5.0 Environmental Impacts of Postulated Accidents

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

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

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

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

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

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

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

5.1 Postulated Plant Accidents

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

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

5.1.1 Design-Basis Accidents

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

The DBAs are evaluated by both the licensee and the staff to ensure that the plant can withstand normal accidents and abnormal transients and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. A number of these postulated accidents are not expected to occur during the life of the plant but are evaluated to establish the design basis for the preventive and mitigative safety systems of the facility. The acceptance criteria for DBAs are described in 10 CFR Part 50 and 10 CFR Part 100.

The environmental impacts of DBAs are evaluated during the initial licensing process, and the ability of the plant to withstand these accidents is demonstrated to be acceptable before issuance of the operating license (OL). The results of these evaluations are found in this section and in license documentation such as the applicant's final safety analysis report (FSAR), the staff's safety evaluation report (SER), and the Final Environmental Statement (FES). A licensee is required to maintain the acceptable design and performance criteria throughout the life of the plant, including any extended-life operation. The consequences for these events are evaluated for the hypothetical maximally exposed individual; as such, changes in the plant environment will not affect these evaluations. Because of the requirements that continuous acceptability of the consequences and aging management programs be in effect for license renewal, the environmental impacts as calculated for DBAs should not differ significantly from initial licensing assessments over the life of the plant, including the license renewal period. Accordingly, the design of the plant relative to DBAs during the extended period is considered to remain acceptable and the environmental impacts of those accidents were not examined further in the GEIS.

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

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

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

Based on information in the GEIS, the Commission found that

> The NRC staff has concluded that the environmental impacts of design-basis accidents are of small significance for all plants.

In its Environmental Report (ER), Rochester Gas and Electric Corporation (RG&E) stated that "no new information existed for the issues that would invalidate the GEIS conclusions" (RG&E 2002). The staff has not identified any new and significant information during its independent review of the Ginna ER, the staff's site visit, the scoping process, or its evaluation of other available information. Therefore, the staff concludes that there are no impacts related to this issue beyond those discussed in the GEIS.

5.1.2 Severe Accidents

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

Severe accidents initiated by external phenomena such as tornadoes, floods, earthquakes, and fires have not traditionally been discussed in quantitative terms in FESs and were not considered specifically for the Ginna site in the GEIS (NRC 1996). However, in the GEIS, the staff did evaluate existing impact assessments performed by the NRC and by the industry at 44 nuclear plants in the United States and concluded that the risk from beyond-design-basis earthquakes at existing nuclear power plants is SMALL. Additionally, the staff concluded that the risks from other external events are adequately addressed by a generic consideration of internally initiated severe accidents.

Based on information in the GEIS, the Commission found that

> The probability-weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to groundwater, and societal and economic impacts from severe accidents are small for all plants. However, alternatives to mitigate severe accidents must be considered for all plants that have not considered such alternatives.

Therefore, the Commission has designated mitigation of severe accidents as a Category 2 issue in 10 CFR Part 51, Subpart A, Appendix B, Table B-1. This issue, applicable to Ginna, is listed in Table 5-2.

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

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

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

5.2 Severe Accident Mitigation Alternatives

10 CFR 51.53(c)(3)(ii)(L) requires that license renewal applicants consider alternatives to mitigate severe accidents if the staff has not previously evaluated SAMAs for the applicant's plant in an environmental impact statement (EIS) or related supplement or in an environmental assessment. The purpose of this consideration is to ensure that plant changes (i.e., hardware, procedures, and training) with the potential for improving severe accident safety performance are identified and evaluated. SAMAs have not been previously considered for Ginna; therefore, the remainder of Chapter 5 addresses those alternatives.

5.2.1 Introduction

This section presents a summary of the SAMA evaluation for Ginna conducted by RG&E and described in the ER (RG&E 2002) and of the NRC's review of that evaluation. The details of the review are described in the NRC staff evaluation that was prepared by the staff with contract assistance from Information Systems Laboratories, Inc. The entire evaluation is presented in Appendix G.

The SAMA evaluation for Ginna was a four step process. In the first step, RG&E quantified the level of risk associated with potential reactor accidents using the plant-specific probabilistic safety assessment (PSA) and other risk models.

The second step was the examination of the major risk contributors to identify areas where plant improvements might have the greatest chance to reduce risk. Then possible ways of reducing those risks were identified. Common ways of reducing risk are changes to components, systems, procedures, and training. RG&E identified approximately 200 potential SAMAs. Using a set of screening criteria, the number of SAMAs requiring further consideration was reduced to 20. Further refinement and review of these 20 SAMAs eliminated 12 from further consideration.

In the third step, the benefits and costs for the remaining eight candidate SAMAs were estimated. Estimates were made of how much each proposed SAMA could reduce risk. Those estimates were developed in terms of dollars in accordance with NRC guidance for performing regulatory analyses (NRC 1997). The costs of implementing the proposed SAMAs were also estimated.

Finally in the fourth step, the costs and benefits of each of the eight final SAMAs were compared to determine whether the SAMA was cost-beneficial, meaning the benefits of the SAMA were greater than the costs (a positive cost-benefit). In the final analysis, two of these SAMAs were determined to be cost-beneficial for Ginna.

Each of these four steps is discussed in more detail in the sections that follow.

5.2.2 Estimate of Risk for Ginna

RG&E submitted an assessment of SAMAs for Ginna as part of the ER (RG&E 2002) and provided a revised assessment in response to staff information requests (RG&E 2003). This assessment was based on the most recent Ginna PSA (including the Level 1 and 2 analyses), a plant-specific offsite consequence analysis performed using the MELCOR Accident Consequence Code System 2 (MACCS2) (essentially a Level 3 PSA model), and the Ginna Individual Plant Examination of External Events (IPEEE) (RG&E 1997a, 1998a, 1998b, 1998c).

The most recent PSA is a refinement of the plant-specific PSA presented in the Ginna Individual Plant Examination (IPE) (RG&E 1994, 1997b, 1997c). The baseline core damage frequency (CDF) for Ginna is approximately 4.0×10^{-5} per year, based on internally-initiated events at power and at shutdown, and fire and internal flooding events at power. RG&E did not include the contribution to CDF from seismic events in these estimates. RG&E concluded that the existing IPEEE and Seismic Qualification Utility Group (SQUG) evaluations had adequately identified potential plant improvements to address seismic events. The breakdown of CDF by initiating event/accident class is summarized in Table 5-3. Fires, internal floods, shutdown events, and steam generator tube ruptures are the dominant contributors to the CDF.

Table 5-3. Core Damage Frequency for R.E. Ginna Nuclear Power Plant (Revision 4.2 of PSA)

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

RG&E estimated the dose from all postulated accidents to the population within 80 km (50 mi) of the Ginna site to be approximately 0.163 person-Sv (16.300 person-rem). The breakdown of the population dose by containment release mode is summarized in Table 5-4. Bypass events

(SGTR and interfacing system LOCA) and late containment failures dominate the population dose.

Table 5-4. Breakdown of Population Dose by Containment Release Mode

Containment Release Mode	Population Dose Person-Sv Per Year	(Person-Rem Per Year)	Percent Contribution
SGTR[(a)]	0.063	6.300	39
ISLOCAs	0.044	4.400	27
Early containment failure	0.020	2.000	12
Late containment failure[(b)]	0.030	3.000	19
No containment failure	0.006	0.600	3
Total	**0.163**	**16.300**	**100**

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

The staff has reviewed RG&E's data and evaluation methods and concludes that the quality of the risk analyses is adequate to support an assessment of the risk reduction potential for the candidate SAMAs. Accordingly, the staff based its assessment of offsite risk on the CDF and offsite doses provided by RG&E.

5.2.3 Potential Design Improvements

Once the most risk significant parts of the plant design and operation were identified, RG&E searched for ways to reduce those risks. To identify potential plant improvements, RG&E reviewed improvements identified in the Ginna IPE and IPEEE processes, SAMA analyses submitted for other nuclear power plants, and NRC and industry documents discussing potential plant improvements. RG&E also reviewed the importance measures and dominant cutsets of the Ginna PSA and considered insights provided by Ginna plant staff. RG&E identified approximately 200 potential risk-reducing improvements to plant components, systems, procedures, and training (SAMAs).

All but 20 of these SAMAs were removed from further consideration because (1) the SAMA was not applicable at Ginna due to design differences, (2) the SAMA would involve major plant design and/or structural changes that would clearly be well in excess of the maximum attainable benefit, or (3) the SAMA would provide only minimal risk reduction.

These 20 candidate SAMAs were further defined and then reviewed based on the following considerations: (1) ability to implement the change at Ginna (i.e., assessment of design challenges or physical limitations), (2) the risk reduction that would realistically be achieved, and (3) whether implementation of the change would increase vulnerabilities in other areas.

Using this evaluation process, all but eight of the candidate SAMAs were removed from further consideration.
The staff reviewed the screening methods used by RG&E and their results and concluded that they were systematic and comprehensive.

5.2.4 Evaluation of Risk Reduction Potential and Cost of Design Improvements

RG&E calculated the potential risk reduction for the remaining eight SAMAs. The potential benefits were developed by adding the estimated present dollar value of the averted public exposure, offsite property damage, occupational exposure, and onsite costs associated with each SAMA. RG&E estimated the costs of implementing the eight remaining SAMAs through application of engineering judgement and site-specific cost estimates.

The staff reviewed RG&E's calculations of the potential risk reduction and concluded that they are reasonable and conservative. Therefore, the staff based its estimates of averted risk for the SAMAs on RG&E's risk reduction estimates. The staff reviewed the cost estimates and concluded that they are sufficient and appropriate for use in the SAMA evaluation.

5.2.5 Cost-Benefit Comparison

Based on the more detailed evaluations of potential risk reduction and cost discussed above, RG&E determined that two of the eight remaining SAMAs were cost beneficial. RG&E performed additional analyses to determine the impact of certain parameter choices such as the discount rate on the calculations. RG&E also evaluated the impact on SAMA results if the 95^{th}- percentile values of the CDF were used in the cost-benefit analysis instead of the best-estimate CDF values. These analyses did not result in identifying any additional cost-beneficial SAMAs. Therefore, RG&E finally concluded that there were two cost-beneficial SAMAs.

The two SAMAs considered to be potentially cost beneficial include (1) obtaining a skid-mounted, 480-V diesel generator that could be directly connected to one train of the safeguards buses in the event of a failure of the two existing diesel generators; and (2) modifying procedures to allow certain charging pumps to be manually aligned to an alternate power source in the event of a control complex fire, or a fire that disables safeguards train B when the train A charging pump is out of service or fails to run.

The staff reviewed calculation methods and logic arguments used by RG&E in the final cost-benefit comparisons and agreed with their conclusion that two of the original approximately 200 SAMAs are cost beneficial.

5.2.6 Conclusions

The staff reviewed the SAMA analysis provided by RG&E and concluded that the methods used and the implementation of those methods were sound. The treatment of SAMA benefits and costs, the generally large negative net benefits, and the inherently small baseline risks support the general conclusion that the SAMA evaluations performed by RG&E are reasonable and sufficient for the license renewal submittal.

Based on its review of the RG&E SAMA analysis, the staff concludes that two of the candidate SAMAs are cost-beneficial. This is based on conservative treatment of costs and benefits. This conclusion is consistent with the low residual level of risk indicated in the Ginna PSA and the fact that Ginna has already implemented many plant improvements identified from the IPE and IPEEE process. Although two SAMA candidates appear to be cost beneficial, they do not relate to adequately managing the effects of aging during the period of extended operation. Therefore, they need not be implemented as part of the license renewal pursuant to 10 CFR Part 54. RG&E stated that it will consider implementation of these SAMAs through its current plant change process.

5.3 References

10 CFR Part 50. Code of Federal Regulations, Title 10, *Energy,* Part 50, “Domestic Licensing of Production and Utilization Facilities.”

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

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

10 CFR Part 100. Code of Federal Regulations, Title 10, *Energy*, Part 100, “Reactor Site Criteria.”

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U.S. Nuclear Regulatory Commission (NRC). 1999. *Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Main Report,* "Section 6.3 – Transportation, Table 9.1 Summary of findings on NEPA issues for license renewal of nuclear power plants, Final Report." NUREG-1437, Volume 1, Addendum 1, Washington, D.C.