NUREG-0979
Supplement No. 3

Safety Evaluation Report
related to the final design approval of the GESSAR II BWR/6 Nuclear Island Design

Docket No. 50-447

General Electric Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

January 1985

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ABSTRACT

Supplement 3 to the Safety Evaluation Report (SER) for the application filed by General Electric Company for the final design approval for the GE BWR/6 nuclear island design has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. This report supplements the GESSAR II SER (NUREG-0979), issued in April 1983, summarizing the results of the staff's safety review of the GESSAR II BWR/6 nuclear island design. Subject to favorable resolution of the items discussed in this supplement, the staff concludes that the GESSAR II design satisfactorily addresses the severe-accident concerns described in draft NUREG-1070.

TABLE OF CONTENTS

TABLES

ACRONYMS AND INITIALISMS

ADS	automatic depressurization system
ATWS	anticipated transient without scram
BNL	Brookhaven National Laboratory
BOP	balance of plant
BWR	boiling-water reactor
CP	construction permit
CST	condensate storage tank
DGCM	diesel generator common mode
DGS	diesel generator structure
EPA	effective peak acceleration
FDA	Final Design Approval
GE	General Electric Co.
GESSAR	General Electric Standard Safety Analysis Report
GSI	Generic Safety Issue
HCU	hydraulic control unit
HPCS	high-pressure core spray
LOOP	loss of offsite power
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
LWR	light-water reactor
MMI	modified Mercalli intensity
OL	operating license
PDA	Preliminary Design Approval
PRA	probabilistic risk assessment
RCIC	reactor core isolation cooling
RDG	recovery of diesel generator
RHR	residual heat removal
RPV	reactor pressure vessel
SLC	standby liquid control
SSE	safe shutdown earthquake
SSER	Supplement to the Safety Evaluation Report
USI	Unresolved Safety Issue

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

On April 8, 1983, the Nuclear Regulatory Commission staff (staff) issued a Safety Evaluation Report (NUREG-0979) regarding the application by General Electric Company (GE) for a Final Design Approval (FDA) for GE's BWR/6 nuclear island design (GE Standard Safety Analysis Report, GESSAR II). In July 1984, Supplement 1 to the Safety Evaluation Report (SSER 1) was issued for GESSAR II, and on July 27, 1983, the Office of Nuclear Reactor Regulation issued FDA-1 for GE's BWR/6 nuclear island design. This approval allows the GESSAR II design to be referenced in operating license (OL) applications for plants that referenced the GESSAR-238 nuclear island design Preliminary Design Approval (PDA-1) at the construction permit (CP) stage of the licensing process. FDA-1 is the first Final Design Approval issued by the Office of Nuclear Reactor Regulation for a standard nuclear plant design or major portion thereof.

SSER 2 was issued in October 1984. It provided information related to the staff review of GESSAR II for severe-accident concerns. The present supplement (SSER 3) provides more-recent information regarding resolution or update of the open and confirmatory items identified in SSER 2.

Each of the following sections and appendices of this supplement is numbered the same as the SER section or appendix that is being updated, and the discussions are supplementary to and not in lieu of those in the SER unless otherwise noted. Accordingly, Appendix A is a continuation of the chronology of the safety review. Appendix B is an updated list of references. Appendix E lists the principal contributors to this supplement.

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1.8 Summary of Outstanding Issues

SSER 2 listed 10 outstanding issues that were either under staff review or awaiting information. During the course of the staff review of the GE probabilistic risk assessment (PRA) of the BWR/6 nuclear island described in GESSAR II, additional issues have been identified that remain unresolved. The issues relate to severe-accident concerns, and their unresolved status is attributable to the fact that (1) the staff needs to review existing information, (2) GE needs to supply additional information, or (3) the staff needs to consider the issue further. For those items discussed in this supplement, the relevant section is indicated in parentheses following the item.

Issue	Status
Containment structural analysis	Under review
Hydrogen control measures, USI A-48	Under review
Potential design modification	Under review
Safety parameter display system	Awaiting information
Containment emergency sump reliability, USI A-43	Under review
Safety implications of control systems, USI A-47	Under review
Loads, load confirmations, stress limits, GSI B-6	Under review
Passive mechanical failures, GSI B-58	Under review
Beyond-design-basis accidents in spent fuel pool, GSI 82	Under review
External events	
Relay chatter	Under consideration
Consequence analysis	Under review
Pool bypass sequences	Under review

1.9 Confirmatory Issues

SSER 2 listed six confirmatory issues that were either under staff review or awaiting information. The tabulation below shows the current status of each of the six issues as well as the new confirmatory issues.

Issue	Status
Factor of safety against sliding	Awaiting information
Software engineering manual	Awaiting information
Optical isolators	Awaiting information
Combustible gas control	Under review
Station blackout, USI A-44	Under review
Shutdown decay heat removal, USI A-45	Under review

1.10 Interface Information

GESSAR II describes a standard BWR/6 nuclear island design. Consequently, GESSAR II does not describe an entire facility, but is limited in scope to those design and safety features associated with the nuclear island design. The design scope is defined in the SER and GESSAR II Section 1.2. GESSAR II also defines interface requirements that must be imposed on the reference plant (individual applicant referencing GESSAR II) so that the balance of plant (BOP) will provide compatible design features that will ensure the applicability, functional performance, and safe operation of the GESSAR II systems.

A summary of the interface requirements resulting from the staff review of the GESSAR II for severe-accident concerns is presented in Table 1.2 of this supplement. For a complete list of interface requirements, see GESSAR II (Section 1.9) and Table 1.2 of the SER and its supplements.

Table 1.2 Interface items

SER Section	Item
15.6.2.3(1.5)	Critical-component-and-structure list
15.6.2.3(1.5)	Site-specific hazard function analysis
15.6.2.3(1.5)	Seismic analysis interface assumptions

15 TRANSIENT AND ACCIDENT ANALYSIS

15.6 Severe Accidents

15.6.2 Major Review Results and Conclusions From PRA Review

15.6.2.3 External Events

(1) Seismic

The "GESSAR II Seismic Event Analysis" (GE, Sept. 1983) is basically similar in methodology to other seismic risk analyses which have been performed on light-water reactors (LWRs). The purpose of this review is to provide a perspective of the seismic risk and the root causes of this contribution. Design singularities which potentially dominate the seismic risk profile would be identified early in the review process, should they exist. Design modifications would then be considered to reduce the contribution of such singularities to the seismic risk. General Electric Co. (GE) used fault-tree/event-tree methodology in evaluating the seismically induced accident sequences to estimate the frequency of core melt. The same approach to consequence modeling that was used for internal events, was also utilized here to address seismic risk. The GE analysis estimated that the mean seismically induced core-damage frequency attributable to seismic events was 4×10^{-7} per year. GE also estimated that seismic events had a minimal contribution to offsite consequences and resulted in only a 5% increase to the total plant site risk.

The staff and its consultant, Brookhaven National Laboratory (BNL), reviewed the "GESSAR II Seismic Event Analysis." The review concentrated in three major areas of the analysis: the site hazard function, component and structural fragility analysis, and plant systems analysis. The GESSAR II review was the first instance of a seismic risk assessment performed for a standard plant. This presented a number of difficulties for both the GE evaluation and staff review, since the GESSAR II plant is neither sited nor constructed. Site-specific and structural/equipment-specific information needed for the risk assessment was not available. Therefore, representative values had to be assumed in the areas of site hazard characteristics and equipment/structure response, in order to calculate the expected frequency of seismic accident sequences for a typical plant and site.

Much of the staff's review effort centered about assessing the appropriateness of these postulated parameters. In general, the staff and its consultants found the seismic probabilistic risk assessment (PRA) analysis for GESSAR II to be considerably less detailed than analyses performed for plants with specific sites. A number of specific deficiencies were identified in the areas of site hazard function, component/structural fragility values, and seismic system modeling. These findings are not intended as a criticism of the GESSAR II seismic design, which is generally believed to be an improvement over plants of previous designs, but rather, as a determination that the seismic risk study did not model well the risk likely to be contributed by seismic initiators for an actual GESSAR II plant at a typical site.

Because of the deficiencies identified in the GESSAR II seismic study, the staff was not confident about the accident-sequence quantifications. Since these results would be considered in judging the safety acceptability of the GESSAR II design and the benefit of possible plant modifications, the staff conducted a more representative evaluation. The staff and its consultants prepared a limited assessment of the seismic accident sequences, including sensitivity studies, to better reflect the seismic impact for the GESSAR II design. A limited assessment based on published hazard curves and varying fragility assumptions indicates a potential point estimate of seismically induced core-melt contribution of 7×10^{-7} to 1×10^{-3} per year, making seismically induced accidents a potentially significant contributor to the frequency of core damage. This large range was due primarily to the hazard functions and different component and structural capacities. The staff has not yet completed its consequence assessments for seismically induced events. Two new suppression pool bypass sequences have been identified that were not included in GE's seismic evaluation report (Sept. 1983). Although these sequences are expected to have a low probability, their consequences could be large because of the absence of pool scrubbing. The staff is assessing these new sequences and will report on them in a future supplement to the SER. The present supplement (SSER 3) presents the staff's findings in the three major review areas previously identified. Section 15.6.2.3(1.1), which follows, discusses the staff's findings in the area of site hazard function; Section 15.6.2.3(1.2) discusses the staff's findings on component and structural fragility analysis; and Section 15.6.2.3(1.3) discusses the staff's findings on the GESSAR II seismic systems analysis. The staff's reassessment of seismic sequence quantification is presented in Section 15.6.2.3(1.4), and the staff's findings and conclusions appear in Section 15.6.2.3(1.5).

(1.1) Site Hazard Analysis

The basic approach GE used for assessing the core-damage frequency and offsite risk is similar to that used in other PRAs for LWRs. It consists of:

(1) establishing a family of seismic hazard curves

(2) determining seismic fragility of structures, critical components, and equipment

(3) modeling the systems impact of seismic events and constructing appropriate fault trees/event trees

(4) assessing core-damage frequency, and resulting consequences

The first step in assessing the seismic risk for the GESSAR II design was to develop an appropriate site hazard function. This function is an assessment of the seismic activity for a particular site, and is the cumulative probability for exceeding various levels of ground acceleration, that result from seismic events.

Since GESSAR II is a standard plant design, developed to be suitable for a number of locations satisfying the safe shutdown earthquake (SSE) criteria of 0.3 g, it was not possible to develop a site-specific hazard function. Rather, an attempt was made to develop and present a representative hazard curve which would bound the seismic hazard for a range of potential GESSAR II siting locations.

For developing a hazard curve, GE adopted what was called a "pseudo-demand" approach. In this approach, GE resorted to using existing hazard curves available in the literature.* An envelope of these curves would be assumed to represent a median-centered, upper-bound, seismic-hazard curve for most of the potential GESSAR II sites.

Using this approach, GE defined the GESSAR II seismic hazard in terms of effective peak acceleration (EPA). It has been argued (Commonwealth Edison Co., Sept. 1981) that damage-effective ground accelerations which are input to plant structure are limited, and that an upper-bound EPA of no more than 0.8 g can be associated with a modified Mercalli intensity (MMI) of IX. Assuming that MMI IX is the largest possible intensity for the GESSAR II site, GE has chosen an upper-bound cutoff of 0.95 g (20% greater than 0.8 g) and assigned this acceleration a frequency of exceedence of 10^{-8}.

In the second stage of the GE submittal (GE, Dec. 1983), the probability distribution of the frequency of ground motion was provided. In "GESSAR II Seismic Event Uncertainty Analysis" (GE, Dec. 1983), GE used a study by Okrent (1975) to assess the uncertainty associated with the GESSAR II seismic hazard curve. Okrent's study provided the results of a survey of seven experts regarding the frequency of exceeding different ground motion at eleven nuclear power plants. For each plant site, GE used the estimates given by the seven experts to obtain a mean and a standard deviation for each acceleration level. To account for the expert-to-expert uncertainty in seismic exceedance frequency, coefficients of variation (the ratio of standard deviation to the mean) were calculated. To account for differences in the uncertainty between the eleven plant sites, the coefficient of variations for each plant was then used to calculate an average value for the eleven sites and its standard deviation. GE then calculated the 95th percentile of overall variability and used these values to calculate the uncertainty associated with the GESSAR II seismic hazard curve.

The GESSAR II seismic hazard curve presented in the risk assessment was generally represented as a bounding curve applicable for assessing the seismic risk for particular sites. Although GE does not claim the curve would bound all sites, it is presented as the appropriate hazard function for most sites. The staff and its contractors have a number of reservations regarding the approach GE used to generate the curve. The GE hazard curve, which was shown to bound several published curves for sites in the eastern United States, was also shown to be less than others appearing in the literature. Details of the staff findings are found in the BNL review of the GESSAR II Seismic Safety Analysis (Sept. 1984) and the staff's "Review of Seismic Hazard and Fragility in the GESSAR II Probabilistic Risk Assessment" (Oct. 1984).

There are a number of aspects of the study which lack sufficient documentation for support such as:

(1) the upper bound cutoff to EPA of 0.95 g

(2) the potential for site-specific soil behavior impacting plant response

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*Commonwealth Edison Co., Sept. 1981; Power Authority of the State of New York, 1982; Philadelphia Electric Co., Apr. 1983; and Cornell, 1968.

(3) assigning a mean frequency of 10^{-8} to EPA of 0.95 g

(4) the approach GE used to estimate the uncertainty associated with the hazard curve

(5) presenting a single seismic hazard curve as a representative of the majority of the potential GESSAR II sites

Although the GE hazard curve can be considered appropriate for many sites in the eastern United States, the staff cannot conclude that the hazard curve is appropriate for the majority of sites in the U.S. For these reasons, the staff will require that a site-specific hazard analysis be conducted by any utility applicant that references the GESSAR II design, demonstrating that the proposed site hazard curve is bounded by the GESSAR II curve, appropriately considering uncertainties. A site-specific seismic hazard assessment will be required of the utility applicant at the construction permit (CP) stage if the site hazard curve is not bounded by the GESSAR II hazard curve. The results of this assessment will be used to determine the acceptability of the specific site. As part of its evaluation of the GESSAR II seismic risk assessment, staff/BNL performed a limited sensitivity study. The impact of alternate hazard functions was studied, and is discussed in Section 15.6.2.3(1.4) that follows.

(1.2) Seismic Fragility Analysis

To assess the response of the GESSAR II plant to seismic events, it was necessary to model the likelihood of structural and component failure from seismically induced accelerations. Structural and component failure is represented by log normally distributed fragility curves.

Since GESSAR II is not a constructed plant, actual structures and components could not be cited to support various fragility values incorporated into the PRA analysis. Instead, GE assembled representative fragility values for its analysis. The staff/BNL found that the fragility values applied in the GESSAR II analysis do not appear to be reasonably representative to cover all potential GESSAR II sites.

The approach used by GE in developing the structural and component fragility data for GESSAR II is similar to the methodology used in seismic PRAs for the Zion, Indian Point, Limerick, and Millstone plants. A log-normal probability model for capacity was assumed, and the median peak ground acceleration capacities based on extrapolation of the original design analysis were estimated. Seismic test data, limited analytical calculations, professional judgment, and the results from previous PRA studies were utilized to arrive at final values.

The variability for fragility values was not calculated directly. Rather, combined coefficients of variation were assumed, and it was stated that the values were conservative in comparison with values documented elsewhere. GE provided sensitivity analysis to demonstrate that larger variations would have little impact on core-melt frequency.

The staff/BNL concluded that the assessment of variability was not conservative relative to values used in the past for site-specific PRAs. Similarly, in the supporting uncertainty analysis, the total variabilities were separated into randomness and uncertainty components based only on professional judgment and not on specific calculations. Additionally, the staff has reservations about the completeness of the identified important structures, components, and their associated failure modes. The scope of the GESSAR II design includes only the nuclear island. Table 15.1 lists structures, components, and failure modes not considered in the GE analysis, which should be included in the site-specific analysis that is to be performed by utility applicants who reference GESSAR II at the CP stage.

The calculations for the median capacities of structures and components provided by GE were reviewed. The purpose of this review was to gain an understanding of how the median capacities were determined and to assess the reasonableness of the methods and assumptions made. Problems were identified with some of the values chosen by GE for incorporation into the seismic risk assessment. The more significant structural problem areas are listed below.

Structure	Identified problem with fragility assessment
Containment anchor bolts	Inelastic energy absorption not correctly modeled
Auxiliary building foundation	Effects of sliding and rocking questioned
Reactor pressure vessel skirt connection	Anchor bolt energy absorption not correctly modeled
Shield building	Flexure failure mode should be based on ultimate capacity and soil failures not considered
Containment shell above suppression pool	Fragility overestimated
Seismic Category I structures	Deficiencies in analysis with respect to soil failure modes

Similarly, for various components, potential problems with the GE analysis were identified. The more significant are listed below.

Component	Identified problem with fragility assessment
Piping	Pipe support capacity not addressed
Control rod drive guide tubes	Fuel rod deflection not modeled
Residual heat removal heat exchanger	Medium capacity higher than computed at another plant, basis not provided
Hydraulic control units	Several errors in analysis regarding use of response spectral, and use of results from other PRAs
Standby liquid control tank	Incorrect use of 1.4 factor for response spectrum considerations

Component	Identified problem with fragility assessment
Electrical power equipment	Floor spectral accelerations incorrectly used to describe capacity

The staff/BNL are also concerned with relay chatter as a component failure mode. A recent assessment (Lawrence Livermore National Laboratory, Apr. 1984) suggested that the chattering of mechanical relays during a seismic event could result in the tripping of numerous systems and components required to respond successfully to the seismic event. In this situation, considerable human action would be required to recover needed systems for preventing or mitigating core damage that could result from beyond design bases seismic events. The staff has identified its concerns to GE and will be reviewing this issue to determine the potential impact from relay chatter. The staff will report on this issue in a later supplement to this SER.

For the reasons summarized above and detailed in the staff/BNL evaluation, the median capacity values developed by GE may not be realistic for a specific GESSAR II plant. The staff/BNL considered two cases of alternate median capacity values and associated total variability (logarithmic standard deviations) as being more representative than those developed by GE. These values are based on values obtained from previous PRA studies (i.e., Zion, Indian Point, Limerick, and Millstone). Table 15.2 gives the two cases of alternate values along with the values reported by GE (Sept. 1983; Dec. 1983).

Case 1 parameter values for equipment are essentially the same as the values used in the Limerick PRA. The fragility analysis performed by GE did not provide assurance that equipment in GESSAR II is significantly stronger than similiar equipment installed in other recent BWR plants. On this basis, the capacities of the equipment at Limerick (Philadelphia Electric Co., Apr. 1983) were used as the Case 1 values. Exceptions were the values for the service water system, relay chatter, condensate storage tanks, and the diesel generator structural failure mode. For these components, values from other PRA studies which were considered more appropriate were used, because of the differences in the Limerick design.

The building capacities were assumed by the staff/BNL to be 1.50 g median with a total logarithmic standard deviation of 0.50. This capacity level represents potential structure sliding and failure of interconnecting piping, and is based in part on the results of the Millstone PRA. Because the drywell, containment, and shield building are all attached to a common foundation mat, it was assumed in the analysis that the responses for these structures are perfectly dependent. Relay chatter was added as a component failure mode, with an assumed capacity of 0.6 g.

The Case 2 fragility parameter values represent small variations of the Case 1 parameter values. On the basis of other PRA results, some of the values were changed from Case 1. In general, the capacities were lowered. In addition, the human-error probabilities associated with failure to reset the relay were considered to range from 50% to 10%. The staff used these above considerations to quantify the seismic core-melt contributor, the results of which are presented in Section 15.6.2.3(1.4).

The depth of fragility analysis performed in the GESSAR II seismic event analysis reports (GE, Sept. 1983; GE, Dec. 1983) does not provide confidence that the results of the fragility parameters conservatively represent all potential GESSAR II sites. Not all structures, components, and failure modes have been included in the analysis (see Table 15.1). Soft soil conditions may lead to rocking or sliding structure modes which could produce failure of interconnecting piping. Also, structure frequency, inelastic energy absorption, and damping factors used by GE may be nonconservative for rock sites.

The lack of detail and supporting data in the fragility calculations leads to the concern that the seismic margin may be lower than that asserted by GE for specific site applications.

Therefore, the staff proposed component and structural fragility values which it believes to be more representative. The staff/BNL used these values in calculating the GESSAR II seismic accident sequences (discussed in Section 15.6.2.3(1.3).

(1.3) <u>Systems Analysis</u>

Event-tree/fault-tree methods were applied in the GESSAR II study to analyze the seismically induced accident sequences and to calculate the resultant core-damage frequency. This is the basic approach utilized in other PRA studies. Seismic system fault trees were developed for eight different systems: the high-pressure core spray (HPCS), reactor-core isolation cooling (RCIC), low-pressure coolant injection (LPCI), low-pressure core spray (LPCS), automatic depressurization (ADS), residual heat removal (RHR), standy liquid control (SLC), and the scram system.

For development of the seismic system fault trees, random failures were represented by single developed events, where no details of different component contributions are explicitly included in the fault-tree evaluation. In addition to the random failures of safety-related systems, the seismic system fault trees also contain seismically induced faults, treated in an explicit manner. These faults vary from electrical component failures (such as loss of power from a division) to mechanical component failures (such as the failure of RHR heat exchangers).

GE developed five functional event trees to specifically model the seismic accident sequences. Two of these trees can be further characterized as loss-of-offsite power (LOOP) event trees and anticipated transient without scram (ATWS) event trees. The other three event trees address mainly building-related failures. Details of these trees can be found in the BNL GESSAR II seismic evaluation.

On the basis of the accident sequences identified by the five functional event trees, Boolean expressions were reported in the GESSAR II analysis. These expressions were reduced further into minimal cutsets. By using conventional techniques, the hazard function and component fragility distributions were combined to produce the core-damage frequency of the accident sequences.

Two sets of results were presented in the two GESSAR II seismic event analyses (GE, Sept. 1983; GE, Dec. 1983): the first document contains only a point-estimate evaluation and the total core-damage frequency was found to be 4×10^{-7}.

The second document, which contains the seismic uncertainty analysis, calculated the core-damage frequency to be 6×10^{-7} with the 5th and 95th percentile estimates of 1×10^{-8} and 2×10^{-6}, respectively. Uncertainties introduced in the process of expert sampling were considered in the evaluation of the GE hazard function. Coefficients of randomness and uncertainty were used to describe the dispersion in the GE fragility curves.

The GESSAR II seismic event analysis was performed using event-tree/fault-tree methodology. The inclusion of random failures into the seismic event fault trees was an improvement over some earlier PRAs. However, in the course of its evaluation the staff identified a number of potential deficiencies in the GESSAR II analyses. These are discussed briefly below.

In the "GESSAR II Seismic Event Analysis" (Sept. 1983) twelve components were considered to be of significance, and were included in the critical-component list. The critical-component list was judged to be incomplete as there are components identified as important in other PRAs that are not included in the GE list. The staff/BNL prepared a list to include the items they considered important and that list was used for the staff's systems assessment (see Table 15.1).

There are several instances in the GESSAR II system fault-tree analysis where similar components functionally in parallel are treated as independent. It has been found in other PRA studies that an appropriate modeling of this dependence is to conservatively assume that the probability of failure of all parallel components that are similar corresponds to the failure probability of the weakest component.

This issue of dependence also applies for similar power divisions. GE modeled the seismically induced electric power failures by defining an event for each of the three electric power divisions. This event is intended to include all the electric-power-related failures that may contribute to the unavailability of that division. The staff believes that this approach is non-conservative in light of the multiple dissimilar electric components that are susceptible to earthquake-related failures.

Examination of the GESSAR II seismic ATWS event tree and the GESSAR II internal ATWS event tree indicates that the limit-high-level function, LH, is absent from the seismic event tree. This function is to describe operator action in controlling the injection systems so that the water level is below the main steamlines should the automatic high-water-level trip (level 8) fail. This level-control function is a part of the ATWS emergency procedure guidelines. Given the fact that the operator has already successfully inhibited the automatic depressurization function, failure to control water level would also result in core damage. Thus, by not considering the LH function in the seismic ATWS functional event tree, the GESSAR analysis has underestimated the ATWS-core-damage-frequency contribution during seismic events.

In the staff/BNL analysis of the GESSAR II design, certain modifications were made to the system models utilized by GE. In many instances, new system fault trees and functional event trees were developed. BNL developed nine seismic system fault trees. These trees contain events that have not been considered in the GE analysis. The ac electric-power fault tree is such a fault tree.

HPCS and the RCIC system fault trees were developed utilizing the additional components identified earlier. Both of these trees consider the random failures as developed events. In addition, ac power is also included as a developed event. Failure of the pumps or the emergency service water system is assumed to result in the disabling of the high-pressure system. The failure of the water suction to these high-pressure systems is modeled under an "and" gate. A seismically related failure of the condensate storage tank (CST) would result in the loss of the first water source and it is assumed in the staff/BNL model that it would also cause the low CST tank level instrumentation to fail, thus disabling the automatic transfer function from the CST to the suppression pool.

The loss of suppression pool water is modeled in those cases in which failure to manually transfer the water suction to the suppression pool or the failure of the RHR heat exchanger would result in the loss of this second water source.

Relay chatter is described in the fault trees. It is assumed in the analysis that all relays are totally correlated and that when chatter occurs in one, all the others also chatter. Furthermore, it is assumed that relay chatter by itself does not lead to failure of a high-pressure system; it also requires the failure of the operator to manually reset the system.

Large uncertainties exist in this area but an attempt was made to reflect relay chatter impact. It should be noted that this simplified relay chatter model is considered rather crude and is intended to yield only very preliminary results.

A modified ADS fault tree was also developed in which the random failures and ac power are included as developed events. Relay chatter is also considered as a contributor to the failure of the ADS system.

The two low-pressure systems, LPCS and LPCI, and the RHR system are modeled very similarly. Besides the consideration of the random failures, these trees include relay chatter and manual failure to reset, seismically related pump failure, loss of ac power, and failure of the heat exchanger. Since all of these systems take suction from the suppression pool, failure of the RHR heat exchanger is assumed to result in the draining of the suppression pool leading to the loss of suction to the low-pressure systems. This assumed interdependency is being examined in the ongoing consequence evaluation.

BNL modified the SLC system fault tree to include, in addition to those events that are discussed in the low-pressure systems, a basic event describing the seismically induced failure of the SLC tanks.

The scram system model is almost identical to that of the GE analysis, with the exception that the weighting factor of failure of the shroud support, CRD guide tubes, and the HCUs are assumed to be unity.

Lastly, the seismic fault tree of the power system was modified. This tree is developed to model both the ac and the dc system. No distinction is made regarding the different divisions that are in the GESSAR II design; this is consistent with the earlier discussion of dependence. Relay chatter is included in this tree to represent the chatter failure mode of numerous relays within the electrical system. The other components considered are: diesel generator control panel, the 125-V dc bus, diesel generator heat vent, the 480-V switchgear, the 480-V transformer breaker, and the 4-kV bus/switchgear.

As a result of these issues, new seismic functional event trees were developed to model better the progression of various accident sequences. These new trees are described in the staff/BNL GESSAR II seismic evaluation. They reflect changes to the progression of loss of offsite power sequences, including common-mode diesel generator failures, and ATWS events.

(1.4) Staff/BNL Analysis

In the staff/BNL analysis approximately 600 minimal cutsets were generated and these cutsets were evaluated by a screening analysis based on the median capacity and the random failure probability of the components. About 60 of them, deemed to be significant, were grouped in eight categories.

These eight groups of accident sequences are integrated with the various hazard functions and component fragilities to arrive at the core-damage frequency. Uncertainty in the hazard functions, as well as the fragilities, were considered in the quantification.

A summary of the results is presented in Table 15.3. In this table, the eight categories of core-damage sequences are enumerated on the left side; two sets of core-damage frequencies are presented for each category using either the GE fragility values for GESSAR II structures and components or the staff/BNL case 1 fragilities. It should be noted that these calculations were performed using the staff/BNL system fault trees and functional event trees and the GESSAR II hazard curves. It should also be recognized that these values constitute point estimates. A rigorous uncertainty analysis was not conducted. However, the sensitivity study discussed in the summary section gives some scope to the very large ranges potentially associated with these estimates.

The first group of accident sequences involves building failures. It can be seen that with staff/BNL analysis using the GE fragilities, the core-damage frequency due to building failure is 6×10^{-7}, whereas if the building fragilities are changed to those of the staff/BNL's, the core-damage frequency is increased by an order of magnitude. Table 15.2 identifies the capacities of these buildings identified in Table 15.1. These capacities are dominated by the piping capacity because of sliding or rocking between buildings. However, in the case of the drywell, containment building, and shield building, a single capacity value was assumed. This was done since these structures are supported on a common basemat. A simultaneous failure of the RPV, drywell, and containment was estimated to be approximately 1×10^{-9}.

The GE analysis calculated a building-related core-damage frequency of 6.6×10^{-8}. The difference between this value and the 6×10^{-7} value calculated by the staff/BNL using GE fragilities is due to the staff/BNL assumption that failure of the diesel buildings or the shielding building of the control building would result in core damage.

The second group of accident sequences contains those ATWS accident sequences with loss of both onsite and offsite power; but the buildings are intact. The contribution of loss of onsite power comes from a three-diesel-generator common-mode, 3-DGCM, failure, or common-mode failure of the diesel generator structure, DGS. It is pertinent to point out that the assumption that all three diesels will experience common-mode structural failure is conservative since they are located in two different buildings and since the division 3 diesel is designated to be of a different design; these factors all contribute significantly in reducing the correlation between them. The increased frequency of core damage identified in Table 15.3 (from 1.9×10^{-9} (GE) to 4.2×10^{-7} (staff) can be ascribed to the change of DGS capacity from 2.09 g to 1.5 g, and of the 3-DGCM random failure from the GE value of 1.0×10^{-4} to the staff BNL estimate of 4.2×10^{-4}.

The third group of accident sequences is also related to ATWS events. It characterizes those sequences with loss of offsite power and two-diesel-generator common-mode, 2-DGCM, random failure. The two orders of magnitude increase in core-damage frequency is attributable to the lower capacity values of the scram system components.

The fourth group of accident sequences involves events with loss of offsite power but with onsite power and successful scram. The sequences in this category can be divided into two groups: the first one contains all the electrical components and service water; the second group contains the heat-exchanger-related failures. Changing the fragility values affects the result: the 3.2×10^{-5} value is to be compared with the value of 2.7×10^{-6} using GE fragilities. The difference is due to the more-detailed modeling of the electrical components in the staff/BNL analysis.

The fifth group describes the ATWS sequences with only onsite power. The sequences here can be divided into two groups: the electrical components and the SLC system components constitute the first, and the ADS-inhibit and level-control constitute the second. The increase from GE fragilities to staff/BNL fragilities influences the core-damage frequency by a factor of 5.

The sixth group resembles those LOOP sequences with a 2-DGCM random failure. They are dominated by the electrical components, the recovery of onsite diesels within 24 hours (RDG24), and the heat exchanger. The use of the staff/BNL fragilities increases the core-damage frequency by approximately a factor of 30.

The seventh group is very similar to the second, except that it consists of transient instead of ATWS events. Results show that the capacity of the DGS component is less dominant and only about a factor of 3 is noted with the change in fragilities.

The last group represents accident sequences without loss of offsite power. Changes in component fragilities appear to have substantial impact on the core-damage frequency, from 8.2×10^{-8} to 1.5×10^{-5}.

Summary

As part of the review, the staff/BNL performed an assessment of the core-damage frequency due to seismic events for the GESSAR II design. Results show an

overall increase in the total core-damage frequency compared to the GE analysis. Similar increases are also noted when individual accident sequences are compared. Sequences that are omitted in the GE analysis were assessed. As a result of the system analysis modeling changes, there is approximately an order of magnitude increase from GE value of 6×10^{-7} to the staff/BNL value of 6.3×10^{-6}.

Adopting the staff/BNL fragilities, the system analysis yields another factor of 10 increase to 6.7×10^{-5}. Most of this increase comes from the inclusion of relay chatter into the seismic analysis.

A sensitivity study conducted by the staff/BNL indicated that worst-case fragility values and unfavorable siting locations could increase the point estimate to approximately 10^{-3} per year. This large range is a result of a number of uncertainties regarding the GESSAR II site, specific component and structural fragilities, and impact of relay chatter.

The staff is still evaluating the potential for significant pool bypass sequences. Exclusive of these considerations and assuming the GE seismic hazard curve, staff/BNL case 1 fragility values and system analysis, seismic risk is believed to be similar to the internal-events risk, since the total core-damage frequency is approximately equal and the release categories are similar. However, as part of the design modification review, the staff is continuing its evaluation of the dominant seismic accident sequences to determine if reasonable avenues exist for reducing plant risk.

(1.5) Conclusions

From its evaluation of the GESSAR II seismic risk assessment, the staff has concluded that the analysis as submitted by GE does not support a complete assessment of seismic risk for the GESSAR II design. This is due to identified deficiencies in the areas of site hazard function, structure/component fragility analysis, and seismic systems modeling.

The staff performed a limited assessment and sensitivity analysis to quantify the potential core-damage contribution for the GESSAR II design at a typical site. Utilizing more-representative fragility values and improved systems modeling, the staff/BNL calculated a point estimate for core-melt frequency from seismic events of 6.7×10^{-5} per year which is comparable to the contribution from internal events. This estimate and its comparison contains considerable uncertainty. Using the highest site-specific hazard function found in the literature search and utilizing more-pessimistic fragility values, the staff/BNL sensitivity study indicated a core-melt frequency of approximately 10^{-3} per year. This would, however, be considered a combination of low likelihood, which could be controlled through the review of a site-specific application. The staff has not completed its consequence analysis for seismically initiated events. Additionally, outstanding concerns remain regarding the impact of relay chatter. Actions in this area may be available to reduce seismically induced core-damage events. The staff will address these issues in a future supplement to the SER.

Because of the wide range of uncertainties potentially associated with the seismic impact for the GESSAR II plant, the staff will require that a utility applicant referencing the GESSAR II design take certain actions regarding the seismic contribution to risk (in line with the staff's assessment). These actions are:

(1) Perform a site-specific hazard function analysis, and justify that it is bounded by the GESSAR II hazard curve.

(2) Develop a critical-components-and-structures list for the plant. Provide and justify fragility values for all critical structures and components and show that they are bounded by the values presented in the GESSAR II seismic-risk study. For the critical components not included in the GESSAR II list, an applicant must satisfy the Case 1 alternate fragilities presented in this supplement.

(3) Provide an identification of all additional seismic analysis assumptions utilized by the GESSAR II analysis and show that the as-built plant satisfied the assumptions.

In the event that these analyses indicate that the above conditions are not met, the utility applicant shall demonstrate that this does not result in any significant increment in risk.

Table 15.1 Additional structures, components, and failure modes omitted from the seismic PRA which should be included in the site-specific analysis

Item	Comment
Structures	
Crib house	-
Retaining walls	May protect buried safety-related piping
Stack	May fall and impact safety-related structures or components
Fuel building	-
Components	
Service water pump	Important in Zion PRA
Buried piping	-
Cable trays	-
Battery racks	-
Condensate storage tank	Assumed failed by GE
Diesel oil tanks and piping	-
Diesel generator control panels	-
Diesel generator bus	-
Diesel generator heat and vent	-
Diesel generator	Low capacity in Millstone PRA
Reactor pressure vessel internals	GE considered only shroud support
Instrumentation panels	Relay chatter may be important
Failure Modes	
Liquefaction	Highly site dependent
Differential settlement	Highly site dependent
Slope failure	Highly site dependent
Piping failure due to connecting building, rocking, or sliding	Important in Zion PRA

Table 15.2 General Electric Co. and NRC staff/BNL fragility values

Structure/component	General Electric Co.[1]		NRC staff/BNL Case 1		NRC staff/BNL Case 2	
	Median, g	β_c	Median, g	β_c	Median, g	β_c
Ceramic insulator	*	*	0.20	0.32	0.20	0.32
Pump	*	*	1.81	0.61	1.81	0.61
Piping	*	*	-[2]	-[2]	-[2]	-[2]
Heat exchanger	*	*	1.09	0.47	1.09	0.47
Valve (hydraulic or air)	*	*	-[2]	-[2]	-[2]	-[2]
Valve (check or spring)	*	*	-[2]	-[2]	-[2]	-[2]
Shroud support	*	*	0.67	0.43	0.67	0.49
Control rod drive guide tube	*	*	1.37	0.45	1.37	0.48
Hydraulic control unit	*	*	1.24	0.63	1.24	0.63
Standby liquid control tank	*	*	1.33	0.33	1.33	0.33
Electric power						
Diesel generator panel[3]	*	*	1.56	0.52	1.50	0.67
125-V dc bus[3]	*	*	1.49	0.56	1.49	0.56
480-V transformer[3]	*	*	1.49	0.56	1.39	0.66
480-V switchgear[3]	*	*	1.46	0.58	1.46	0.58
4-kV switchgear[3]	*	*	1.46	0.58	1.46	0.58
Relay chatter[3]	*	*	0.60	0.67	0.60	0.67
Reactor pressure vessel	*	*	1.25	0.40	1.25	0.59
Auxiliary building	*	*	1.50	0.50	1.50	0.41
Drywell	*	*	1.50	0.50	1.50	0.50
Containment	*	*	1.50	0.50	1.50	0.50
Shield building	*	*	1.50	0.50	1.50	0.50

Table 15.2 General Electric Co. and NRC staff/BNL fragility values (continued)

			NRC staff/BNL			
	General Electric Co.[1]		Case 1		Case 2	
Structure/component	Median, g	β_c	Median, g	β_c	Median, g	β_c
Control building	*	*	1.50	0.50	1.50	0.40
Diesel generator building	*	*	1.50	0.50	1.50	0.50
Diesel generator heat & vent[3]	*	*	1.55	0.51	1.50	0.65
Service water system[3]	*	*	1.50	0.45	1.50	0.79
Condensate storage tank[3]	*	*	0.80	0.39	0.24	0.39
Diesel generator structural failure	-	-	1.50	0.50	0.91	0.49

[1] Values obtained from "GESSAR II Seismic Event Analysis" (GE, Sept. 1983), unless otherwise indicated.

[2] Not included in systems analysis since capacities are relatively high.

[3] Values were assumed for GE, since they were not given in "GESSAR II Seismic Event Analysis" (GE, Sept. 1983) or in "GESSAR II Seismic Event Uncertainty Analysis" (GE, Dec. 1983).

*Proprietary information withheld from public disclosure pursuant to 10 CFR 2.790.

Table 15.3 Sequence analysis results using GESSAR II hazard curves with BNL system models

Sequence	Mean sequence frequency, per year* GE fragilities	Staff/BNL fragilities
1. Building failures	*	5.88×10^{-6}
2. LOOP-ATWS sequences (with loss of onsite power)	*	4.22×10^{-7}
3. LOOP-ATWS (with 2-DGCM failures)	*	6.97×10^{-8}
4. LOOP (with only onsite power)	*	3.24×10^{-5}
5. LOOP-ATWS (with only onsite power	*	1.17×10^{-5}
6. LOOP (with 2-DGCM failure)	*	8.54×10^{-7}
7. LOOP (with loss of onsite power)	*	7.69×10^{-7}
8. Transients (with offsite power available)	*	1.47×10^{-5}
Total core melt	*	6.68×10^{-5}

* Point estimates.

** Proprietary information withheld from public disclosure pursuant to 10 CFR 2.790.

APPENDIX A

CONTINUATION OF CHRONOLOGY

August 10, 1984	Letter from applicant requesting exemption from revised license fee schedule for GESSAR II review.
August 20, 1984	Letter from applicant transmitting draft amendments to GESSAR II.
September 25, 1984	Letter to applicant requesting review of proposed SSER to determine if it contains any proprietary information.
October 15, 1984	Letter from applicant requesting Commission to take up severe-accident policy paper for vote at earliest possible date.
October 16, 1984	Letter to applicant transmitting advance copy of Supplement 2 to SER which contains five proprietary tables.

APPENDIX B

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APPENDIX E

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Supplement 3 to the Safety Evaluation Report (SER) for the application filed by General Electric Company for the final design approval for the GE BWR/6 nuclear island design has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. This report supplements the GESSAR II SER (NUREG-0979), issued in April 1983, summarizing the results of the staff's safety review of the GESSAR II BWR/6 nuclear island design. Subject to favorable resolution of the items discussed in this supplement, the staff concludes that the GESSAR II design satisfactorily addresses the severe-accident concerns described in draft NUREG-1070.

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BWR/6 NUCLEAR ISLAND DESIGN

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