This is divided into SGTRs that occur in the presence of a ruptured secondary side (4.63E-06 per year) and those that occur with an intact secondary side (1.32E-06 per year). These events represent 14.95% of the total 3.98E-05 combined plant CDF. This is now consistent with Table E.1-3 of the revised SAMA analysis (See Attachment 2). These events also represent 80.7% of the LERF.

A wet steam generator condition arises as a result of the availability of feedwater during the SGTR core damage sequences. The Ginna Station feedwater system consists of a preferred

AFW system and a Standby AFW system. The preferred AFW consists of two motor-driven feedwater pumps and one turbine-driven AFW pump. The Standby AFW system is located away from the preferred AFW system and consists of two motor-driven pumps. In addition, the main feedwater system may be used for FW control during a SGTR. Consequently, the likelihood that a SGTR core damage event occurs without the presence of a wet steam generator is low. This is confirmed by a review of the core damage cutsets. RG&E credits RCS inventory availability in the SG post-core damage as providing scrubbing to the fission product discharge. Note that all thermally induced SGTR events are considered dry and unisolated, as the event is typically induced by a depressurized secondary side and unavailability of feedwater.

It should be noted that the SG WET assumption has no impact on LERF, as all ruptured SG states are considered to be LERF, nor does this assumption impact individual SAMA assessments, as the proposed changes do not significantly overlap with preventing or mitigating SGTRs.

Response to RAI 7g:
The SBO contribution identified in Table E.1.2 of the ER refers to SBOs associated with internal events only. The revised estimate of this value is presented in the response to RAI 7i provided in the January 31, 2003 submittal. Many fire and flood scenarios result in SBO-like conditions. The combined contribution of fire, floods and SBOs implies a 56.0% contribution to core damage. Removal of SBO-like scenarios from these event categories results in a potential 24.8% reduction in CDF. The total dose reduction associated with the removal of SBO events is 7.07 man-rem per year. This is compared to an "at power" dose of 14.7 man-rem per year.

Response to RAI 7h:
The initial SAMA analysis submitted in the ER included a modeling assumption that ISLOCAs that lead to an auxiliary building release will be scrubbed such that there would only be a 1% release. This assumption affected the calculation of the ISLOCA contribution to the MAB. However, the scrubbing was not credited in the benefits evaluation. It was further assumed in the benefits calculation that the entire ISLOCA CDF also progressed to a large early release. Also it was assumed that all core damage sequences involving penetration 111 would be ISLOCAs even if the event did not result in containment bypass. The net impact of these assumptions was to underestimate the ISLOCA contribution to MAB and overestimate the associated benefit.

Peer review comments cited a need for improvements to the ISLOCA modeling. Included in PSA Revision 4.2 was a Bayesian update using plant-specific data (no failure over operating history) for the rates of passive failure of stainless steel check valves and MOVs. The revised SAMA analysis removed scrubbing credit in both the MAB and benefits calculation and reduced the CDF contribution to Penetration 111. This latter conservative assumption associated with Penetration 111 is retained in the current modeling, however, incorporation of PSA peer review findings have resulted in a decrease of the CDF contribution of Penetration 111 from 1.576E-06 per year to 3.56E-07 per year. The benefit associated with SAMA 5 is a reduction of the CDF associated with Penetration 111 from 3.56E-07 to 2.85E-07 per year. This results in a net CDF reduction of 7.1E-08 per year. Assuming this reduction to be entirely a reduction in bypass events (i.e., ISLOCAs), the dose reduction was estimated to be about 1.25 person-rem per year. This value may be compared to the total ISLOCA contribution (2.5E-07 per year) of 4.4 person rem per year (See Attachment 2). Thus, the current approach produces consistent results.

Based on the revised analysis, SAMA 5 results in a dose reduction of 1.25 person-rem per year and a benefit of approximately $45K (see Attachment 2). Given the estimated cost for this modification is approximately $1M, this SAMA is not considered to be cost beneficial.

4.14 Severe Accident Mitigation Alternatives

NRC

The environmental report must contain a consideration of alternatives to mitigate severe accidents " . . . [i]f the staff has not previously considered severe accident mitigation alternatives for the applicant's plant in an environmental impact statement or related supplement or in an environmental assessment . . . ." 10 CFR 51.53(c)(3)(ii)(L)

"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." 10 CFR Part 51, Subpart A, Appendix B, Table B-1 (Issue 76)

The term "accident" refers to any unintentional event (i.e., outside the normal or expected plant operational envelope) that results in the release or a potential for release of radioactive material to the environment. Generally, the NRC categorizes accidents as "design basis" or "severe." Design basis accidents are those for which the risk is great enough that an applicant is required to design and construct a plant to prevent unacceptable accident consequences. Severe accidents are those considered too unlikely to warrant design controls.

Historically, the NRC has not included in its environmental impact statements or environmental assessments any analysis of alternative ways to mitigate the environmental impacts of severe accidents. A 1989 court decision ruled that, in the absence of an NRC finding that severe accidents are remote and speculative, severe accident mitigation alternatives (SAMAs) should be considered in the NEPA analysis [Limerick Ecology Action v. NRC, 869 F.d 719 (3rd Cir. 1989)]. For most plants, including Ginna Station, license renewal is the first licensing action that would necessitate consideration of SAMAs.

The NRC concluded in its generic license renewal rulemaking that the unmitigated environmental impacts from severe accidents met the Category 1 criteria, but the NRC made consideration of mitigation alternatives a Category 2 issue because ongoing regulatory programs related to mitigation (i.e., Individual Plant Examination and Accident Management) have not been completed for all plants. Since these programs have identified plant programmatic and procedural improvements (and, in a few cases, minor modifications) as cost-effective in reducing severe accident and risk consequences, the NRC thought it premature to draw a generic conclusion as to whether severe accident mitigation would be required for license renewal.

Site-specific information to be presented in the environmental report includes: (1) potential SAMAs; (2) benefits, costs, and net value of implementing potential SAMAs; and (3) sensitivity of the analysis to changes to key underlying assumptions. This section of the environmental report is a synopsis of key site-specific SAMA information. Additional details, as called out in the following sections, are provided in Appendix E.

4.14.1 Methodology Overview

The methodology used to perform the Ginna Station SAMA cost-benefit analysis is based primarily on the handbook used by the NRC to analyze the benefits and costs of its regulatory activities, NUREG/BR-0184 (Ref. 4.14-1), subject to Ginna Station-specific considerations.

Environmental impact statements and environmental reports are prepared using a sliding scale in which impacts of greater concern and mitigative measures of greater potential value receive more detailed analysis than do impacts of less concern and mitigative measures of less potential value. Accordingly, RG&E used less detailed feasibility investigation and cost estimation techniques for SAMAs having disproportionately high costs and low benefits, and more detailed techniques for the most viable candidates.

The following is a brief outline of the approach taken in this SAMA analysis:

- Establish the Base Case – Use NUREG/BR-0184 and the current Ginna Station probabilistic safety assessment (PSA) model at the time of evaluation to evaluate the following severe accident impacts:

  - Offsite exposure costs – Monetary value of consequences (dose) to offsite population:

    Use the Ginna Station PSA model to determine the total accident frequency, which is a function of core damage and containment release frequencies. Use the Melcor Accident Consequences Code System (MACCS) to convert release input to public dose, and the methodology described in NUREG/BR-0184 to convert dose to present-worth dollars based on valuation of $2,000 per person-rem and present-worth discount factor.

  - Offsite economic costs – Monetary value of damage to offsite property:

    Use the Ginna Station PSA model to determine total accident frequency (core damage frequency and containment release frequency); MACCS to convert release input to offsite property damage; and the NRC's NUREG/BR-0184 methodology to convert offsite property damage estimate to present-worth dollars.

  - Onsite exposure costs – Monetary value of dose to workers:

    Use NUREG/BR-0184 best estimate occupational dose values for immediate and long-term dose, then apply the NUREG/BR-0184 methodology to convert dose to present-worth dollars based on valuation of $2,000 per person-rem and present-worth discount factor.

  - Onsite economic costs – Monetary value of damage to onsite property:

    Use NUREG/BR-0184 best estimate cleanup, decontamination, and replacement power costs; then apply the NUREG/BR-0184 methodology to convert onsite property damage estimate to present-worth dollars.

- SAMA Identification – Identify potential SAMAs from the following sources:

  Ginna Station PSA results and staff insights regarding the significant contributors to risk and plant design; SAMA analyses submitted in support of license renewal activities for other nuclear power plants; and NRC and industry documentation discussing potential plant improvements.

- Disposition of SAMAs – Eliminate candidates based on cost-benefit analysis:

  - SAMA impacts – Calculate impacts (i.e., onsite/offsite dose and damages) by using the plant model to simulate revised plant risk following implementation of each individual SAMA.

  - SAMA benefits – Calculate benefits for each SAMA in terms of averted consequences. Averted consequences are the arithmetic differences between the calculated impacts for the base case and the revised impacts following implementation of each individual SAMA.

  - Cost estimate – Estimate the cost of implementing each SAMA. The detail of the cost estimate must be commensurate with the benefit; if a benefit is low, it is not necessary to perform a detailed cost estimate to determine that the SAMA is not cost beneficial—engineering judgment can be applied.

- Sensitivity Analysis – Determine the effect that changing the discount rate would have on the cost-benefit calculation.

- Conclusions – Identify SAMAs that are cost beneficial, if any, and implementation plans or bases for not implementing.

The RG&E SAMA analysis for Ginna Station is presented in the following sections. These sections provide a detailed discussion of the process presented above.

4.14.2 Establishing the Base Case

The purpose of establishing the base case is to provide the baseline for determining the risk reductions (benefits) that would be attributable to the implementation of potential SAMAs. The primary source of data relating to the base case is the Ginna Station PSA model. Severe accident risk is calculated through use of the Ginna Station PSA model and the MACCS2 Level 3 model. The Ginna Station PSA model uses PSA techniques to:

- Develop an understanding of severe accident behavior;
- Understand the most likely severe accident consequences;
- Gain a quantitative understanding of the overall probabilities of core damage and fission product releases; and
- Evaluate hardware and procedure changes to assess the overall probabilities of core damage and fission product releases.

The Ginna Station PSA model includes internal events (e.g., loss of feedwater event, loss-of-coolant accident), external events (fires and flooding), and shutdown events. The model has been upgraded since completion of the Individual Plant Examination

and Individual Plant Examination for External Events (Ref. 4.14-2; Ref. 4.14-3; Ref. 4.14-4; Ref. 4.14-5), and it has been significantly modified to accommodate generic and plant-specific operating data, as well as risk-important plant design and procedural changes implemented since 1994 (e.g., relocation of service water piping to avoid battery room floods and steam generator replacement). Appendix Section E.1 provides additional information pertaining to the evolution of the Ginna Station PSA model, the current risk profile for the station, and risk-important modifications.

The Ginna Station PSA model describes the results of the first two levels of the Ginna Station probabilistic risk assessment for the plant. These levels are defined as follows: Level 1 determines core damage frequencies based on system analyses and human-factor evaluations; and Level 2 evaluates the impact of severe accident phenomena on radiological releases and quantifies the condition of the containment and the characteristics of the release of fission products to the environment.

Using the results of these analyses, the next step is to perform a Level 3 PSA analysis, which calculates the hypothetical impacts of severe accidents on the surrounding environment and members of the public. The MACCS2 computer code is used for determining the offsite impacts for the Level 3 analysis, whereas the magnitude of the onsite impacts (in terms of cleanup and decontamination costs and occupational dose) are based on information provided in NUREG/BR-0184. The principal phenomena analyzed are: atmospheric transport of radionuclides; mitigating actions (i.e., evacuation, condemnation of contaminated crops and milk) based on dose projection; dose accumulation by a number of pathways, including food and water ingestion; and economic costs. Input for the Level 3 analysis includes the reactor core radionuclide inventory, Ginna Station source terms (as applied to the Ginna Station PSA model), site meteorological data, projected population distribution (within a 50-mile radius) for the year 2030, emergency response evacuation modeling, and economic data. Appendix Section E.2 describes the MACCS2 input data, assumptions, and results.

4.14.2.1 Offsite Exposure Costs

The Level 3 base case analysis shows an annual offsite exposure risk of 16.26 person-rem. This calculated value is converted to a monetary equivalent (dollars) via application of the NRC's conversion factor of $2,000 per person-rem. This monetary equivalent is then discounted to present value using the NRC standard formula (Ref. 4.14-1):

$$W_{pha} = C \times Z_{pha}$$

where:

W_{pha} = monetary value of public health risk after discounting ($)

$C = [1 - \exp(-rt_f)]/r$

where:

t_f = years remaining until end of facility life (20 years)

r = real discount rate (as fraction) (0.07)

Z_{pha} = monetary value of public health (accident) risk per year before discounting ($/year)

Using a 20-year period for remaining plant life and a seven percent discount rate results in a value of approximately 10.76 for C. Therefore, calculating the discounted monetary equivalent of public health risk involves multiplying the dose (person-rem per year) by $2,000 and by the C value, approximately 10.76. The resulting monetary equivalent $350,000

4.14.2.2 Offsite Economic Costs

The Level 3 analysis shows that the offsite property loss factor multiplied by accident frequency yields an annual offsite economic risk of $86,613. Calculated values for offsite economic costs caused by severe accidents are also discounted to present value. Discounting is performed in the same manner as for the Offsite Exposure Costs discussed above. The resulting monetary equivalent is $932,000.

4.14.2.3 Onsite Exposure Costs

Values for occupational exposure associated with severe accidents are not derived from the Ginna Station PSA model, but instead are obtained from information published by the NRC. Occupational exposure consists of "immediate dose" and "long-term dose." The best-estimate value provided by the NRC for immediate occupational dose is 3,300 person-rem, and long-term occupational dose is 20,000 person-rem (over a ten-year cleanup period). The following equations are applied to these values to calculate monetary equivalents.

Immediate Dose

For a currently operating facility, the NRC, in NUREG/BR-0184, recommends calculating the immediate dose present value with the following equation:

Equation (1):

$$W_{IO} = \left(F_S D_{IO_S} - F_A D_{IO_A}\right) R \frac{1 - e^{-rt_f}}{r} \qquad (1)$$

where:

W_{IO} = monetary value of accident risk avoided due to immediate occupational dose, after discounting ($)

R = monetary equivalent of unit dose ($/person-rem)

F = accident frequency (events/year)

D_{IO} = immediate occupational dose (person-rem/event)

s = subscript denoting status quo (current conditions)

A = subscript denoting after implementation of proposed action

r = real discount rate
t_f = years remaining until end of facility life

The values used in the analysis are:

R = $2,000/person-rem
r = 0.07
D_{IO} = 3,300 person-rem/accident (best estimate)
t_f = 20 years

Assuming F_A is zero for the base case, the monetary value of the immediate dose associated with Ginna Station's accident risk is:

$$W_{IO} = \left(F_S D_{IO_S}\right) R \frac{1-e^{-rt_f}}{r}$$

$$= 3300 * F * \$2000 * \frac{1-e^{-.07*20}}{.07}$$

The core damage frequency (CDF) for the base case is 3.98E-05 per year; therefore,

W_{IO} = $3,000

Long-term Dose

For a currently operating facility, the NRC, in NUREG/BR-0184, recommends calculating the long-term dose present value with the following equation:

Equation (2):

$$W_{LTO} = \left(F_S D_{LTO_S} - F_A D_{LTO_A}\right) R * \frac{1-e^{-rt_f}}{r} * \frac{1-e^{-rm}}{rm} \qquad (2)$$

where:

W_{LTO} = monetary value of accident risk-avoided long-term doses, after discounting ($)
F = accident frequency (events/year)
s = subscript denoting status quo (current conditions)
A = subscript denoting after implementation of proposed action
t_f = years remaining until end of facility life
r = real discount rate
R = monetary equivalent of unit dose ($/person-rem)
D_{LTO} = long-term occupational dose (person-rem/event)
m = years over which long-term doses accrue

The values used in the analysis are:

R = $2,000/person-rem
r = 0.07
D_{LTO} = 20,000 person-rem/accident (best estimate)
m = "as long as 10 years"
t_f = 20 years

Assuming F_A is zero for the base case, the monetary value of the long-term dose associated with the plant accident risk is:

$$W_{LTO} = (F_S D_{LTO_S}) R * \frac{1-e^{-rt_f}}{r} * \frac{1-e^{-rm}}{rm}$$

$$= (F_S \times 20000)\$2000 * \frac{1-e^{-.07*20}}{.07} * \frac{1-e^{-.07*10}}{.07*10}$$

The CDF (F) for the base case is 3.98E-05 per year; therefore,

W_{LTO} = $12,000

Total Occupational Exposures

Combining Equations (1) and (2) above and using the above numerical values, the long-term accident related onsite (occupational) bounding dose (W_O) is equivalent to:

$W_O = W_{IO} + W_{LTO}$ = $15,000

4.14.2.4 Onsite Economic Costs

Onsite economic costs are considered to include costs associated with cleanup/ decontamination, replacement power, and repair/refurbishment. Each of these factors is discussed in the following sections.

Cleanup and Decontamination

The total undiscounted cost estimate of cleanup and decontamination of a power facility subsequent to a severe accident is estimated by the NRC, in NUREG/BR-0184, at $1.5E+09. Assuming the $1.5E+09 estimate is spread evenly over a 10-year period for cleanup and applying a seven percent real discount rate, the cost translates into a net present value of $1.1E+09 for a single event. This quantity is derived from the following equation:

$$PV_{CD} = \left(\frac{C_{CD}}{m}\right)\left(\frac{1-e^{-rm}}{r}\right)$$

where:

PV_{CD} = present value of the cost of cleanup/decontamination ($)
C_{CD} = total cost of the cleanup/decontamination effort ($1.5E+09)
m = cleanup period (10 years)
r = real discount rate (7 percent)

Therefore:

$$PV_{CD} = \left(\frac{\$1.5E+09}{10}\right)\left(\frac{1-e^{-.07*10}}{.07}\right)$$

$$PV_{CD} = \$1.079E+09$$

This cost is integrated over the license renewal period as follows:

$$U_{CD} = PV_{CD}\frac{1-e^{-rt_f}}{r}$$

where:

U_{CD} = net present value of cleanup/decontamination over the life of the plant ($)
t_f = years remaining until end of facility life

Based upon the values previously assumed:

$$U_{CD} = \$1.161E+10$$

Replacement Power

Replacement power costs, U_{RP}, are an additional contributor to onsite costs. These are calculated in accordance with NUREG/BR-0184, Sections 5.7.6.4 and 5.6.7.2. Since replacement power will be needed for the time period following a severe accident and for the remainder of the expected generating plant life, long-term replacement power calculations have been used. Values used in the calculations are based on the 910-megawatts (electric) [MW(e)] reference plant.

$$PV_{RP} = \left(\frac{\$1.2E+08}{r}\right)\left(1-e^{-rt_f}\right)^2$$

where:

PV_{RP} = present value of the cost of replacement power for a single event ($)
t_f = years remaining until end of facility life
r = real discount rate

This equation was developed per NUREG/BR-0184 for discount rates between 5 percent and 10 percent only. It was developed using the constant $1.2E+08, which has no intrinsic meaning, but is a substitute for a string of non-constant replacement power costs that occur over the lifetime of a "generic" reactor after an event.

To account for the entire lifetime of the facility, U_{RP} was then calculated from PV_{RP}, as follows:

$$U_{RP} = \frac{PV_{RP}}{r}\left(1-e^{-rt_f}\right)^2$$

where:

U_{RP} = present value of the cost of replacement power over the life of the facility ($)

Based upon values previously assumed:

U_{RP} = $7.89E+09

Applying the correction for a 490 MW(e) Ginna Station versus 910 MW(e) for the "generic" reactor, U_{RP} = $4.25E+09

Repair and Refurbishment

RG&E has no plans for major repair/refurbishment following a severe accident; therefore, there is no contribution to averted onsite costs from this source.

Total Onsite Economic Cost

The total onsite economic cost is the sum of the cleanup/decontamination cost (U_{CD}) and the replacement power cost (U_{RP}) multiplied by the CDF (3.98E-05 per year). Therefore, the total onsite economic cost is $631,000.

4.14.2.5 Maximum Attainable Benefit

The present-dollar value equivalent for severe accidents at Ginna Station is the sum of the offsite exposure costs, offsite economic costs, onsite exposure costs, and onsite economic costs. Table 4.14-1 lists each of these values for the base case as calculated in the previous sections. As shown, the monetized value of severe accident risk is approximately $1,928,000.

The maximum theoretical benefit is based upon the elimination of all plant risk and equates to the base case severe accident risk described above. Therefore, the maximum attainable benefit is $1,928,000.

4.14.3 SAMA Identification

RG&E identified candidate modifications by focusing on station risk and design characteristics. RG&E considered insights into possible Ginna Station-specific improvements gained through the development and use of the Ginna Station PSA model over the past decade. RG&E focused on the dominant risk sequences identified by the model, as well as the results of other risk-importance studies to further focus the evaluation. Appendix Section E.1 provides details of the Ginna Station risk profile. Additional insights were gained from reviewing candidate modifications identified in previous license renewal SAMA evaluations submitted to the NRC by other licensees. As conceptual modifications were formulated, RG&E balanced the order-of-magnitude cost against the maximum attainable benefit. Those conceptual ideas whose cost would greatly exceed the maximum attainable benefit were not considered further.

4.14.4 Cost-Benefit Analysis

The cost-benefit analysis involved developing Ginna Station-specific SAMA descriptions and cost-benefit analyses for the viable candidate SAMAs. RG&E developed general descriptions as to how each potential SAMA would be implemented to provide a basis for bounding benefit and cost estimates. Each SAMA description provides the analysts with a detailed description that can be compared with the current plant configuration and processes. Appendix Section E.3 provides a description for each candidate SAMA.

Table 4.14-1
Estimated Present Dollar Value Equivalent for Severe Accidents at Ginna Station

Parameter	Present Dollar Value
Onsite Economic Costs	$631,000
Offsite Economic Costs	$932,000
Onsite Exposure Costs	$15,000
Offsite Exposure Costs	$350,000
Total	**$1,928,000**

RG&E then prepared site-specific cost estimates for implementing each candidate SAMA. Conservatively, the cost estimates included neither the cost of replacement power during extended outages required to implement the modifications, nor the contingency costs associated with unforeseen implementation obstacles. Estimates were presented in terms of dollar values at the time of implementation or estimation, and were not adjusted to present-day dollars.

Consistent with the methodology presented in Section 4.14.2, RG&E calculated the maximum benefit for each potential SAMA. The methodology for determining if a SAMA is beneficial consists of determining whether the benefit provided by implementation of the SAMA exceeds the expected cost of implementation. The benefit is defined as the sum of the reductions in the dollar equivalents for each severe accident impact (offsite exposure costs, offsite economic costs, occupational exposure costs, and onsite economic costs) resulting from the implementation of a SAMA.

The result of implementation of each SAMA would be a change in the Ginna Station severe accident risk (i.e., a change in frequency or consequence of severe accidents)[1]. The methodology for calculating the magnitude of these changes is straightforward. First, the Ginna Station severe accident risk after implementation of each SAMA was calculated using the same methodology as for the base case. A spreadsheet was then used to combine the results of the Level 2 model with the Level 3 model to calculate the post-SAMA risks. The results of the benefit analysis for each of the SAMAs are presented in Section 4.14.5.

As described above for the base case, values for avoided public and occupational health risk (benefits) were converted to a monetary equivalent (dollars) via application of the NRC's conversion factor of $2,000 per person-rem (Ref. 4.14-1)

[1] Frequency x consequence = risk.

and discounted to present value. Values for avoided offsite economic costs were also discounted to present value. The formula used for calculating net value for each SAMA is as follows:

Net value = ($APE + $AOC + $AOE + $AOSC) – COE

where:

$APE = monetized value of averted public exposure ($)
$AOC = monetized value of averted offsite costs ($)
$AOE = monetized value of averted occupational exposure ($)
$AOSC= monetized value of averted onsite costs ($)
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 the SAMA would not be considered cost-beneficial. The projected cost of each SAMA (COE) was derived by knowledgeable Ginna Station staff. RG&E staff prepared screening level plant-specific cost estimates that address the major cost considerations for implementing each SAMA. Additional detail for the candidate SAMA cost estimates is provided in Appendix Section E.3.

4.14.5 Results

RG&E used Revision 4.2 of the Ginna Station PSA model (dated November 2002) and developed a limited Level 3 model to conduct the SAMA analysis. Using these models, RG&E analyzed eight plant-specific alternatives for mitigating Ginna Station severe accident impacts. Table 4.14-2 presents the analysis results, including the percentage of CDF reduction, the estimated benefit, the estimated cost of the enhancement, and the net benefit for each of the candidate SAMAs evaluated. The cost-benefit evaluation indicates two candidate SAMAs are potentially cost beneficial for mitigating the consequences of a severe accident. These include:

- Obtaining a skid-mounted 480-volt (V) diesel generator (SAMA No. 1).
- Modifying procedures to allow charging pumps B and C to be manually aligned to Bus 14 (SAMA No. 4).

In NUREG/BR-0184, the NRC recommends using a seven percent real (i.e., inflation-adjusted) discount rate for value-impact analyses and notes that a three percent discount rate should be used for sensitivity analyses to indicate the sensitivity of the results to the choice of discount rate. This reduced discount rate takes into account the additional uncertainties (i.e., interest rate fluctuations) in predicting costs for activities that would take place several years in the future. Although by using a three percent discount rate the magnitude of the net benefit increases for each of the candidate SAMAs, no additional SAMA candidates have a positive net benefit.

Table 4.14-2
Disposition of SAMAs Related to Ginna Station

SAMA No.	Potential Enhancement	CDF Reduction	Estimated Benefit	Estimated Cost of Enhancement	Screening Result and Discussion
1	Obtain a skid-mounted 480V diesel generator	24.8%	$944,000	$400,000	Positive net benefit of $544,000. Implementation would potentially mitigate all station blackout sequences. (Using a 3 percent discount rate, the net benefit is $924,000.)
2	Obtain a third fire water source independent of existing suction source for the motor- and diesel-driven fire pumps	1.8%	$70,000	$200,000	Negative net benefit of $130,000. [Using a 3 percent discount rate, the net benefit is ($102,000).]
3	Add a standby charging pump powered from a protected AC source	11.2%	$107,000	$1,100,000	Negative net benefit of $993,000. [Using a 3 percent discount rate, the net benefit is ($947,000)]
4	Modify procedures to allow charging pump B or C to be manually aligned to Bus 14	9.1%	$83,000	$20,000	Positive net benefit of $63,000. Implementation would potentially mitigate fires requiring entry into procedure "Alternative Shutdown for Control Complex Fire" or mitigate fires that would disable train B when the A charging pump fails to run. (Using a 3 percent discount rate, the net benefit is $98,000.)
5	Add redundant check valves in the two RHR injection lines to the RCS	0.2%	$45,000	$1,000,000	Negative net benefit of $955,000. [Using a 3 percent discount rate, the net benefit is ($936,000).]

Table 4.14-2 (continued)
Disposition of SAMAs Related to Ginna Station

SAMA No.	Potential Enhancement	CDF Reduction	Estimated Benefit	Estimated Cost of Enhancement	Screening Result and Discussion
6	Modify motor-driven AFW pump cooling system to be independent of SW	1.8%	$13,000	$200,000	Negative net benefit of $187,000. [Using a 3 percent discount rate, the net benefit is ($181,000).]
7	Modify AOV 112C to fail closed and AOV 112B to fail open on loss of instrument air	2.0%	$14,000	$50,000	Negative net benefit of $36,000. [Using a 3 percent discount rate, the net benefit is ($30,000).]
8	Reconfigure the PORVs so they transfer automatically from instrument air to N2 on low pressure and convert N2 supply line AOV to DC powered motor-operated valve	1.6%	$24,000	$400,000	Negative net benefit of $376,000. [Using a 3 percent discount rate, the net benefit is ($367,000).]

AC = alternating current
AFW = auxiliary feedwater
AOV = air-operated valve
CDF = core damage frequency
DC = direct current
ISLOCA = Interfacing System Loss-of-Coolant Accident
PORV = power-operated relief valve
RCS = reactor coolant system
RHR = residual heat removal
SW = service water
V = volt

Previous applicants have used a factor of 2 applied to the benefit results to account for modeling uncertainties; however, RG&E did not feel this was warranted. Where necessary, RG&E conducted a bounding analysis and recognizes that some of the benefits may be overestimated. Given that the RGE model fully and conservatively considers internal and external floods and fire, the assessments of this model are conservative and bounding for both internal and external events. Regardless, the impact of increasing the benefits by a factor of two does not impact conclusions of this report with regard to the SAMA net benefit (i.e., the SAMAs with a negative net benefits increased by a factor of two still result in a negative net benefit).

In the GEIS, the NRC concluded that the probability-weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to groundwater, and societal and economic impacts of severe accidents are of small significance for all plants. RG&E concurs with that conclusion and addressed site-specific measures to mitigate severe accidents in this analysis. RG&E determined the potentially cost-beneficial SAMAs identified do not relate to adequately managing the effects of aging and, therefore, would not be required to be implemented pursuant to 10 CFR 54.

However, RG&E has historically identified and implemented various plant improvements at Ginna Station in order to reduce the CDF and the consequences of postulated accidents. Accordingly, RG&E will continue to consider implementation of these potentially cost-beneficial modifications through the current plant change process.

APPENDIX E. SEVERE ACCIDENT MITIGATION ALTERNATIVES

E.1 Ginna Station PSA Model and Risk Profile

E.1.1 PSA Model Background

In response to Generic Letter 88-20, "Individual Plant Examination of Severe Accident Vulnerabilities," and its supplements, Rochester Gas and Electric Corporation (RG&E) performed a Level 1 and full-scale Level 2 probabilistic safety assessment (PSA) for R.E. Ginna Nuclear Power Plant (Ginna Station). In March 1994, RG&E submitted a report to the U.S. Nuclear Regulatory Commission (NRC) documenting the methodology and a summary of the final results. This original Individual Plant Examination (IPE) constituted what has been historically designated as Revision 0 of the Ginna Station PSA (Ref. E.1-1). The purpose of the IPE was to achieve the following objectives:

a. Develop an appreciation for severe accident behavior;

b. Understand the most likely severe accident sequences that could occur;

c. Gain a more quantitative understanding of the overall probabilities of core damage and fission product releases; and

d. Reduce, if necessary, the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would prevent or mitigate severe accidents.

In addition, the information obtained through achievement of the above objectives has been used for many other purposes (e.g., on-line maintenance). As such, RG&E incorporated many other features and attempted to address additional issues beyond those required by Generic Letter 88-20. Consequently, the IPE is considered a subset of the PSA since the PSA is intended to be used for future issues and concerns.

Since that time, RG&E has expanded the original models and factored into the analysis several items, such as a change to an 18-month fuel cycle; replacement of the steam generators; conversion to Improved Technical Specifications; monitoring of system, structure, and component performance under the Maintenance Rule; and analysis of the risk from internal fires, floods, and shutdown operation. In addition, the NRC raised several questions concerning the original models, and these questions have subsequently been addressed. All these updates constitute model Revisions 1 through 4.2, with Revision 4.0 being the most recent submitted to the NRC in February 2002. The following paragraphs provide a brief summary of each revision.

Revision 1 (Ref. E.1-2) was produced primarily in response to questions raised by the NRC as a result of the original IPE submittal. An extensive re-analysis of the Level 1 PSA was performed, including significant enhancement of the modeling for human reliability.

Revision 2 (Ref. E.1-3) provided a supplement to Revision 1 and provided a detailed Level 2 (containment performance) analysis. This submittal completed the response to the original IPE questions.

Revision 3 was completed in January 2000, and incorporated the risk from internal fires and the risk during shutdown operation. Additional upgrades were performed in the model for internal flooding and quantification of initiating event frequencies and common-cause failures to incorporate the most recent operating data and industry advances. Additional human reliability analysis was performed to address operator actions during fires, floods, and shutdown. The Level 2 analysis was also updated by merging selected results from the previous detailed Level 2 analysis with the simplified methodology advocated by the NRC in NUREG/CR-6595.

Revision 4 (Ref. E.1-4) accounts for a major modification performed in December 2000, to eliminate the dominant contributor to core damage frequency that was identified during the updated flooding analysis in Revision 3 (large Service Water flood in the Battery rooms). The generic and plant-specific data for component failures have been updated from the time frame used in the original IPE (1980s) to account for industry and plant-specific operation through 2000. RG&E also explicitly modeled the risk from hydrogen and other exothermic explosions.

An industry peer review was performed in May 2002. In preparation for the peer review, RG&E conducted an internal self-assessment. Incorporation of the results of this self-assessment generated Revision 4.1, which was used for the peer review and the inititial severe accident mitigation alternative (SAMA) analysis. While the peer review findings could not be incorporated into the model in time to support the initial assessment included in the ER, RG&E did account for anticipated model impacts in the analysis of the candidate SAMAs. The findings of the peer review have been incorporated into the Revision 4.2 model or have otherwise been accounted for in the revised SAMA assessment.

Revision 4.2 incorporated the most significant Findings and Observations from the Westinghouse Owner's Group Peer Review conducted in May 2002. These changes include (a) refining the Bayesian updating process and re-evaluating common cause failures to remove conservatism; (b) strengthening the methodology for incorporating the effect of dependency among post-initiator actions during diagnostic and post-diagnostic phases; (c) removing inconsistencies within event trees for transients, steam generator tube rupture, and anticipated transient without scram; (d) increasing frequencies for loss of offsite power to include all severe weather events; and (e) removing frequencies for loss of spent fuel pool cooling and fuel handling accidents from the core damage frequency for separate analysis.

Results of the Revision 4.2 model are presented in the following sections. Note that the changes in the model have resulted in changes to the distribution of contributors to both CDF and LERF. The actual absolute change to plant CDF was small, and the reassessment of LERF contribution has resulted in an increase in the estimated overall LERF.

E.1.2 Ginna Station Risk Profile

The current total core damage frequency (CDF) is 3.98E-05. Table E.1-1 provides a ranking of the accident scenarios contributing greater than two percent of the overall CDF. Table E.1-1 indicates that external events dominate the risk profile for Ginna Station. Specifically, floods within the Turbine Building and fires within the Control Room, Diesel Generator rooms, Turbine Building, or battery rooms dominate the results. Control Room fires where evacuation is required and only a limited set of equipment is available dominate fire events. Other fire locations fail either alternating current or direct current electrical trains, which also limits the available equipment. Flooding events are dominated by floods within the Turbine Building, as floods in this location can fail the 4160 VAC buses that supply all four 480-VAC safeguards buses.

As noted earlier, the full-scope Level 2 analysis has been replaced by a simplified approach based on NUREG/CR-6595. The large early release frequency used in the SAMA analysis is 5.74E-06. The results of the Level 2 analysis are shown in Table E.1-2 and indicate that steam generator tube ruptures dominate the large early release frequency.

Revision 4.2 removed scrubbing factors that had been applied to the LERF frequency for bypass events. This notably increased LERF contributors from ISLOCA and SGTRs. The approach also relocated the shutdown releases and the containment overpressure scenarios due to simultaneous loss of sprays and containment recirculation fan coolers to the late release category, as these events will result in significant radioactive releases a considerable time after the onset of core damage.

Table E.1-1
CDF Contributions by Accident Type

Accident Type	Scenario	CDF (1/yr)	Accident Type Total (per year)	Percent of CDF	Accident Type Total
ATWS	ATWS	2.03E-07	2.03E-07	0.51%	0.51%
Fire	Fire – Aux Bldg	4.32E-07		1.09%	
	Fire – Battery Room	7.95E-07		2.00%	
	Fire – Control Room	7.56E-06		19.01%	
	Fire – DG	5.76E-07		1.45%	
	Fire – Turb Bldg	1.06E-06		2.67%	
	Fire – Other	1.01E-06	1.14E-05	2.53%	28.74%
Flood	Flood – Aux Bldg	1.06E-06		2.67%	
	Flood – Relay Room	1.05E-06		2.64%	
	Flood – Turb Bldg	6.16E-06		15.49%	
	Flood – Other	5.06E-07	8.78E-06	1.27%	22.07%
LOCA < 2 inch	SGTR	5.95E-06		14.95%	
	Small LOCA	2.55E-06	8.49E-06	6.40%	21.35%
LOCA > 2 inch	ISLOCA	2.50E-07		0.63%	
	Med/Large LOCA	7.00E-07	9.50E-07	1.76%	2.39%
SBO	SBO	2.10E-06	2.10E-06	5.28%	5.28%
Shutdown	Shutdown – RHR	6.23E-06		15.67%	
	Shutdown – Other	5.75E-07	6.81E-06	1.45%	17.11%
Transient	HELB	5.81E-07		1.46%	
	Transient – Other	4.33E-07	1.01E-06	1.09%	2.55%
TOTAL			**3.98E-05**		**100.00%**

Table E.1-2
Contribution to the Large Early Release Frequency by Accident Type

Accident Type	Percent Contribution
Steam generator tube rupture	80.7
Containment isolation failures	7.3
Containment failure at high reactor coolant system pressure when the reactor vessel ruptures	7.2
Interfacing system loss-of-coolant accident	4.3
Temperature-induced steam generator tube rupture	0.4
Containment failure at low reactor coolant system pressure when the reactor vessel ruptures	0.1

E.1.3 Importance Analysis

The importance of systems and components is a significant insight into the risk profile for Ginna Station. RG&E has generated two types of importance measures, Fussell-Vesely (F-V) and Risk Achievement Worth (RAW), the results of which are briefly summarized below. To support the Ginna Station PSA, RG&E combined these two importance measures with the F-V value greater than 0.05 at the system level (greater than 0.005 at the component level) and the RAW greater than 10 at the system level (greater than 2 at the component level) to indicate "high" risk significance. Revision 4.2 PSA importance analysis results are provided in Table E.1-3.

Table E.1-3
Level 1 CDF Importance Analysis Results

EIN	DESCRIPTION	PROB	F-V	RAW
Initiating Events				
FI0CR3-1	Fire in Zone CR-3 (Control Room Fire Scenario 1 or 2)	1.70E-03	1.58E-01	93.93
TX000RHR	Loss of RHR During Shutdown	4.06E-04	1.51E-01	374.32
FL000TB3	Steam Flooding Event in Turbine Building	6.32E-03	8.05E-02	13.65
LI0SGTRA	Steam Generator Tube Rupture in SG A	2.44E-03	7.50E-02	31.65
ACLOPSHTDN	Loss of Offsite Power During 24-hour Period when Shutdown	4.19E-04	7.49E-02	179.09
LI0SGTRB	Steam Generator Tube Rupture in SG B	2.44E-03	7.47E-02	31.51
Human Errors				
FSHFDCR-3-X	Fire brigade fail to manually suppress fire in Control Room	1.85E-02	1.90E-01	11.086
XXHFGSGTRE	Operators fail to respond to signals indicating SGTR (Early)	2.00E-03	8.55E-02	43.62
RHHFDREC04	Operators fail to recover RHR system before onset of boiling (4-12 hours)	1.00E-02	2.22E-02	3.2
RCHFPCDTR2	Operator fails to cool down to RHR after SI fails - SGTR	9.66E-03	3.05E-02	4.13
SWHFDSTART	Operator fails to start SW pump	6.44E-03	1.86E-02	3.87
IFHFDAFWSW	Operators fail to locally align SW to TDAFW and SAFW suction following CR evacuation for floods or fires (ER-FIRE)	1.39E-02	1.82E-02	2.29
XXHFGNOAFW	Operators fail to diagnose a loss of all AFW	3.55E-04	2.19E-02	62.67
IFHFDTBISL	Failure to isolate large TB flood	1.88E-03	9.18E-03	5.87
CVHFDPMPST	Operators fail to manually load charging pump	6.44E-03	8.12E-03	2.25
FSHFDTSCLT	Operators fail to use the TSC Battery Charger for long-term loss of AC Train B	7.29E-03	8.00E-03	2.09
AXHFPSAFWX	Operator fails to align and start SAFW pumps C & D	2.58E-03	7.60E-03	3.94
RRHFDRECRC-M	Operator fails to correctly shift the RHR system to recirculation and isolate CS-MBLOCA	5.25E-03	8.10E-03	2.53
RCHFPCOOLD	Operators fail to correctly shift the RHR after ARV sticks open or overfill occurs during SGTR	9.66E-03	1.98E-02	3.02
RCHFPCDDPR	Operators fail to cool down and depressurize to prevent SG overfill during SGTR	6.44E-03	1.61E-02	3.48

Table E.1-3 (continued)
Level 1 CDF Importance Analysis Results

EIN	DESCRIPTION	PROB	F-V	RAW
RHHFDREC24	Operators fail to recover RHR system before onset of boiling (12 – 24 hours)	5.00E-03	1.11E-02	3.21
MSHFPISOLR	Operators fail to isolate a ruptured steam generator	6.44E-03	8.40E-03	2.30
Test and Maintenance Activities				
CVTMCHPMPA	Test or maintenance renders charging pump A unavailable	7.04E-02	9.03E-02	2.192
DGTM00001B	Diesel Generator KDG01B unavailable due to testing or maintenance	1.74E-02	3.86E-02	3.177
DGTM00001A	Diesel Generator KDG01A unavailable due to testing or maintenance	1.55E-02	3.00E-02	2.903
AFTM0TDAFW	TDAFW pump out-of –service for maintenance	1.34E-02	2.57E-02	2.889
Systems				
DG			3.20E-01	1919.596
RCS			2.92E-01	8351851
Fire Protection			2.84E-01	25145.02
RHR			2.21E-01	647980.3
Offsite Power			1.82E-01	10344.21
CVCS			1.71E-01	25145.02
MS			1.40E-01	164.3292
CCW			1.09E-01	13090.92
AFW			9.71E-02	941888.8
SW			7.94E-02	8364.804
Motor-Operated Valves				
738A	MOV 738A fails to open	3.90E-03	1.30E-02	4.315
738B	MOV 738B fails to open	3.90E-03	1.28E-02	4.257
Air-Operated Valves				
430	PORV PCV-430 fail to reseat after steam relief	5.00E-03	6.57E-03	2.306
Pumps, Compressors, and Fans				
PAF03	Failure of TDAFW pump train components	9.68E-03	1.72E-02	2.762
Major Electrical Components				
KDG01B/run	Diesel Generator B fails to run	4.22E-02	8.01E-02	2.815
KDG01A/run	Diesel Generator A fails to run	4.22E-02	7.47E-02	2.693
KDG01B/start	Diesel Generator B fails to start	1.01E-02	4.10E-02	5.013
KDG01A/start	Diesel Generator A fails to start	1.01E-02	3.96E-02	4.880
IBPDPCBCB	120 VAC Instrument Bus C (IBPDPCBCB) bus faults	2.09E-05	1.57E-02	745.85

E.1.4 Station Design Features and Improvements

There are several unique and important features of the Ginna Station that contribute to core damage prevention. In addition, as a result of the insights obtained from the Ginna Station PSA, several identified vulnerabilities have resulted in station modifications and procedural changes.

E.1.4.1 Station Design Features Important to Core Damage Prevention

Station design features that are important to core damage prevention include the SAFW system, limited requirements for ventilation, the service water system design, and use of the City Water system. These are briefly discussed in the following paragraphs.

The SAFW system comprises two 100 percent motor-driven pumps that are completely redundant to the preferred auxiliary feedwater system. The SAFW system was installed to mitigate the potential common-mode failures of the preferred auxiliary feedwater system (e.g., high energy line breaks in the Intermediate Building). As such, four motor-driven auxiliary feedwater pumps and one turbine-driven auxiliary feedwater pump are available, any one of which can facilitate steam generator cooling.

The Ginna Station layout typically does not include the use of compartments or rooms to protect various trains from one another. Instead, system components are generally grouped together on one floor level. This configuration eliminates the need for dynamic equipment cooling by enabling passive cooling to occur via the large air volumes and recirculation.

The service water system design is one of a large loop header that is supplied by four pumps. Two pumps are powered from one electrical train and two pumps are powered from a second electrical train. In-series motor-operated valves are also provided at various points to isolate non-critical loads on the loop header. Any one of the four service water pumps can provide cooling water to any system load. This design allows significant flexibility, which reduces the service water contribution to core damage.

The City Water system is used to supply plant domestic loads and the yard fire loop. The yard fire loop consists of the fire hydrants that are located outside the power block. Sprinkler systems and hose reels within the power block are supplied by two onsite fire pumps (one motor-driven and one diesel-driven). In the event that all service water is lost, the City Water system can be used to supply the SAFW system and provide cooling water to the diesel generators.

E.1.4.2 Summary of Station Modifications

As a result of the insights gained from the Ginna Station PSA, RG&E has implemented station modifications or procedural changes to address identified vulnerabilities. The following vulnerabilities have been addressed:

- **Standby auxiliary feedwater system out-of-service activities** – The SAFW system is specifically credited for providing steam generator cooling water in the event of a high-energy line break in the Intermediate or Turbine Building.

Procedural modifications were made to avoid situations in which both trains of the SAFW system could be taken out of service at the same time.

- **Removal of large service water piping within battery rooms** – The service water piping that ran through the two battery rooms was relocated to avoid the potential loss of both battery rooms due to failure in isolating non-safety related service water line breaks prior to flooding the rooms and failing direct-current equipment.
- **Procedural guidance for relay room internal floods** – The procedure, "Alternate Shutdown for Control Complex Fire," was revised to also apply to relay room floods. Previously the relay room procedure only addressed fire.
- **Fire in the diesel generator B vault** – A new procedure was developed to instruct plant personnel to manually close the Bus 18 breakers to prevent a station blackout condition in the event of a worst-case fire that fails the B electrical train (Buses 16 and 17) and offsite power and control power to Bus 18 of electrical train A, which, in turn, would result in loss of all service water.
- **Guidance in control room evacuation due to fires** – Changes were made to the control room evacuation procedures to require entry into the emergency operating procedures to provide necessary core cooling.

E.2 Melcor Accident Consequences Code System Modeling

This section of Appendix E describes the assumptions made and the results of modeling performed to assess the risks and consequences of severe accidents (U.S. Nuclear Regulatory Commission Class 9).

The Level 3 analysis was performed using the Melcor Accident Consequences Code System (MACCS) 2 code (Ref. E.2-1). MACCS2 simulates the impacts of severe accidents at nuclear power plants upon the surrounding environment. The principal phenomena considered in MACCS2 are atmospheric transport, mitigative actions based on dose projections, dose accumulation by a number of pathways including food and water ingestion, early and latent health effects, and economic costs. Input for the Level 3 analysis includes the reactor core radionuclide inventory, source terms from the Ginna Station PSA model, site meteorological data, projected population distribution (within a 50-mile radius), emergency response evacuation modeling, and economic data. These inputs are described in the following section.

E.2.1 Input Data

The input data required by MACCS2 are outlined below.

E.2.1.1 Core Inventory

RG&E calculated the core inventory activity for fission products and actinides for the purpose of developing sources for use in dose calculations. The core inventory data are presented in Table E.2-1. The core inventory was evaluated at the end of a 525-day fuel cycle and was conservatively based on plant operation at 102 percent of the power level [1,550 megawatts (thermal)] to allow for calibration error. The equilibrium core at the end of a fuel cycle is assumed to consist of fuel assemblies with three different burnups, i.e., approximately 1/3 of the core is subjected to one fuel cycle, 1/3 of the core to two fuel cycles, and 1/3 of the core to three fuel cycles. Minor variations in fuel irradiation times and duration of refueling outages will have a slight impact on the estimated inventory of long-lived isotopes in the core. However, these changes will have an insignificant impact on the radiological consequences of postulated accidents.

E.2.1.2 Source Terms

The atmospheric source terms used in the MACCS2 model were obtained from the latest Level 2 Ginna Station PSA model analysis.

E.2.1.3 Meteorological Data

Ginna Station meteorological data for calendar years 1992, 1993, and 1994, were considered. For these years, consecutive hourly meteorological data (wind speed, wind direction, stability class, and precipitation) were placed in MACCS2 format. Where data blocks were missing in the source files, supplementary information was

Table E.2-1
Ginna Station Core Inventory

Nuclide	Fraction	Nuclide	Fraction
Kr-85	4.98E+05	Tc-99m	6.94E+07
Kr-85m	1.11E+07	Ru-103	6.34E+07
Kr-87	2.13E+07	Ru-105	4.34E+07
Kr-88	3.00E+07	Ru-106	2.25E+07
Xe-131m	4.55E+05	Rh-105	3.98E+07
Xe-133	8.19E+07	Sb-127	4.50E+06
Xe-133m	2.67E+06	Sb-129	1.34E+07
Xe-135	2.17E+07	Te-127	4.45E+06
Xe-135m	1.67E+07	Te-127m	5.81E+05
Xe-138	7.04E+07	Te-129	1.32E+07
I-131	4.16E+07	Te-129m	1.96E+06
I-132	6.03E+07	Te-131m	6.05E+06
I-133	8.53E+07	Te-132	5.93E+07
I-134	9.35E+07	Ba-139	7.62E+07
I-135	7.97E+07	Ba-140	7.34E+07
Rb-86	1.01E+05	La-140	7.87E+07
Cs-134	9.46E+06	La-141	6.95E+07
Cs-136	2.48E+06	La-142	6.72E+07
Cs-137	5.43E+06	Ce-141	6.97E+07
Sr-89	4.07E+07	Ce-143	6.47E+07
Sr-90	3.94E+06	Ce-144	5.43E+07
Sr-91	5.06E+07	Pr-143	6.27E+07
Sr-92	5.47E+07	Nd-147	2.79E+07
Y-90	4.09E+06	Pu-238	1.98E+05
Y-91	5.25E+07	Pu-239	1.61E+04
Y-92	5.49E+07	Pu-240	2.44E+04
Y-93	6.34E+07	Pu-241	5.41E+06
Nb-95	7.13E+07	Np-239	8.45E+08
Zr-95	7.07E+07	Am-241	6.87E+03
Zr-97	7.03E+07	Cm-242	1.48E+06
Mo-99	7.92E+07	Cm-244	2.10E+05

derived from meteorological data obtained from the National Oceanic & Atmospheric Administration (NOAA) from the Rochester Airport, approximately 15 miles west of Ginna Station (Ref. E.2-2). The available NOAA data were insufficient to calculate the stability factors; therefore, these factors were taken from the National Climatic Data Center (Ref. E.2-3). Comparison of the meteorological data for years 1992-1994 were used to demonstrate that the 1992 data set is both a reasonable and conservative data year for use as a representative year for the offsite risk calculation (see Appendix Section E.2.3 for a discussion of the sensitivity case for weather).

E.2.1.4 Emergency Response

To determine the appropriate emergency response assumptions, RG&E reviewed the Ginna Station Nuclear Emergency Response Plan (Ref. E.2-4) and the New York State Radiological Emergency Preparedness Plan (Ref. E.2-5) coupled with local geographic and demographic characteristics. RG&E determined that a 7,200-second evacuation delay time and a 1.8 meters per second evacuation speed were appropriate. RG&E also assumed that 95 percent of the population surrounding the plant would evacuate in an emergency.

E.2.1.5 Population Distribution

For consistency within the site data file, RG&E initially used a projected year-2000 population distribution in the base case analysis and performed a sensitivity analysis on the projected year-2030 population distribution (see Section E.2.3) to determine the effect of increased population on the offsite consequences. The results indicate that the average increase across all the release categories is greater than 20 percent for both dose and economic cost and, therefore, the year 2030 population projection is used in the analysis. This also accounts for increased population near the end of plant life.

To generate the population input data, RG&E used the RSICC code SECPOP90: Sector Population, Land Fraction, and Economic Estimation Program (Ref. E.2-6) as the baseline population distribution for estimating the projected population used in the analysis. The 50-mile region includes the Rochester Metropolitan Area and 13 counties that are completely or partially within the 50-mile radius. SECPOP90 provides the population distribution by sectional rosette centered on the Station and divided into 9 radial intervals out to 50 miles. The rosette consists of 16 directional sectors, the first of which is centered on due north, the second on 22.5 degrees east of north, and so on. The total 1990 population residing in the 50-mile radius region was estimated to be 1,222,212 persons.

SECPOP90 uses year 1990 block level census data to calculate the population within each rosette section. Given that the year 2000 census data at the block level were not available at the time the Level 3 model was prepared, the SECPOP90 population data input file could not be updated by block group before the rosette population matrix was generated. Therefore, the 1990 population numbers were updated by rosette sector after running SECPOP90. RG&E extracted county-level data from the year 2000 census data to develop a weighted average population projection for each rosette section. Changes in population between 1990 and 2000 were calculated under the assumption that increase or decrease in the population for

each rosette section within a given county were the same as those for the county as a whole and that residents are uniformly distributed throughout each county and within the portion of the county contained within a rosette section. Specifically, the 1990 rosette population value was projected for year 2000 by using the ratio of 1990 to 2000 county populations multiplied by the estimated fraction of each county included within the respective rosette section. The county population change factors were applied to the respective rosette section to generate a population distribution for year 2000. The total year 2000 50-mile radius population estimate is 1,260,679 persons.

The 50-mile population data presented in Section 2.7 of the environmental report were calculated using Geographic Information System techniques and year 2000 census data at the census block level. Using this technique the 50-mile population was estimated to be 1.25 million. This comparison demonstrates that the projection method used in the Level 3 model is reasonable.

The year 2000 to year 2030 projection was developed using the same methodology with county population projections obtained from Cornell University for year 2020 (Ref. E.2-7) as input for determining long-term population trends. Yearly growth rates for each county between 1990 to 2000, 2000 to 2020, and 1990 to 2020, were averaged and used to calculate a 30-year growth rate that was applied to the year 2000 population projection, thus creating a year 2030 projection. To account for non-linear population growth, RG&E incorporated a 10 percent population multiplier into the projection. The total 50-mile population projected for year 2030 is estimated to be 1.57 million.

E.2.1.6 Land Fractions

Land fractions represent the portions of the total surface area that are land for each sector, and they are calculated using an algorithm that weights the county-level land fraction data. This is possible because the code contains a county level database with the land fractions for each county and every record in the block level database includes the area of the block and a code to indicate which county in the U.S. the block resides.

RG&E used the values generated by the SECPOP90 code for each rosette section directly in the analysis.

E.2.1.7 Regional Economic Data

Agricultural economic data required for MACCS2 include (Tables E.2-2 and E.2-3):

1) the fraction of land devoted to farming;
2) the farmland property values;
3) the total annual farm sales; and
4) the fraction of farm sales resulting from dairy production.

The SECPOP90 database includes county economic data derived from the year 1990 census and various other government documents dated 1992 to 1994. For

Table E.2-2
MACCS2 Agricultural Data

County	Fraction of Land Devoted to Farming	Fraction of Farm Sales Resulting from Dairy Production	Total Annual Farm Sales ($/hectare)	Farmland Property Values ($/hectare)
Cayuga	0.567551	0.548727	1,133	3,270
Genesee	0.540334	0.446257	1,585	3,324
Livingston	0.487920	0.547562	913	3,354
Monroe	0.244337	0.126225	1,149	5,329
Onondaga	0.294566	0.528181	1,192	3,753
Ontario	0.450810	0.421387	1,036	4,333
Orleans	0.572415	0.110997	1,071	3,279
Oswego	0.168057	0.348064	7,58	3,468
Seneca	0.564681	0.362439	864	3,245
Steuben	0.391511	0.531443	557	2,295
Wayne	0.432344	0.140398	1,590	4,777
Wyoming	0.513566	0.818453	1,707	3,431
Yates	0.484063	0.423508	949	4,654

Table E.2-3
Per Capita Regional Economic Data

County	Farm Wealth Value ($/hectare)	Non-Farm Wealth Value ($/person)
Cayuga	3,270	100,317
Genesee	3,324	108,797
Livingston	3,354	107,174
Monroe	5,329	139,306
Onondaga	3,753	129,254
Ontario	4,333	128,273
Orleans	3,279	90,333
Oswego	3,468	101,637
Seneca	3,245	104,222
Steuben	2,295	129,213
Wayne	4,777	110,002
Wyoming	3,431	88,504
Yates	4,654	93,849

preparation of the Ginna Station Level 3 model the SECPOP90 site input file was manually updated to circa 2000 for the 13 counties within 50 miles of the plant. Therefore, the Level 3 input files contain updated values for each economic region and, hence, for each sector. The agricultural economic data were updated using available data from the 1997 Census of Agriculture (Ref. E.2-8) supplemented by data available through other federal agencies (Ref. E.2-9; Ref. E.2-10; Ref. E.2-11; Ref. E.2-12).

Additional regional economic data factored into the Ginna Station risk analysis includes the value of farm wealth, the fraction of farm wealth in the region due to improvements, and the value of non-farm wealth. The value of farm wealth and non-farm wealth by county are presented in Table E.2-3. The fraction of farm wealth in the region due to improvements was calculated to be 0.11 using the average farm wealth (Table E.2-3) and the average value of farm real estate (Ref. E.2-9).

E.2.1.8 Food Pathway Assumptions

The MACCS2 ingestion model preprocessor, COMIDA2, was used to model the ingestion pathway. Crop season and share data were not used, as the ingestion model uses diet assumptions versus agricultural production to define food intake. However, the COMIDA2 code does require input for waterborne nuclides of concern for the water ingestion model, as well as, food. RG&E identified the four nuclides, Sr-89, Sr-90, Cs-134, and Cs-137, as input to the ingestion model.

Based on the size, Lake Ontario could be treated as an ocean watershed with zero uptake. However, RG&E conservatively treated the Lake as a lake watershed since, unlike an ocean, it is a source of drinking and irrigation water.

E.2.1.9 Deposition Velocities

RG&E calculated a Ginna Station specific deposition velocity value of 0.2 meters per second. The range of values recommended in NUREG/CR-4551 (Ref. E.2-13) is 0.03 to 3.0 with a specific recommendation of 0.3. Considering the surrounding terrain and the formula provided in NUREG/CR-4551, a site-specific value was calculated.

E.2.2 Results

The result of the Level 3 model is a matrix of offsite exposure and offsite property costs associated with a postulated severe accident in each release category. This matrix was combined with the results of the Level 2 model to yield the probabilistic offsite dose and probabilistic offsite property damage resulting from the analyzed plant configuration. Using the bounding base case (year 2030 population projection plus 10 percent and 10 percent source term increase), the offsite exposure risk for Ginna Station is 16.26 person-rem per year. Table E.2-4 provides the baseline exposures associated with each release category. The offsite exposure risk was calculated by multiplying the frequency of the release by the dose.

The bounding base case offsite economic risk is approximately $86,613 per year. Table E.2-4 also provides the base case offsite economic costs associated with each

Table E.2-4
Summary of Offsite Consequences

Release Category	Frequency	Offsite Dose (person-rem)	Offsite Dose Risk	Offsite Economic Costs ($)	Offsite Economic Risk ($)
INTACT	2.38E-05	2.27E+04	0.54	2.82E+07	$671
ISLOCA	2.50E-07	1.76E+07	4.40	2.27E+10	$5,680
LOCI	4.17E-07	3.38E+06	1.41	1.11E+10	$4,630
SGTR WET (Ruptured)	4.63E-06	1.15E+06	5.32	9.43E+09	$43,700
SGTR DRY (Ruptured)	0.00E+00	4.62E+06	0.00	1.82E+10	$0
SGTR Intact	1.32E-06	6.89E+05	0.91	5.62E+09	$7,420
LATE_GLOBAL	1.05E-06[a]	9.39E+05	0.99	9.41E+09	$9,880
LATE_SMALL	1.05E-06[a]	4.51E+05	0.47	2.19E+09	$2,300
TISGTR	2.4E-08	4.72E+06	0.11	1.90E+10	$456
HPRCS	4.13E-07	1.36E+06	0.56	1.06E+10	$4,380
LPRCS	7.95E-09	1.94E+05	0.002	8.07E+08	$6
Total "at power"	3.3E-05	[redacted]	14.71	[redacted]	79,123
Shutdown	6.81E-06	2.26E+05[b]	1.54	1.10E+09	$7,490
Total	3.98E-05	[redacted]	16.26	[redacted]	$86,613

a. Late failures were equally divided between "global" and "small" containment failures
b. Event impacts were conservatively estimated as one half of late small leak containment failures
ARV = atmospheric relief valve
HPRCS = high pressure reactor coolant system break
ISLOCA = interfacing system loss-of-coolant accident
LOCI = loss of containment isolation
LPRCS = low pressure reactor coolant system break
SGTR = steam generator tube rupture
TISGTR = thermally induced steam generator tube rupture

release category. The economic risk for each release category was calculated by multiplying its frequency by the corresponding economic costs.

The final result of a Level 3 evaluation of a SAMA is a value of the cumulative dose expected to be received by offsite individuals and a value of the expected offsite property losses due to severe accidents given the plant configuration under evaluation.

E.2.3 Sensitivity Analysis

Sensitivity analyses were performed to assess variations in certain input factors including weather, population projections, and fission product release.

E.2.3.1 Weather

Data from the years 1992 to 1994 were input into the MACCS2 code for the base case. Dose and cost results for each release category was compared to the average for the three-year period. The results show that the total dose and cost results for the most severe release category (ISLOCA) are within 12 percent of the average. This indicates that the offsite consequences are not highly sensitive to year-to-year variations in weather for the years evaluated. While there is no single year in which all release cases yield the most conservative results, the 1992 data yield results above the three-year average for all releases. Therefore, the 1992 meteorological data are both reasonable and conservative for use in the base case calculation.

E.2.3.2 Population

The initial base case evaluation was performed using year 2000 data, and a sensitivity case was performed using projections to year 2030 plus 10 percent. The results indicate the projected population would increase 25 percent over the year 2000 50-mile population, and the resulting effect on the offsite consequences averaged greater than a 20 percent increase for both offsite dose and economic costs. Given the significance of this increase, the year 2030 population projection plus 10 percent was used in the analysis.

E.2.3.3 Fission Product Release

A sensitivity analysis was performed for a 10 percent increase in fission product release. The core inventory was increased by 10 percent while maintaining the release fractions. While short-term dose effects are proportional to the releases, the impact of long-term dose effects associated with groundshine, resuspension, and ingestion is limited by the use of MACCS2 interdiction triggers, which are based on U.S. Environmental Protection Agency Protective Action Guide dose limits. These triggers impact population relocation, ingestion, and long-term land uses. A 10 percent increase in the source term results in an approximate 7 percent increase in population dose increase.

E.2.3.4 Conclusion

The magnitude of the results presented above indicates that the variation in population and source term should be considered in the offsite consequence calculation. Therefore, in order to bound these uncertainties, RG&E used the year 2030 population projection plus 10 percent and the 10 percent source term increase as input into the MACCS2 model for the base case calculation, as well as the evaluation of each potential modification. This represents a bounding analysis for the purposes of evaluating the offsite consequences for Ginna Station during the period of extended operation.

E.3 SAMA Assessment Sheets

This section includes an evaluation summary for each of the eight SAMAs RG&E evaluated in the cost-benefit analysis. Each summary includes a Ginna Station-specific description of the candidate SAMA, a discussion of the potential benefits, a summary of the evaluation and resulting benefits, and a discussion of the associated implementation costs.

SEVERE ACCIDENT MITIGATION ALTERNATIVE ASSESSMENT SHEET

SAMA No. 1
TITLE: Obtain a skid-mounted 480V diesel generator

Description:
Obtain a skid-mounted 480-volt (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. Rather than relying on station blackout (SBO) mitigation equipment that is alternating current (AC)-independent, an additional skid-mounted diesel capable of carrying SBO mitigation loads could be added to make the SBO mitigation strategy be alternate AC. The size of the diesel is 1000 kilowatts. The diesel would not be safety-related, and would be subject to quality assurance controls per NRC Regulatory Guide 1.155.

SAMA Benefits:
RG&E assumes that all SBO and induced SBO (e.g., fire and flood) sequences would be mitigated with the availability of a skid-mounted diesel generator.

Evaluation:
RG&E assumes that the failure rate for the skid-mounted diesel generator is the same as for the existing diesel generators [i.e., failure to start (FTS) = 1.01E-02 and failure to run (FTR) = 4.22E-02]. Analysts conservatively assume a failure rate of 0.01 for the operators correctly connecting the diesel generator to a safeguards train. The reduction in CDF and LERF were simulated by changing the value of SBO from 1.0 to 0.0623 (i.e., 0.0101 + 0.0422 + 0.01) in both the CDF and large early release frequency (LERF) cutset files. The resulting delta CDF value is 9.87E-06, and the delta LERF is 3.13E-07. The reduction in population dose is estimated to be 7.07 person-rem per year. Note that the contribution of SBO-like sequences to the CDF is 1.05E-05 per year.

Cost of Implementation:
RG&E estimates the cost of the skid-mounted diesel to be approximately $250,000. Additional costs related to training, procedure revision, and documentation are estimated at $100,000, and breakers, cabling, fuel storage, and oil abatement facilities are estimated to cost an additional $50,000, for a total cost of $400,000.

SEVERE ACCIDENT MITIGATION ALTERNATIVE ASSESSMENT SHEET

SAMA No. 2
TITLE: Obtain a third fire water source independent of existing suction source for the motor- and diesel-driven fire pumps

Description:
Obtain a third fire water source independent of existing suction source for the motor- and diesel-driven fire pumps (potentially a portable connection to the discharge canal). This would be used in the event of a total loss of the screenhouse due to a fire or flood or loss of all service water suction due to environmental causes (e.g., frazil ice, seagrass, etc.). The pump should be of comparable size to the current pumps, since the functions would be comparable. The pump could be connected to the existing fire water piping and used for fire suppression or as a source of suction to the auxiliary feedwater pumps. It need not be safety-related, but would be subject to specified quality assurance requirements.

SAMA Benefits:
This SAMA would mitigate the loss of all auxiliary feedwater due to a failure of the service water suction source or a global failure of the screenhouse equipment due to fire or flooding (either in the screenhouse or other areas that will fail the equipment e.g., relay room), or loss of service water suction due to environmental concerns.

Evaluation:
RG&E assumes that the failure rate for the new diesel-driven fire pump is the same as for the existing one (i.e., FTS = 2.50E-03 and FTR = 5.64E-03). Analysts assume a failure rate of 0.1 for the operators correctly connecting the new diesel-driven pump to the SAFW system. Since use of the yard loop is always an option in these cases (i.e., event AXHFDCITYW or AXHFPSAFWX, but not both, is in all of the cutsets), the value of 0.1 assumes dependence with these other events (i.e., if operators fail to use the yard loop, there is an increased probability that they will fail to use the portable diesel pump). Simulate this by changing the value of AXHFDCITYW from 1.39E-02 to 1.39E-03 (i.e., 1.39E-02 * 0.1, since this failure dominates the equipment failures) and AXHFDSAFWX from 2.58E-03 to 2.58E-04 in both the CDF and LERF cutset files.

This new pump could also be used to recover fire events where the existing diesel-driven fire pump fails. Again, assume that the failure rate for the new diesel-driven fire pump is the same as for the existing one (i.e., FTS = 2.50E-03 and FTR = 5.64E-03). In this case, however, the operator failure would be independent of any other human failure and is estimated at 0.01. Simulate this model change by changing the value of events FSDGFPFP01 and FSDGAPFP01 from 5.64E-03 and 2.50E-03, respectively, to 8.82E-05 [i.e., 5.64E-03 * (5.64E-03 + 0.01)] and 3.13E-05 [i.e., 2.50E-03 * (2.50E-03 + 0.01)].

Thus the total expected change is 7.30E-07 for CDF and 2.6E-08 for LERF. The reduction in population dose is estimated to be 0.53 person-rem per year.

SEVERE ACCIDENT MITIGATION ALTERNATIVE ASSESSMENT SHEET

SAMA No. 2 (continued)
TITLE: Obtain a third fire water source independent of existing suction source for the motor- and diesel-driven fire pumps

Cost of Implementation:
RG&E estimates the cost of the diesel motor-driven pump to be $100,000, and estimates the associated procedure revisions, training, and documentation to be $50,000. The breaker and cabling would add an additional $50,000, for total cost of $200,000.

SEVERE ACCIDENT MITIGATION ALTERNATIVE ASSESSMENT SHEET

SAMA No. 3
TITLE: Add a standby charging pump powered from a protected AC source

Description:
This SAMA involves adding a standby charging pump powered from a protected power source and located in the Intermediate or Turbine Building, or SAFW Pump Building. These locations would avoid the failure mechanisms discussed below. It would not have to be safety-related, and so could be powered from Bus 13 or 15 in the Turbine Building. It would have to be mounted so its failure would not adversely affect safety-related equipment. Connections to existing charging lines would have to be safety-related. Bus 13 or 15 would have to be upgraded to achieve the quality assurance requirements for a protected AC source. Significant technical issues to resolve include providing a high volume, primary-grade-quality water source, including the capability to inject borated water.

SAMA Benefits:
This new pump could be used to mitigate fires requiring entry into procedure "Alternate Shutdown for Control Complex Fire" or fires disabling train B, where the A charging pump is out of service or fails to run. It could also be used to mitigate fires in the Charging Pump Room, floods in the Auxiliary Building that fill the basement to a level that will fail all charging, or other failures of all three pumps.

Evaluation:
RG&E assumes that all cutsets that have the following:

a) Charging pump A out of service or failed directly (i.e., not by the initiator or support system failure), or

b) an Auxiliary Building flood that is sufficiently large to fill the basement to a critical height and disable all three charging pumps (event IFAZDABISL), or

c) a Charging Pump Room fire (event FI000CHG),

can be mitigated by using the Intermediate Building charging pump powered from Bus 14. Analysts assume that the Intermediate Building pump would autostart on low flow or pressure (i.e., without operator action). The failure rates for starting and running of the pump are 5.11E-05 and 7.18E-04, respectively (i.e., the same as the existing pumps). RG&E simulated this modification by:

a) Changing the value of CVTMCHPMPA from 7.04E-02 to 5.41E-05 [i.e., 7.04E-02 * (5.11E-05 + 7.18E-04)] in both the CDF and LERF cutset files;

b) Changing the value of CVMPAPCH1A from 5.11E-05 to 2.61E-09 [i.e., 5.11E-05 * 5.11E-05] and CVMPFPCH1A from 7.18E-04 to 5.16E-07 [i.e., 7.18E-04 * 7.18E-04];'

c) Changing the value of IFAZDABISL from 0.1 to 7.69E-5 [i.e., 0.1 * (5.11E-05 + 7.18E-04)] in both the CDF and LERF cutset files. Note that this is a very conservative number in that it does not take into account failures of the support systems for the new pump (i.e., suction source, AC power, etc.); and

SEVERE ACCIDENT MITIGATION ALTERNATIVE ASSESSMENT SHEET

SAMA No. 3 (continued)
TITLE: Add a standby charging pump powered from a protected AC source

d) Changing the value of FI000CHG from 5.4E-03 to 4.15E-06 [i.e., 5.4E-03 * (5.11E-05 + 7.18E-04)] in both the CDF and LERF cutset files. (Note that event FI000CHG does not appear in either file).

The resulting delta CDF is 4.44E-06, and the delta LERF is 2.25E-07. The reduction in population dose is estimated to be 0.40 person-rem per year.

Cost of Implementation:
Cost of hardware modifications is estimated to be greater than $1,000,000 for the pump, piping, valves, engineering analysis, hangers, supports, bus upgrades, cabling, and instrumentation. Procedure revisions, training, and documentation are estimated at $100,000, for a total of $1.1 million.

SEVERE ACCIDENT MITIGATION ALTERNATIVE ASSESSMENT SHEET

SAMA No. 4
TITLE: Modify procedures to allow charging pump B or C to be manually aligned to Bus 14

Description:
This SAMA involves a procedure modification to allow charging pump B or C to be manually realigned 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. An existing spare cable could be routed from Bus 14 to either pump B or C using existing connections.

SAMA Benefits:
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.

Evaluation:
RG&E assumes all cutsets in which charging pump A is out of service or failed directly (i.e., not by the initiator or support system failure), can be mitigated by swinging the B or C pump to Bus 14. Analysts conservatively assume that the failure rate for the operators swapping the pump over is 8.21E-03 (0.1 times the value of CVHFDSUCTN), and the failure rates for starting and running the pump are 5.11E-05 and 7.18E-04, respectively. RG&E simulated this modification by:

a) Changing the value of CVTMCHPMPA from 7.04E-02 to 6.32E-04 [i.e., 7.04E-02 * (0.00821 + 5.11E-05 + 7.18E-04)] in both the CDF and LERF cutset files; and

b) Changing the value of CVMPAPCH1A from 5.11E-05 to 4.59E-07 [i.e., 5.11E-05 * (0.00821 + 5.11E-05 + 7.18E-04)] and CVMPFPCH1A from 7.18E-04 to 6.45E-06 [i.e., 7.18E-04 * (0.00821 + 5.11E-05 + 7.18E-04)] in both the CDF and LERF cutset files.

The resulting delta CDF is 3.61E-06, and the delta LERF is 1.51E-07. The reduction in population dose is 0.28 person-rem per year.

Cost of Implementation:
RG&E estimates the modification to the procedure and associated training costs to be $20,000.

SEVERE ACCIDENT MITIGATION ALTERNATIVE ASSESSMENT SHEET

SAMA No. 5
TITLE: Add redundant check valves in the two RHR injection lines to the RCS

Description:
Install redundant check valves upstream of check valves 853A and 853B. Currently, the position of the 853A and 853B check valve obturators are checked on a refueling outage frequency to ensure the check valves have properly closed. However, if the check valve fails or leaks in between refueling outages, there is no indication of this condition. A spurious safety injection (SI) would cause motor-operated valves (MOV) 852A and 852B to open, allowing the 2250 pounds per square inch (psi) reactor coolant to directly interface with the 600 psi residual heat removal (RHR) piping, potentially resulting in a loss-of-coolant accident (LOCA) in the Auxiliary Building, which could not be isolated. A second check valve in each of these lines would reduce the probability of this event. The new check valves would be Safety Class 1, 2500 psi rated, safety related.

SAMA Benefits:
Adding redundant check valves in series with check valves 853A and 853B would reduce the ISLOCA frequency in the two RHR injection lines.

Evaluation:
This modification would reduce the ISLOCA frequency for those two lines through penetration 111 (although it would not affect the line containing 720 and 721), since the new alignment would require failure of both check valves and the MOV, or both check valves and an inadvertent opening of the MOV, or a spurious SI signal which opens the MOV.

Based on Table 8-4 equation 2 from the PSA final report, the probability of an ISLOCA in a line with two check valves and a normally closed MOV is:

$$\lambda_T = \left\{\left[T^2\left(\lambda_L^2 + 2\lambda_L\lambda_R + \lambda_R^2\right) + \lambda_H T(\lambda_L + \lambda_R) + T(CCF_L\lambda_L + CCF_R\lambda_R)\right] * \left[T(\lambda_{ML} + \lambda_{MR}) + \lambda_{MH}\right]\right\} / PCF$$

However, this equation assumes that the MOV is locked closed and, therefore, not subject to an operator opening the valve, or opening due to an inadvertent SI. Since that is not the case for this line, equation 2 must be modified to account for these two events. In addition, there is the potential for the operators to close the MOV, if it is inadvertently opened or opens on a spurious SI. The probability that operators fail to close the MOV is 0.043. Therefore, the equation becomes:

$$\lambda_T = \left[\left(T^2\left(\lambda_L^2 + 2\lambda_L\lambda_R + \lambda_R^2\right) + \lambda_H T(\lambda_L + \lambda_R) + T(CCF_R\lambda_R + CCF_L\lambda_L)\right) * \left(T(\lambda_{ML} + \lambda_{MR}) + \lambda_{MH} + 0.043\left(\lambda_{MO} + T_{SI}(\lambda_{MS})\right)\right)\right] / PCF$$

Also used is an equation that finds the probability of the physical failure of MOVs 720 and 721:

$$P_{720/721} = T^2\left[\lambda_{ML}^2 + 2\lambda_{ML}\lambda_{MR} + \lambda_{MR}^2\right] + \left[T\lambda_{MH}(\lambda_{ML} + \lambda_{MR})\right] + T\left[CCF_{ML}\lambda_{ML} + CCF_{MR}\lambda_{MR}\right]$$

SEVERE ACCIDENT MITIGATION ALTERNATIVE ASSESSMENT SHEET

SAMA No. 5 (continued)
TITLE: Add redundant check valves in the two RHR injection lines to the RCS

Using the data values from Table 8-5 of the PSA final report gives:

$\lambda_T = 2.836E-07 yr^{-1}$ and $P_{720/721} = 1.189E-05 yr^{-1}$

Now that we know λ_T and $P_{720/721}$ we can find the total CDF from this penetration:

$$CDF_{PENT} = 2 * \lambda_T + P_{720/721} = 2 * 2.836E-07 yr^{-1} + 1.189E-05 yr^{-1} = 1.246E-05 yr^{-1}$$

Multiplying the pipe break probability of 2.29E-02 by CDF_{PENT} results in a total CDF of 2.853E-07 yr^{-1}.

Since a third of the RHR piping that would be exposed to Reactor Coolant System (RCS) pressure is inside containment, it was assumed that the LERF for this penetration would be a third of the CDF, or 9.51E-08. The current CDF contribution from this penetration is 3.56E-07, while the current LERF is 1.19E-07; therefore, the resulting delta CDF is 7.10E-08 per year, and the delta LERF is 2.4E-08 per year. The population dose reduction would be 1.25 person-rem per year.

Cost of Implementation:
RG&E estimates the purchase, installation, analysis, and documentation of this modification is estimated to be at least $1,000,000. There is little room for installation of these check valves, which adds to the complexity of the installation/analysis. It is expected that additional supports would also be required to maintain this piping as Seismic Category I.

SEVERE ACCIDENT MITIGATION ALTERNATIVE ASSESSMENT SHEET

SAMA No. 6
TITLE: Modify the motor-driven AFW pump cooling system to be independent of SW

Description:
Modify the motor-driven AFW pump cooling system to be independent of SW. This would route AFW flow from the discharge of the pumps through a breakdown orifice to self-cool the outboard bearings and lube oil coolers. This would eliminate the dependency on the SW and fire water systems for cooling those components.

SAMA Benefits:
This SAMA would prevent failure of the motor-driven AFW pumps in the event of a loss of all suction to the fire and SW pumps, or a loss of the screenhouse due to fire or flood.

Evaluation:
RG&E assumes all cutsets that involve a loss of all AFW due to a failure of the SW suction source or a global failure of the screenhouse equipment due to fire or flooding [either in the screenhouse or other areas that will fail the equipment (e.g., relay room)] will no longer lead to core damage due to the availability of the motor-driven pumps. Failure rates for the motor-driven AFW pumps to start and run are 7.61E-04 and 6.91E-04, respectively. Analysts simulate this by changing the value of AXHFDCITYW from 1.39E-02 to 2.02E-05 [i.e., 1.39E-02 * (7.61E-04 + 6.91E-04)] and AXHFPSAFWX from 2.58E-03 to 3.75E-06[i.e., 2.58E-03 * (7.61E-04 + 6.91E-04)] in both the CDF and LERF cutset files. The expected delta CDF is estimated to be 7.20E-07and the delta LERF is estimated to be 9.00E-09. The resulting reduction in population dose is estimated to be 0.028 person-rem per year.

Cost of Implementation:
RG&E estimates the cost of this safety-related modification, including parts, construction, analysis, testing, and documentation to be approximately $200,000.

SEVERE ACCIDENT MITIGATION ALTERNATIVE ASSESSMENT SHEET

SAMA No. 7
TITLE: Modify AOV 112C to fail closed and AOV 112B to fail open on loss of instrument air

Description:
This SAMA involves a modification to air-operated valve (AOV) 112C to fail closed and AOV 112 B to fail open on loss of instrument air. This change would allow the refueling water storage tank (RWST) to become the suction source for charging, instead of the volume control tank (VCT), which has limited volume.

SAMA Benefits:
This SAMA would eliminate the need for manual operator actions on low VCT levels (manual actions are required to prevent introducing air into the charging system when the VCT voids).

Evaluation:
This modification would eliminate the need for operators to manually switch over the suction source from the VCT to the RWST (event CVHFDSUCTN). RG&E assumes all cutsets that contain event CVHFDSUCTN can be mitigated by this modification. Analysts simulate this change by setting CVHFDSUCTN to false in both the CDF and LERF cutset files. The resulting reduction in CDF is 7.90E-07 and the reduction in LERF is 5.00E-09. The resulting reduction in population dose is estimated to be 0.025 person-rem per year.

Cost of Implementation:
This change would require swapping the valve operators as well as making post-modification control system adjustments and operating procedure changes. RG&E expects the cost of this modification to be approximately $50,000 for components, design, engineering, analysis, testing, and documentation.

SEVERE ACCIDENT MITIGATION ALTERNATIVE ASSESSMENT SHEET

SAMA No. 8
TITLE: Reconfigure the PORVs so they transfer automatically from instrument air to N2 on low pressure and convert N2 supply line AOV to DC powered motor-operated valve

Description:
This SAMA involves reconfiguration of the power-operated relief valves (PORVs) so they transfer automatically from instrument air to N2 on low pressure and convert the N2 supply line AOV to DC powered motor-operated valve.

SAMA Benefits:
This SAMA would mitigate scenarios where the PORVs are not available due to a loss of instrument air, particularly for feed-and-bleed operations or rapid depressurization of the RCS.

Evaluation:
In order to quantify the effect of this modification, the model was altered to add a flag event to the gates representing failures of instrument air to the PORVs (430 and 431C). The following changes were made:

a) Under RC302A, replace RC321 with SDR1011 AND RC321 RCAAIA0430

b) Under RC310A, replace RC351 with SDR1014 AND RC351 RCAAIA431C

c) Under RC320, replace RC321 with SDR1011

d) Under RC350, replace RC351 with SDR1014

RCAAIA0430 and RCAAIA431C are flag events with a value of 1.0 that can be used to identify sequences where the PORVs fail due to loss of instrument air. Setting RCAAIA0430 and RCAAIA431C equal to 4.76E-03 (the failure rate of the components in the nitrogen system) results in a delta CDF of 6.40E-07 and a delta LERF of 9.5E-08. Note that these values are conservative since the failure rate for the nitrogen system does not include support systems failures (e.g., direct current power) that may fail independently or be failed by the other failures in the cutset. The reduction in population dose is estimated to be 0.14 person-rem per year.

Cost of Implementation:
Implementation of this SAMA would require logic and instrumentation changes as well as replacement of one or two safety-related environmentally qualified solenoid valves. Reanalysis of certain accident scenarios, such as anticipated transients without scram, may also be needed. Extensive changes to procedures, training, and documentation would also be needed. RG&E estimates the cost to be approximately $400,000.

E.4 References

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Ref. E.1-2 R.C. Mecredy (RG&E) letter to the Document Control Desk (NRC), "Generic Letter 88-20, Level 1 Probabilistic Safety Assessment (PSA)." January 15, 1997.

Ref. E.1-3 R.C. Mecredy (RG&E) letter to the Document Control Desk (NRC), "Generic Letter 88-20, Level 2 Probabilistic Safety Assessment." August 30, 1997.

Ref. E.1-4 R.C. Mecredy (RG&E) letter to the Document Control Desk (NRC). "Ginna Station Probabilistic Safety Assessment (PSA), Final Report." Revision 4. February 15, 2002.

Ref. E.2-1 Oak Ridge National Laboratory. RSICC Computer Code Collection, MACCS2, Version 1.12, CCC-652 Code Package. 1997.

Ref. E.2-2 National Oceanic & Atmospheric Administration, National Climatic Data Center (NCDC). "Theoretical Meteorological Year (TMY) Data for Rochester, NY – DATSAV3 Surface." CD-ROM (TMY Data for 1992, 1993, and 1994).

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Ref. E.2-4 RG&E. *Ginna Station Nuclear Emergency Response Plan (NERP).* Revision 20. Rochester, NY. October 19, 2000.

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Ref. E.2-10 U.S. Department of Labor, Bureau of Labor Statistics, "Consumer Price Index – U.S. City Average." ftp://ftp.bls.gov/pub/special.requests/cpi/cpiai.txt. Accessed May 7, 2002.

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Ref. E.2-13 J.L. Spring, et al. *Evaluation of Severe Accident Risks: Quantification of Major Input Parameters MACCS Input.* NUREG/CR-4551, SAND86-1309. Vol. 2, Rev. 1, Part 7. December 1990.