50-447

October 16, 1984

Docket No. 00007447

Dr. Glenn G. Sherwood, Manager
Safety & Licensing Operations
Nuclear Power Systems Division
General Electric Company
175 Curtner Avenue, Mail Code 682
San Jose, California 95125

Dear Dr. Sherwood:

SUBJECT: ISSUANCE OF SSER 2 FOR GESSAR II

Enclosed for your use is an advance copy of SSER 2 for GESSAR II. The SSER contains 5 tables that identify information considered to be company proprietary by G.E. These tables have been marked accordingly and will be withheld from public disclosure. Advance copies of SSER 2 were provided to the ACRS on October 5, 1984.

Final copies will be provided when they are received from the printer. Any significant changes in the final copies from the advance copy will be identified.

Sincerely,

Original signed by
Cecil O. Thomas, Chief
Standardization and Special
Projects Branch
Division of Licensing

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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ABSTRACT

Supplement 2 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 (GESSAR II) 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.

The review is carried out in accordance with the procedures for demonstrating the acceptability of the design for the severe-accident concerns described in draft NUREG-1070, "NRC Policy on Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulation." Supplement 2 also provides more recent information regarding resolution or update of the confirmatory items and FDA-1 conditions identified in SSER 1.

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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ACRONYMS AND INITIALISMS

ADS	automatic depressurization system
AFWS	auxiliary feedwater system
ALARA	as low as reasonably achievable
APS	American Physical Society
ARI	alternate rod insertion
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTPO	Accident Source Term Program Office
ATWS	anticipated transient without scram
BNL	Brookhaven National Laboratory
BOP	balance of plant
BTP	Branch Technical Position
BWR	boiling water reactor
CDF	core-damage frequency
CECO	Commonwealth Edison Co.
CFR	Code of Federal Regulations
CP	construction permit
CRAC	Computation of Reactor Accident Consequences
CRD	control rod drive
CRGR	Committee to Review Generic Requirements
CS	containment spray
DEGB	double-ended guillotine break
DF	decontamination factor
DHR	decay-heat removal
ECCS	emergency core cooling system
EPZ	emergency planning zone
ERIS	Emergency Response Information System
ESF	engineered safety features
FDA	Final Design Approval
FSAR	Final Safety Analysis Report
GDC	General Design Criteri(on)(a)
GE	General Electric Company
GESSAR	General Electric Standard Safety Analysis Report
GSI	generic safety issue
HELB	high-energy line break
HPCI	high-pressure core injection
HPCS	high-pressure core spray
HVAC	heating, ventilating, and air conditioning
ICS	integrated control system
INPO	Institute of Nuclear Power Operations
ISI	inservice inspection

LCS	leakage control system
LLNL	Lawrence Livermore National Laboratory
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LWR	light-water reactor
MAAC	Mid Atlantic Area Reliability Council
MCPR	minimum critical power ratio
MSIV	main steam isolation valve
MSIVLCS	main steam isolation valve leakage control system
MTTR	mean time to repair
NDE	nondestructive examination
NPSH	net positive suction head
OBE	operating basis earthquake
OL	operating license
QUEST	Quantitative Uncertainty Estimation of Source Terms
PASNY	Power Authority of the State of New York
PCS	power conversion system
PDA	Preliminary Design Approval
PECO	Philadelphia Electric Co.
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PWR	pressurized water reactor
QA	quality assurance
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RES	Office of Nuclear Regulatory Research, NRC
RG	regulatory guide
RHR	reactor heat removal
RPS	reactor protection system
RPV	reactor pressure vessel
SAR	safety analysis report
SBGT	standby gas treatment
SCFH	standby cubic feet per hour
SDV	scram discharge volume
SER	Safety Evaluation Report
SLCS	standby liquid control system
SORV	stuck-open relief valve
SROA	safety-related operator action
SRP	Standard Review Plan (NUREG-0800)
SRV	safety/relief valve
SSE	safe shutdown earthquake
SSER	supplement to the safety evaluation report
SSI	soil-structure interaction
SWS	service water system

TMI-2	Three Mile Island, Unit 2
UPPS	ultimate plant protection system
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 General Electric Company'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.

This report is the second supplement to the Safety Evaluation Report (SSER 2). It provides more recent information regarding resolution or update of the confirmatory items and FDA conditions identified in SSER 1. It also provides a discussion and evaluation relating to the staff's review of GESSAR II for severe-accident concerns. The evaluation is based on GE submittals through July 1984. The evaluation includes independent as well as confirmatory analysis by the staff and its contractor, Brookhaven National Laboratory (BNL). The analysis is carried out in accordance with the procedures for demonstrating the acceptability of the design for severe-accident concerns described in draft NUREG-1070, "NRC Policy on Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulation."

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 C has been updated to include a further discussion of unresolved safety issues and a discussion of medium-priority and high-priority generic safety issues required by draft NUREG-1070. Appendix E is a list of principal contributors to this supplement. Appendix G is a discussion of compliance with the CP/ML rule (10 CFR 50.34(f)) and in Appendix H, compliance with the Standard Review Plan (SRP) rule (10 CFR 50.34(g)) is discussed.

The NRC Licensing Project Manager for GESSAR II is Mr. Dino Scaletti. Mr. Scaletti may be reached by calling him at (301) 492-9787 or by writing to him at the Division of Licensing, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

1.8 Summary of Outstanding Issues

In SSER 1, all outstanding issues had been resolved to the staff's satisfaction. Three of the issues that were potentially outstanding were made conditions of FDA-1. These issues are to be resolved before a construction permit or operating license is issued to the first applicants referencing GESSAR II. Presently one FDA-1 condition has been resolved (fuel rod internal pressure), and this will be reflected in FDA-1 at a later date.

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 or (2) GE needs to supply additional information. The outstanding issues as well as the FDA-1 conditions are tabulated below. For those items discussed in this supplement, the relevant section is indicated in parentheses following the item.

Issue	Status
Fuel rod internal pressure (4.2.1)	FDA-1 condition (resolved)
Humphrey issues	FDA-1 condition
Postaccident monitoring instrumentation	FDA-1 condition
External events (seismic PRA) (15.6.2.3)	Under review
Containment structural analysis (15.6.2 and Appendix G)	Awaiting information
Hydrogen control measures, USI A-48 (Appendixes C and G)	Awaiting information
Potential design modification (15.6.4)	Under review
Safety parameter display system (Appendix G)	Under review
Containment emergency sump reliability, USI A-43 (Appendix C)	Under review
Safety implications of control systems, USI A-47 (Appendix C)	Under review
Loads, load confirmations, stress limits, GSI B-6 (Appendix C)	Under review
Passive mechanical failures, GSI B-58 (Appendix C)	Under review
Beyond-design-basis accidents in spent fuel pool, GSI B-2 (Appendix C)	Under review

1.9 Confirmatory Issues

SSER 1 listed nine confirmatory issues that were either under staff review or awaiting information. The tabulation below shows the current status of each of the nine issues as well as the new confirmatory issues. For those items discussed in this supplement, the relevant section is indicated in parentheses following the item.

Issue	Status
Soil-structure interaction (3.7.1)	Resolved
Fuel assembly seismic-and-LOCA loads (4.2.1)	Resolved
Overheating of gadolinia fuel pellets (4.2.3)	Resolved
Containment long-term response (6.2.1)	Resolved
Subcompartment analysis (6.2.1)	Resolved

Issue	Status
Revised small-break LOCA methods (6.3.3)	Resolved
Factor of safety against sliding	Awaiting information
Software engineering manual	Awaiting information
Optical isolators	Awaiting information
Containment repressurization (6.2.2)	Resolved
Combustible gas control	Under review
Station blackout, USI A-44 (Appendix C)	Awaiting information
Shutdown decay heat removal, USI A-45 (Appendix C)	Awaiting information

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 SSER 1.

Table 1.2 Interface items

SER Section	Item
4.2.3	Fuel rod mechanical fracturing
4.2.3	Fuel assembly structural damage
4.2.4	Post-irradiation surveillance
6.2.1	Subcompartment pressure analysis
6.2.2	Containment repressurization
15.6.2	Quality assurance and interface requirements
15.6.2.3	Internal and external flooding analysis
15.6.2.3	Aircraft strike
15.6.2.3	Snow and ice loadings
15.6.3.1	System interaction (USI A-17)
15.6.3.2	Behavior of BWR Mark III containments (GSI B-10)
15.6.3.2	Proposed requirements for improving the reliability of open cycle service water systems (GSI 51)
15.6.3.2	Probability of core melt due to component cooling water system (GSI 65)
Appendix G	CP/ML rule items

Note: USI = unresolved safety issue; GSI = generic safety issue.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.7 Seismic Design

3.7.1 Seismic Input

As reported in the SER, there were two outstanding confirmatory issues related to GE's soil-structure interaction (SSI) analyses. The first issue dealt with the input location of the design ground motion in the finite element SSI analysis. GE applied input design ground motion (Regulatory Guide (RG) 1.60) at the grade level rather than at the foundation level in the free field as specified by the staff acceptance criteria. The second issue arose from the fact that GE's SSI analysis using a compliance function approach also deconvolved the input motion from the grade level to the foundation level. Thus, the staff's requirement regarding the input location of the design ground motion was not met.

In order to resolve these issues, GE had committed to perform an additional eight cases of SSI analyses of the reactor building using the compliance-function approach with RG 1.60 motion as input at the foundation level to demonstrate that the existing design envelopes and design parameters exceed those produced by these additional analyses. By a letter dated August 17, 1983, GE submitted results of the additional analyses for the staff's review.

GE performed the additional eight analyses using the CLASSI series of computer programs. The following two items emerged from the staff's review of these CLASSI analyses: (1) the CLASSI approach included kinematic interaction effects which modified the input motion at the foundation level. Kinematic interaction occurs as a result of wave scattering at the soil-foundation interface. Both kinematic constraints imposed by the geometry of the problem and the differences in stiffness between the structure and the soil contribute to the resultant wave scattering. For an embedded structure subjected to vertically propagating shear waves, the net effect is typically a reduction in rigid body translational response, accompanied by the introduction of a rigid body rocking component; and (2) GE assumed that the reactor building behaves like an embedded cylinder in direct contact with the soil for evaluating the embedment effects.

The first concern arises from the fact that the GE approach still alters (reduces) the input motion at the foundation level in the free field from the RG 1.60 motion and complies neither with an earlier GE commitment nor with the staff's position that the RG 1.60 motion be used at the foundation level in the free field. The cause for the second concern arises because the reactor building is surrounded by the control and auxiliary building and thereby physically separated from the soil; however, to evaluate embedment effects, GE assumed the reactor building was in direct contact with the soil.

The staff met with GE and GE's consultant on February 16, 1984, to discuss SSI analyses and the above concerns. As a result of this meeting, GE performed an additional analysis for the bounding soil case by completely disregarding kinematic interaction and conservatively assuming the reactor building to be surface founded. The comparison of spectra resulting from this analysis with the

GESSAR II design envelopes indicates that except for the low frequency range ($\leq$4 Hz), the GESSAR II design envelopes still bound the new spectra with considerable margin. Therefore, on the basis of the above review findings and the consideration of GE's previous commitment to limit the fundamental frequencies of structures, equipment, and components above 4 Hz, the staff concludes that all issues relative to the input location of the design ground motion and SSI analyses are resolved.

4 REACTOR

4.2 Fuel System Design

4.2.1 Design Bases

4.2.1.1 Fuel System Damage Criteria

4.2.1.1.6 Fuel and Poison Rod Pressures

In the SER, the staff stated that GE has proposed a design basis on rod internal pressure which does not limit the fuel rod internal pressure to a value less than the reactor coolant system (RCS) pressure as specified in Standard Review Plan (SRP) Section 4.2.II.A.1.f. The staff made this issue a condition of FDA-1 pending a demonstration by GE that a fuel rod internal pressure greater than RCS pressure would not (1) lead to fuel system damage during normal operation and anticipated operational occurrences, (2) prevent control rod insertion when required, (3) result in an underestimate of the number of fuel failures in or radiological consequences of postulated accidents, or (4) lead to loss of coolable geometry.

In a letter dated February 25, 1983, GE described a design basis for rod pressure in which the effects of fuel rod internal pressure during normal steady-state operation will not result in fuel failure from excessive cladding pressure loading. GE contended that a rod internal pressure limit of less than or equal to the RCS pressure is not necessary. Instead, GE proposed that the rod pressure be limited so that the instantaneous cladding creepout rate from internal pressure greater than RCS pressure is not expected to exceed the instantaneous fuel swelling rate.

To demonstrate that this proposed criterion is acceptable in terms of conditions (1) through (4) above, GE demonstrated that for the design-basis transients and accidents of interest in a BWR, either the cladding does not heat up significantly or the existing fuel damage criteria used are still applicable when the initial fuel rod internal pressure exceeds the initial RCS pressure.

In the case where the cladding does not heat up significantly, that is, the safety limit minimum critical power ratio (MCPR) is not exceeded, there is no significant change in the fuel rod geometry so that control rod insertion and bundle coolability will be maintained.

For those events in which the cladding does heat up significantly above its normal temperature, GE has demonstrated that there are other criteria which assure that conditions (1) through (4) will not occur. For example, the loss-of-coolant accident (LOCA) event is governed by the criteria set forth in 10 CFR 50.46 that the cladding temperature will not exceed 2200°F, the maximum amount of local oxidation on any fuel rod will not exceed 17%, and that a coolable geometry will be maintained. These criteria are independent of the initial internal pressure of the fuel rod. However, the internal pressure for the fuel rod is taken into account explicitly in determining the stored energy and in calculating the amount of fuel rod swelling and rupturing. In addition, the number of failed fuel rods assumed for radiological calculations is 100% of those in the core. Therefore, a rod internal pressure greater than the RCS pressure will not result in underestimating the radiological consequences of a LOCA. Thus, a fuel rod internal pressure greater than RCS pressure is acceptable for LOCA.

Similarly GE has evaluated the rod drop accident and has demonstrated, in response to a staff question, that the criterion for fuel failure in a rod drop accident is still applicable (GE letter, April 23, 1984).

The staff reviewed all the transients and accidents postulated for a BWR and concludes that there is no case in which a fuel rod internal pressure greater than the RCS pressure results in an existing specified acceptable fuel design limit becoming invalid.

Therefore, the GE design criterion for rod internal pressure is acceptable.

4.2.1.2 Fuel Rod Failure Criteria

4.2.1.2.8 Fuel Rod Mechanical Fracturing

In the SER, the staff stated that mechanical fracturing of the fuel rod was a confirmatory issue because the review of GE's document, NEDE-21175-3-P, was not yet completed. This term "mechanical fracturing" refers to a cladding defect that is caused by an externally applied force such as a hydraulic load or a load derived from core plate motion. These loads are bounded by the loads of a LOCA and safe shutdown earthquake (SSE), and the mechanical fracturing analysis is usually done as a part of the seismic-and-LOCA loads analysis (see Section 4.2.3.3.4 of this supplement). GE has described the seismic-and-LOCA loads analysis, including fuel rod mechanical fracturing, in NEDE-21175-3-P, which the SER has referenced.

The staff has completed its review of NEDE-21175-3-P and approved that report for use in licensing safety analyses (NRC letter, October 20, 1983). The staff concludes that fuel rod mechanical fracture criteria in NEDE-21175-3-P are acceptable for GESSAR II.

4.2.1.3 Fuel Coolability Criteria

4.2.1.3.4 Fuel Assembly Structural Damage From External Forces

In the SER, the staff stated that the analysis of seismic-and-LOCA loads for GESSAR II was a confirmatory issue because the review of NEDE-21175-3-P was not yet completed. Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 and Appendix A to that section state that fuel system coolability should be maintained and that damage (including liftoff) should not be so severe as to prevent control rod insertion when it is required during these low-probability accidents. GE has described the seismic-and-LOCA analysis in NEDE-21175-3-P, which GESSAR II has referenced.

The staff has completed its review of NEDE-21175-3-P and approved that report for use in licensing safety analyses (NRC letter, October 20, 1983). Thus, the staff concludes that the design criteria for seismic-and-LOCA loadings on fuel assemblies are acceptable for GESSAR II.

4.2.3 Design Evaluation

4.2.3.1 Fuel System Damage Evaluation

4.2.3.1.6 Fuel and Poison Rod Pressures

In order to calculate whether the criterion of cladding creepout rate less than fuel swelling rate is met, GE first used the approved fuel performance code GESTR-MECHANICAL (NEDE-23785-1-(P)A) to get fuel and cladding irradiation conditions including temperature, pressure, fluence, etc. Then a bounding peak rod power was chosen for the analysis.

Using these conditions in the cladding creepout and fuel swelling equations, a comparison was made between cladding creepout rate and fuel swelling rate. GE stated (GE letter, February 25, 1983) that confirmation has been done for the current GESSAR II fuel designs of P8x8R and BP8x8R that the cladding creepout rate does not exceed the fuel swelling rate throughout the lifetime of the fuel rod. If for any reason this new fuel design criterion were to be violated, GE could modify such physical parameters as fill rod pressure or peak rod power limit to correct the problem. Since GE used cladding creepout and fuel swelling equations from the approved GESTR-MECHANICAL code, the staff considers the analysis method acceptable.

4.2.3.2 Fuel Rod Failure Evaluation

4.2.3.2.4 Overheating of Fuel Pellets

Fuel melting temperature as a function of exposure (burnup) and gadolinia content (of burnable poison rods) is discussed in Section 2.4.2.5 of NEDE-24011-(P)A. In that report, General Electric stated that the fuel is not expected to melt during normal operation, and that prediction is based on fuel temperature calculations performed with a model described in the proprietary supplement to Amendment 14 of GESSAR II (STN 50-477). Although limited melting during certain events such as an uncontrolled control rod withdrawal is permissible, such melting is not predicted to occur.

The staff reviewed the UO_2 properties (thermal conductivity and melting point) that are important in reaching this conclusion and agrees that UO_2 melting will not be a problem at GESSAR II plants during normal operation and anticipated transients as long as the 1% plastic strain criterion discussed in SRP Section 15.4.2 is not exceeded. At that time, however, the staff also noted that the effects of gadolinia concentration on thermal conductivity and melting temperature were addressed in an unreviewed GE topical report on gadolinia fuel properties (NEDE-20943-(P)A). That report has been replaced by another topical report (NEDE-23785-1-(P)A), which describes revised fuel thermal performance methods and gadolinia properties (Appendix B of NEDE-23785-1-(P)A). The NRC staff has reviewed and approved the more recent report.

General Electric has stated (GE letter, February 2, 1984) that gadolinia properties described in Appendix B of NEDE-23785-1-(P)A are generically applicable to new plants such as GESSAR II and has also confirmed that the applicable limits for overheating of gadolinia fuel remain valid. Because these limits were previously found acceptable for other BWRs and because GESSAR II has utilized approved methods and gadolinia properties to show that these limits continue to be met, the staff considers the issue resolved.

[illegible] Mechanical Fracturing

Since [illegible] ring analysis is site dependent, utility applicants referencing [illegible] must provide a plant-specific analysis of fuel rod mechanical fracturing to conform to the SRP Section 4.2, Appendix A, requirements using the approved model described in NEDE-21175-3-(P)A or an acceptable alternative.

Since the mechanical fracturing analysis is usually done as a part of the seismic-and-LOCA loads analysis, further discussion can be found in Section 4.2.3.3.4.

4.2.3.3 Fuel Coolability Evaluation

4.2.3.3.4 Fuel Assembly Structural Damage from External Forces

The staff has approved the GE topical report NEDE-21175-3(P)A (NRC letter, October 20, 1983) which describes an analytical method for evaluating seismic-and-LOCA loads. The utility applicants referencing GESSAR II must submit plant-specific values of liftoff and acceleration for staff review. If the results show that the vertical liftoff is insignificant, and the accelerations are within the evaluation-basis limits, then the structural integrity and control rod insertability during seismic-and-LOCA events can be assured. Since these analyses are site [illegible]ific, the utility applicants referencing GESSAR II must provide a plant-s[illegible] analysis of seismic-and-LOCA loads to demonstrate conformance with the SR[illegible]tion 4.2, Appendix A, requirements using the approved model describe[illegible] NEDE-21175-3(P)A or an acceptable alternative.

4.2.4 Testing, Inspection and Surveillance Plans

4.2.4.3 Postirradiation Surveillance

In the SER, the staff stated that a commitment to provide a minimal postirradiation surveillance program before reactor startup will be required by utility applicants referencing GESSAR II. In a letter dated November 23, 1983, GE proposed a generic fuel vendor surveillance program, which would satisfy the intent of SRP Section 4.2.II.D.3 that each licensee perform postirradiation fuel surveillance on fuel irradiated in the licensee's reactor. The program proposed by GE will allow that company to assume the responsibility for postirradiation fuel surveillance of GE-designed and GE-manufactured fuel. The staff approved this program in a letter dated June 27, 1984. Therefore, the staff concludes that the license condition for utility applicants referencing GESSAR II is not required as long as they reference the GE fuel surveillance program.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.4 Long-Term Response

The containment long-term response is used to determine the limits for the design pressure and temperature of the containment. In the SER (NUREG-0979), the staff concluded that the containment design pressure of 15 psig and temperature of 185°F were acceptable pending confirmatory evaluation. In the long-term analysis, as reported in NUREG-0979, GE accounted for potential postaccident energy sources including decay heat, pump heat, sensible heat, and metal-water reaction energy.

GE's long-term model also assumed that the containment atmosphere is saturated and at a temperature equal to the suppression-pool temperature at all times. Therefore, the containment pressure is equal to the sum of the partial pressure of air and the saturation pressure of water corresponding to the pool temperature. Thirty minutes following onset of the accident, the containment cooling mode of the residual heat removal (RHR) system is activated and suppression-pool water is circulated through the RHR system heat exchangers establishing an energy transfer path to the service-water system and the ultimate heat sink.

On the basis of the above assumptions, GE calculated a peak suppression pool temperature of 174.7°F for the most limiting RHR system cooling mode; that is, only one operating RHR heat removal system pump is available. The calculated long-term secondary peak containment pressure is 9 psig. The containment is designed for 15 psig and 185°F.

The staff has evaluated the containment analysis performed by GE and has made a comparison between it and other similar Mark III plants, particularly the Perry nuclear plant for which the staff performed a CONTEMPT LT/028 analysis to verify the Perry applicant's peak suppression-pool temperature and containment pressure calculation. The GESSAR II design and the Perry plant both have a reactor power level of 3,651 MWt as the initial condition for their analyses. In addition they have the same containment type, a free-standing steel shell each with a free air volume of 1.14 million cubic feet and suppression pool surface area of 5,900 ft^2. The GESSAR II design has approximately 11% more suppression-pool water volume at the minimum level than does Perry (117,510 ft^3 vs. 105,950 ft^3) as well as a larger upper pool makeup capability (37,665 ft^3 vs. 32,830 ft^3). In addition, the GESSAR II containment minimum heat removal capability with one RHR train in operation is approximately 13% greater than that of Perry (187 million Btu/hr vs. 166 million Btu/hr). The Perry applicant's calculations, confirmed by the staff, indicated that the maximum containment pressure was 12 psig and the maximum temperature was 185°F. The valves envelope those calculated for GESSAR II, which, as described above, has a greater heat removal capability than does Perry. On the basis of these considerations as well as a review of the other relevant assumptions and initial conditions used by GE in its analysis, the staff concludes that GE's design containment pressure and temperature valves are acceptable.

6.2.1.6 Subcompartment Pressure Analysis

As indicated in the SER, the forces and moments used to design the reactor cavity subcompartment were obtained through prior engineering experience with similar Mark III designs and were not obtained from a detailed analysis of the GESSAR II design data. The staff considers this to be an interface item and will require utility applicants to provide the forces and moments from a subcompartment analysis that they must perform based on GESSAR II design data. In addition, the staff will complete its analysis on the other subcompartments completed by GE at the time the first utility applicant references GESSAR II.

6.2.2 Secondary Containment

As indicated in SSER 1, the staff has stated that the proposed containment repressurization limit of 90% of containment design pressure was not acceptable. In response to this, GE has indicated that it will comply with the 50% containment repressurization limit. The staff will require that utility applicants referencing GESSAR II provide the analysis to verify that the 50% repressurization limit can be met.

6.3 Emergency Core Cooling System

6.3.3 Performance Evaluation

Revised Small-Break Loss-of-Coolant Accident Methods (TMI) Action Plan Item II.K.3.30

The SER indicated that the draft safety evaluation for the GE small-break model was under management review and that any concerns that were raised would be reported in a supplement to the SER. The draft safety evaluation concludes that the test data comparisons and other information submitted by GE acceptably demonstrate that the existing GE small-break model is in compliance with 10 CFR 50, Appendix K, and that, therefore, no model changes are required. The staff has completed its review and finds the GE model acceptable to resolve Item II.K.3.30.

15 TRANSIENT AND ACCIDENT ANALYSIS

15.6 Severe Accidents

15.6.1 Introduction

In order to achieve design certification through the rulemaking process, the design described in GESSAR II must be shown to be acceptable for severe-accident concerns. The procedure for demonstrating this acceptability is detailed in draft NUREG-1070, "NRC Policy on Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulation."

To summarize from the policy that covers future plants, the following four steps must be accomplished in order for the design to be shown acceptable for severe-accident concerns and to achieve certification through rulemaking.

(A) Demonstration of compliance with current Commission regulations including the Three Mile Island requirements for new plants as reflected in the CP (construction permit) rule (10 CFR 50.34(f)).

(B) Demonstration of technical resolution of all applicable unresolved safety issues (USIs) and the medium- and high-priority generic safety issues (GSIs).

(C) Completion of a probabilistic risk assessment (PRA) and consideration of the severe-accident vulnerabilities it exposes along with the insights that it may add to the assurance of no undue risk to public health, safety, and property.

(D) Completion of a staff review of the design with a conclusion of safety acceptability.

Step (A) was largely accomplished during the initial licensing review and SSER 1, of the GESSAR II submittal, and is documented in the SER (NUREG-0979) which recommended issuance of FDA-1. Discussion of GESSAR II compliance with 10 CFR 50.34(f) can be found in Appendix G to this supplement. However, Item (1)(i) of 10 CFR 50.34(f) remains in part to be performed. Item (1)(i) requires a PRA to seek significant improvements in plant design, which improves containment and core heat removal capability. This action along with Steps (B), (C), and (D) have been performed under the direction of the staff, to demonstrate acceptability for severe accident concerns and to support design certification through rulemaking. Completion of the requirements of these steps is essential for approval of GESSAR II as a design referenceable in new CP applications. Documentation of resolution for these topics is detailed in this supplement. A more complete description of each step and the actions taken to satisfy the requirements follows.

Topic (A): Compliance With Regulations

Item (1)(i) of 10 CFR 50.34(f) requires in part that potential improvements be considered which address the reliability of core and containment heat removal systems that are significant and practical and do not impact excessively on the plant.

To meet this requirement the NRC staff proposed a list (discussed in Section 15.6.3) of potential design improvements for GESSAR II arranged in 14 groupings according to the functional improvements that they potentially achieve.

(1) Accident Management/Human Factors
(2) Reactor Decay Heat Removal
(3) Containment Capability
(4) Containment Heat Removal
(5) Containment Atmosphere Mass Removal
(6) Combustible Gas Control
(7) Containment Spray Systems
(8) Prevention Concepts
(9) AC Power Supplies
(10) DC Power Supplies
(11) ATWS Capability
(12) Seismic Capability
(13) System Simplification
(14) Core Retention Devices

GE was asked to evaluate the proposed list of improvements and to provide cost-benefit analyses to ascertain the desirability of the proposed modifications. To accomplish this, GE assessed the approximate costs associated with the proposed design and the related potential risk reduction. GE submitted this material and the staff currently has it under review. A preliminary assessment of the GE material was made and the staff's conclusions on design improvements are presented in Section 15.6.3 of this supplement. Additional information on the staff's evaluation will be provided in a future supplement to the SER.

Topic (B): Resolution of USIs and GSIs

To satisfy this topic, the staff must conclude that all applicable USIs and high/medium priority GSIs, as identified in NUREG-0933, have been resolved for GESSAR II. Resolution has been accomplished through engineering evaluations and application of insights from the PRA to demonstrate that the issues contribute little risk to the public or in some cases resolution has been deferred for utility applicants that reference GESSAR II.

GE was required to assess all relevant USIs and high/medium priority GSIs. To demonstrate resolution for each USI and GSI, it had to be shown, where possible, that the contribution to the risk to the public from the USI or GSI is insignificant or that there is some other basis through which the public risk can be dismissed. When necessary, the limitations in PRA methodology, completeness, and the scope of design of GESSAR II were supplemented by further engineering assessments to show low societal risk. Where this could not be shown, GE was required to consider what remedial actions are necessary to reduce societal risk. The staff findings on the resolution of applicable USIs and GSIs are found in Appendix C of this supplement. In some cases, the present state of plant design does not permit complete assessment of an issue's risk impact. USI A-17 (dealing with systems interaction) and USI A-47 (dealing with the safety implications of control systems) are in this category. For these situations, deferral of USI or GSI resolution for utility applicants who reference GESSAR II is appropriate.

Topic (C): Completion of Probabilistic Risk Assessment

The GESSAR II PRA was submitted in March 1982 to satisfy in part the severe-accident policy requirement presently identified in draft NUREG-1070. The PRA approach provides a systematic method for evaluating the vulnerability and potential interaction of the plant systems and components. Normal operating systems as well as safety-related and support systems were considered.

The staff and its consultant, Brookhaven National Laboratory (BNL), evaluated the PRA submitted by GE to assess the validity of the assumption, methodologies, and results. A number of modifications to assumptions and methodologies were made in the course of review. The GESSAR II PRA originally dealt only with internal events. Subsequently, upon staff request, uncertainty analyses on internal and external event findings were provided. Further, a source-term sensitivity analysis has been provided to support the consequence analysis.

The source-term methodology used by GE constitutes a departure from previous industry PRAs in that parameters were used to account for phenomena not previously considered. Although GE provided a "best estimate" source term, the staff believes that the state of the art of source-term methodology is such that a range of source terms is all that can be justified. The staff is in the process of developing mechanistic codes for source-term calculations and is obtaining an in-depth review of the methodologies used to develop these codes by the American Physical Society (APS). The appropriate methodologies will be applied to the GESSAR II design, and if the results of the staff's present review change significantly, the staff's review will be supplemented in the course of certification of the GESSAR II design. The source-term methodologies used by the staff and GE are discussed in Section 15.6.2.5.

The GESSAR II PRA assesses the probability of core damage and risk associated with a BWR/6 nuclear island design at a representative average site for a full range of internal events. The impact of selected external events was also considered--seismic, fire, and internal flooding. Staff conclusions on fire and flooding are presented in Section 15.6.2.3. The staff will present the results of its evaluation of GE's seismic analyses in a future supplement. Additionally other external events identified and the impact of these are to be assessed by utility applicants referencing GESSAR II.

A description of the BWR/6 nuclear island, GESSAR II, can be found in Section 1 of the SER.

Topic (D): Conclusion of Safety Acceptability of GESSAR II Design

To resolve this topic, the staff is considering the results of the PRA study along with qualitative input to determine whether the GESSAR II design provides an acceptable level of risk to the public from severe accidents. Staff findings in regard to topics (A), (B), and (C), described above, provide the basis for conclusions of safety acceptability.

Staff evaluation of the GESSAR II PRA is provided in Section 15.6.2 of this supplement. Unresolved safety issues and generic safety issues are discussed in Appendix C, and potential beneficial design improvements are considered in Section 15.6.3. Section 15.6.4 contains staff recommendations and conclusions.

15.6.2 Major Review Results and Conclusions From PRA Review

The probabilistic risk assessment (PRA) on GESSAR II was carried out by General Electric to satisfy the proposed Commission requirements relating to severe accidents (presently embodied in draft NUREG-1070). The intent of the study is to provide a systematic evaluation of the design and operational features of the BWR/6 nuclear island, GESSAR II, based upon the techniques of reliability and risk analysis. Using these analysis tools, it is possible to identify potential weaknesses in the plant design, gain some insight into the overall safety of the design, and evaluate potential modifications to the design which would reduce public risk.

Contrary to the standard NRC licensing evaluations which compare plant design features to deterministic standards, a PRA study investigates the likelihood and consequences of multiple failures and the pathways which could lead to core damage and release of radionuclides. As such it should be considered a complement to, not a substitute for, current licensing standards.

A PRA analysis considers and quantifies the probability of core damage occurring following some upset plant condition (transients or accidents) where failures in plant systems inhibit the ability to respond successfully. Results are presented as accident sequences and their associated frequency of occurrence. The GESSAR II PRA analyzed the core-damage frequency for all recognized internal accident sequences as well as for the external events of earthquake, fire, and internal flooding.

Although calculations of the frequency of core damage are useful, they do not by themselves provide a complete picture of the public risk resulting from plant accidents. This is because the actual consequences of an accident are dependent upon the amount, timing, and pathway of radioactive release, which vary among the various accident sequences. Therefore, the GESSAR II PRA also presented the results as expected consequences. This was done by grouping the various important accident sequences into representative subgroupings having similar behavior. The timing and mode of containment failure was an important element in the analysis. Utilizing consequence models (described in Section 15.6.2.5) it was then possible to express public risk in the form of expected exposure (person-rem) from the various accident groupings. The resulting exposures along with the frequency of the event allow one to determine the importance of accident sequences with respect to public risk. However, because of the nature of the parametric study used for the source-term evaluations, the uncertainty of the risk that is calculated must be considered along with the point values. The cost/benefit evaluations, therefore, are used as a screening tool.

Although the PRA provides very useful insights for evaluating the nuclear island design described in GESSAR II for vulnerabilities to severe accidents, the quantitative risk estimate results must be viewed with some caution since the design is not site specific. Additionally, the nuclear island does not include some balance-of-plant (BOP) features, which could potentially impact plant risk. Because of these considerations, GE had to make assumptions of various interface criteria and of the reliability of various systems and components.

The validity of the PRA insights depends on meeting the interface and reliability assumptions discussed above. GE documented its program for assuring that interface criteria are properly met, and provided a table of interface requirements to be satisfied by a utility applicant in order to validate the PRA results. However, in assembling this list, GE excluded elements in which the assumed component reliability or operator action time satisfied the recognized industry data base, or the assumed values were of little importance to the PRA conclusions (e.g., 100% change in unavailability changes the corresponding overall PRA results by less than 1%).

The staff is concerned that these exclusions may adversely impact the validity of the PRA results. Using an assumed component reliability in the PRA corresponding to a recognized data base does not ensure that a purchased component or system will meet required reliability goals. Likewise, a non-plant-specific operator-action time would not account for plant- or site-specific features that could degrade operator performance. Additionally, the exclusion of components where a 100% change in unavailability impacts PRA results less than 1% is suspect. A 100% change in component unavailability is very minor compared with the possible reliability impact from problems in design, maintenance, or potential common-mode effects.

For the above reasons, the staff believes that a utility applicant referencing the GESSAR II design must provide an evaluation to support the PRA interfaces and assumptions to demonstrate that the PRA is applicable to its proposed plant. Of particular concern are the impact of site-specific hazards such as tornado and hurricane on large non-safety-related BOP structures (i.e., cooling towers). This evaluation must also address the potential for debris damaging safety-related equipment such as buried piping.

In addition to the above evaluation to demonstrate satisfaction of PRA interface assumptions, the staff is concerned that PRA findings remain valid throughout the operational life of a GESSAR II plant. To this end, the staff recommends that any applicant referencing the GESSAR II design propose a program, similar to that discussed in draft NUREG-1070, to ensure the PRA findings remain valid. This program should provide an ongoing evaluation of the plant's operational performance to ensure that plant safety is not being degraded. It should consider transient and accident initiation frequencies, component and system reliability, and operator-action performance. Operational experience should be continually reassessed and compared with the initial PRA assumptions to provide an ongoing validation of the PRA conclusions.

Remedial actions should be taken if operational experience indicates significant PRA assumptions are not being satisfied.

15.6.2.1 Dominant Accident Sequences

(1) Summary of Core-Damage Sequences--Internal Events

The review of the GESSAR II PRA has confirmed that the standard plant core-damage frequency is dominated by transient initiators, with loss of offsite power (primarily station blackout) as the major contributor to the probability of core damage. Tables summarize staff/BNL review results of estimates of the frequency of core damage due to internal events for the nuclear island design as compared with the original GE estimates. The numbers expressed in those tables represent estimates of core damage before the consideration of design improvements. The impact of these improvements is discussed in Section 15.6.3. From staff/BNL review, the point estimate frequency of core damage was 2.2×10^{-5} per year for a site associated with a grid in the Mid Atlantic Area Reliability Council (MAAC). This frequency for core damage is to be compared with the 4.4×10^{-6} value presented in the GESSAR II PRA (for a site in the same region). The frequency of core damage is sensitive to the assumed frequency of the loss of offsite power (LOOP). If a national average estimate for LOOP is used, the staff/BNL estimate of frequency of core damage increases to 3.8×10^{-5}.

As mentioned above, station blackout events (comprising most of sequence T_EUV)* are the major contributor to the probability of core damage. These involve transients caused by a loss of offsite power coupled with failure of the high-pressure (U) and low-pressure (V) injection functions. These sequences contribute 68% to the frequency of core damage for the MAAC site and 79% for the national average site. The important failure combinations leading to this sequence were common-mode failure of the three emergency diesel generators and failure to recover either an onsite or an offsite ac power source in time to avoid core damage, and a failure of two emergency diesel generators coupled with some other failure in the high-pressure core injection systems (high-pressure core spray, HPCS, and reactor core isolation cooling, RCIC). Other LOOP transients (T_EW) involving successful injection with the HPCS, failure to recover offsite power and common mode failure of Divisions 1 and 2 diesel generators or batteries, and failure to recover power in 27 hours constitutes approximately a 4% contribution to the core-damage frequency for the national average site.

The values given for the LOOP events were based upon a 2-hour station blackout capacity in which the plant battery was assumed to supply dc power for operation of the RCIC which provides core cooling. Cooling of the RCIC pump room was a major limiting factor in establishing successful operation periods. Subsequent to the initial submittal of the PRA, GE has indicated that with minor procedure and hardware modifications, the RCIC plus station batteries can supply core cooling in excess of 10 hours. This capability results in reductions in core-damage frequency (CDF) of at best a factor of 3 for this specific accident category. This design modification is discussed in detail in the sequence discussions later in this section. The rest of the results in this section assume only a 2-hour blackout capacity, and the national average grid reliability.

All of the anticipated transient without scram (ATWS) events, taken together, contribute about 9% of the total core-damage frequency (based upon the national average grid reliability). The reassessed core-damage frequency of these sequences was increased significantly as a result of staff review of the original GE submittal, where it represented approximately 1% of the core-damage frequency of the initial GE unrevised core-damage frequency. Although the revised ATWS sequence frequencies were increased considerably in the course of staff

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*Sequences are defined in note to Tables 15.1a and 15.1b.

review, their absolute magnitude at 3.4 x 10^{-6} core-damage events per year is relatively low. These numbers reflect the reduction in risk achieved through ATWS modifications incorporated in the plant design, as a result of staff and industry efforts to resolve ATWS. These modifications are discussed in detail in the ATWS sequence presentation.

Another dominant accident sequence (T_FQW) is one initated by an isolation transient involving failure of feedwater systems and the containment heat-removal systems (RHR and power conversion system). This sequence contributes approximately 5% to the CDF for the national average site. The GE PRA showed essentially no contribution to the CDF impact from this sequence. The staff/BNL evaluation (1.9 x 10^{-6}) resulted in a significant increase from the GE value (4.4 x 10^{-9}). Changes to the analysis of recovery of containment cooling largely contributed to the increase.

All remaining transients and accidents together contribute approximately 5% to total frequency of core damage. The highest remaining categories are isolation transients, at approximately 1.4%, involving failure of high-pressure systems and the automatic depressurization system (ADS). The loss of dc power transient contributes approximately 1%, and the inadvertent opening of relief valve transient contributes approximately 0.3%. There are no dominant LOCA sequences; together all LOCA events contribute less than 0.1% to core damage. Table 15.1 compares the GE and BNL results for the dominant sequences of internal events.

(2) Modifications to PRA Analysis Resulting From Review

As mentioned previously, the staff/BNL review results differed significantly from GE's core-damage frequencies. A difference of a factor of 5 between the staff/BNL point value (MAAC site) and 9 for the national average site can be attributed to two factors. The first is related to additional dependencies between safety functions that the review identified and to certain modeling modifications that were made to the accident sequence event trees and system fault tree parts of the GESSAR II PRA. The dependencies between the safety functions exist as a result of the use of common support systems for different systems in the GESSAR II design. Dependencies between the initiating events and mitigating systems were also modeled in the staff/BNL analysis. A detailed account of these changes will be presented in the BNL GESSAR II evaluation.

The second factor that led to a difference between staff/BNL and GE results is the different values that were used for the frequencies of some of the accident initiators.

For the MAAC site, the first factor accounts for 72% of the increase; 28% comes from the second factor. For the national average site, 38% of the increase is due to the first factor and 62% is due to the second factor. It is noteworthy that the GESSAR II PRA did not present a national average frequency for LOOP and, hence, the second staff/BNL point value should be regarded with this perspective.

The staff/BNL review of the GESSAR II PRA identified the following major areas of disagreement in modeling methods and data between BNL and GE. These are discussed below:

(a) The probability of common mode failure of all three emergency diesel generators: The GESSAR II design includes two emergency diesel generators that supply emergency power to the various safety loads in the event of a LOOP, and a third, separate, and dedicated diesel generator that powers the HPCS system. According to the design, this third diesel will be housed in a different building, will be supplied by a different manufacturer, and further, all efforts will be made to ensure separation and diversity between this diesel and the other two. On the basis of this information and on the analysis presented in NUREG/CR-2989, "Reliability of Emergency AC Power Systems at Nuclear Power Plants," the staff assessed the probability of common-mode failure of the three diesels at 4×10^{-4} per demand. This value is to be compared with the initial 6×10^{-4} value used in the GESSAR II PRA. In September 1983, GE submitted an updated common-mode-failure analysis for the diesel generators in which the use of the 1×10^{-4} (or smaller) value for this failure is advocated. The staff/BNL performed a limited review of this submittal and do not believe that the 1×10^{-4} value is adequately supported by the submitted analysis. In any event, even if the value of 1×10^{-4} were to be used in the assessment the core-damage frequency (staff/BNL value) would be reduced by about 32% for the MAAC site and 40% for the national average site, and the relative ranking of the various accident sequences would not change.

(b) The frequency of failure of the reactor protection system: The staff/BNL review used a failure rate of 3×10^{-5} per demand (1×10^{-5} for the mechanical subsystem and 2×10^{-5} for the electrical subsystem), as proposed in NUREG-0460, "Anticipated Transients Without Scram for LWRs." The GESSAR II PRA used a failure probability of 1×10^{-7} per demand for failure to scram (this value includes the failure of the alternate rod insertion (ARI) system). The staff/BNL believe that such a low failure probability for a system cannot be supported by analysis alone and there is no experiential evidence that can support such a low value. If the 10^{-7} value is used, the staff's estimated frequency of core damage would decrease by 15% for the MAAC site case and 9% for the national average site.

(c) The frequency of transients: There is a difference of about a factor of 2.5 in the frequency of turbine trip and loss-of-feedwater transients between the staff/BNL estimates and those presented in the GESSAR II PRA. The staff/BNL values are based solely on the operating experience of the various boiling water reactor (BWR) power plants. The GESSAR II PRA values are also based on the operating experience of BWRs but modified to include the effect of design changes in the GESSAR II design, as well as the fact that the frequency of transients in operating plants exhibits a reduction with operating time ("burn in" effect). The staff/BNL believe that, in general, there is merit in these approaches. They did not, however, agree with the details of the analysis that supported the GESSAR II PRA arguments. Since the frequency of these transients has a small effect on the frequency of core damage, as well as on the relative ranking of the various accident sequences, the decision was made not to allocate substantial review resources to pursue this point further.

If the GE values were to be used in all these three areas (paragraphs a, b, and c above), the staff's estimate of frequency of core damage would decrease by 55% to the value of approximately 1×10^{-5} for the MAAC site, and by a comparable amount for the national average site.

(d) Containment event tree - loss of decay heat removal: In the GESSAR II PRA, the core-damage frequency for sequences with successful injection but with loss of decay heat removal - Class 2 sequences: CT2T, CT2L, CT2A - is calculated in two stages. The first stage evaluates the failure of the decay heat removal system from the onset of the transient or LOCA up to the time the containment pressure and temperature are about 29 psig and 250°F, respectively. The second stage, utilizing containment event trees, continues the evaluation of the sequences up to containment failure and core damage. This stage calculates the probability of RHR recovery from the endpoint of the first stage (containment pressure of 29 psig) up to containment failure (58 psig), and considered the failure probability of core injection systems under the adverse containment conditions. Tables 15.1a and 1b present the accident sequence frequencies (first stage in Tables 15.1a and 1b) and the core-damage frequencies (second stage in Table 15.2) for the Class 2 sequences as calculated in the GESSAR II PRA. The impact of the containment event trees analysis given by the ratios between the first and second stage (reduction factor) is also presented in Table 15.2.

The review of the Class 2 containment event trees by the staff/BNL identified three areas that are different from the GE evaluation.

The first area relates to the time at which the containment will fail. This time directly affects the recovery of the containment heat removal systems beyond the first-stage period.

The second area involves the operability of various instrumentation given the fact that the containment temperature is already in excess of 250°F for an extended period of time. No detailed information is available to allow a more realistic modeling of the instrument failures.

The third area of disagreement relates to the ability of the operator to ensure core injection and coolant makeup using the control rod drive (CRD) and condensate booster pumps. On the one hand, there is ample time (order of hours) to assess and arrest the situation. Additional technical personnel will be on site to assist the operator to perform his task. On the other hand, the lack of procedures in dealing with these situations tends to provide less confidence in the operator's ability to succeed. Also, in some of the CT2T sequences initiated by a LOOP (T_EW), a failure to recover the offsite power disables the CRD and condensate booster pumps.

Table 15.2 shows that the re-evaluation of the containment event trees increased the core-damage frequencies by factors of 6.5, 3.2, and 2 for Classes CT2T, CT2L, and CT2A, respectively. These changes are reflected in the revised core-damage frequencies for dominant sequences given in Table 15.1a.

A staff estimate of the frequency of core damage from Class 2 transients yields a value of about 4×10^{-6}. These changes are reflected in staff/BNL revised core-damage frequencies for dominant sequences.

The staff will complete its evaluation in this area. If preliminary evaluations are revised, the staff will present modified sequence frequencies in a future SER supplement.

A discussion of each dominant accident sequence follows; more details can be found in the BNL draft NUREG/CR, "A Review of BWR/6 Standard Plant Probabilistic Risk Assessment," Vol. 1.

15.6.2.2 Description of Sequences

Loss of Offsite Power: T_EUV, T_EW

Loss of all offsite power (LOOP) followed by failure of the onsite emergency diesel generators (station blackout) constitutes the dominant contributing factor to the frequency of core damage. On the basis of the national average LOOP frequencies, sequence T_EUV (primarily station blackout) has a frequency of 3.0×10^{-5} per reactor year, and contributes 79% of the total core-damage frequency for internal events.

Upon the loss of offsite power, the feedwater, power conversion, and condensate injection systems become unavailable because those systems cannot be powered by the emergency ac system (diesel generators). At this point, core cooling could be provided by HPCS, RCIC, or, if the reactor pressure vessel (RPV) is depressurized, the low-pressure coolant-injection systems. However, for the T_EUV sequence, all three diesel generator sets have failed because of random or common-cause effects. In this situation, the HPCS and low-pressure injection systems are unavailable, but the core can continue to be cooled by the steam turbine RCIC system utilizing battery dc control power. With the RCIC operating as described above, core flow is provided by natural circulation and excess reactor vessel pressure is relieved through the safety/relief valves into the suppression pool. The GE evaluation assumed that dc battery capacity and RCIC room heat loads were sufficient to enable operation of the RCIC system for 2 hours. Although no heat removal is available from the suppression pool, the large water volume provides sufficient heat sink capacity during this time period to accommodate core-decay heat. If offsite or onsite ac power is not restored in sufficient time, the RCIC system becomes unavailable when dc control power is depleted, or room temperature exceeds allowable limits. Without further injection, the level of water in the vessel will drop until the core uncovers, and the core is damaged.

Subsequent to the PRA submittal, GE has provided additional information claiming that with procedural modifications and minor design improvements, it is possible to provide core injection for 10 hours or more following loss of all ac power. Assuming, however, that the GESSAR II design does incorporate 10-hour station blackout capability, the frequency of core damage for the T_EUV sequence is estimated to be approximately 1.0×10^{-5} per reactor year (for the national average LOOP frequency). These are maximum reduction estimates not reflecting impacts of human error for successful operation beyond 2 hours.

The T_EUV sequence (with 10-hour station blackout capabilities) is reduced by a factor of 3, which equates to almost a 50% reduction in total core-damage frequency from internal events.

In addition to improvements in dc power capability, GE has proposed a diverse core-injection system called the ultimate plant protection system (UPPS). This system would utilize the plant fire system or fire trucks to provide makeup water for extended periods of time, assuming no ac or dc power was available. The staff has not quantified the impact of this system on core-damage frequency, however, it would likely have a significant impact on the station blackout sequence, which is the major internal contributor to core damage at GESSAR II.

An additional LOOP transient sequence also makes a contribution to the plant's core-damage frequency. Sequence T_EW involves LOOP followed by successful injection and failure of containment heat removal and accounts for approximately 4.2% of total core damage.

Isolation Transients (non-ATWS): T_FQW, T_FQUX

These events are initiated by isolation of the reactor from the turbine (T_F). In the first sequence, isolation is followed by failure of the feedwater system (Q) and failure of the containment heat-removal function including RHR and power conversion system (W). In the PRA, this sequence contributed an insignificant amount to core damage. The GE analysis concluded that 36 hours were available to restore containment cooling, and utilized a containment failure event tree to calculate a greatly reduced core-damage frequency for this event. These assumptions have not been used on PRAs for similar plants, which assumed core damage would occur in 24 hours. The staff/BNL concerns have been identified in the areas of containment failure timing, operator actions, and critical instrument survivability. Specifics of staff/BNL-GE differences are described in Section 15.6.2.1(2)(d) above. The reassessment of this sequence has resulted in a core-damage frequency of 1.9×10^{-6} per year. This makes T_FQW the second leading dominant sequence at approximately 5%, which is comparable with results from similar plants. Even with the large increase in sequence frequency the absolute magnitude of the sequence is quite small.

In the second dominant isolation transient, isolation (T_F) is followed by failure of the feedwater system (Q), failure of HPCS and RCIC coolant injection (U), and failure of timely automatic depressurization system (ADS) actuation (X).

Without ADS, the RPV cannot be depressurized to allow core cooling through use of the low-pressure systems. This sequence was calculated to be 5.3×10^{-7} per reactor year, which contributes approximately 1.4% to the modified core-damage frequency.

Anticipated Transients Without Scram (ATWS): $T_FC_MU_HP_A$, $T_FC_ML_H$, $T_FC_MC_1P_A$, $T_FC_MUP_A$, $T_FC_MP_1U_MP_A$, $T_FC_MC_1C_{21}$, T_FC_MU

Anticipated transients without scram (ATWS) as a group constitute about 9% of the modified staff/BNL CDF values. The GESSAR II PRA reported ATWS contribution as approximately 1% of the initial core-damage frequency ($\sim 4 \times 10^{-6}$ per reactor year). Staff/BNL review resulted in a significantly increased ATWS core-damage frequency. This difference is primarily attributed to differences in assumed probability of mechanical and electrical scram failure.

An ATWS occurs when a nuclear power plant experiences some upset or transient requiring prompt shutdown, but the scram system malfunctions. The seven dominant ATWS sequences shown above constitute the events contributing approximatel

7% of the plant's modified core-damage frequency. Each dominant sequence is initiated by transient T_F, reactor isolation. The top two dominant ATWS sequences consist of (1) those caused by a transient (T_F) followed by mechanical/hydraulic failure of the control rods to insert (C_M) coupled with a failure of the high-pressure injection system (U_H) and failure to inhibit depressurization (P_A)--$T_F C_M U_H P_A$, or (2) failure to control the vessel level (L_H), without successful high-pressure injection--$T_F C_M L_H$. Staff/BNL estimate the core-damage frequencies for the $T_F C_M U_H P_A$ and $T_F C_M L_H$ sequences to be 8.7×10^{-7} and 7.6×10^{-7} per reactor year, respectively, providing about 50% of the total ATWS contribution to the CDF. A number of other ATWS sequences provide the remaining contribution (see Table 15.1a).

The total frequencies presented here are higher than GE's values by more than 2 orders of magnitude. The major discrepancy between the GESSAR II PRA and the staff/BNL review lies with the different probability for the mechanical/hydraulic failure to scram (C_M) used by the staff/BNL, as discussed previously.

The ATWS results presented here reflect improvements and modifications known as the ATWS Resolution Alternative 3a, developed by the staff and incorporated by GE into the GESSAR II design.

The ATWS Resolution Alternative 3a modifications include:

(1) Alternate rod insertion (ARI) function--A system that is diverse and independent from the reactor protection system, meeting IEEE-279 and acting as backup to the electrical portion of the current scram system.

(2) Recirculation pump trip function.

(3) Feedwater control system runback function--Changes in logic to reduce vessel isolation events and permit feedwater runback.

(4) Automatic 86-gpm standby liquid control system (SLCS)--Modified SLCS piping to assure delivery of 86 gpm of poison and automatic actuation circuitry with reliability equivalent to the mechanical portion of the SLCS.

(5) Containment isolation - Closure of containment isolation valves upon occurrence of high containment radiation.

(6) Redundant reactivity control system - To provide automatic initiation signal to bring reactor to subcritical state.

(7) Scram discharge volume modifications:

    (a) Increased number of scram level sensors (with design diversity),
    (b) Nonsubmerged vent line for scram discharge volume,
    (c) Vacuum breakers and redundant vent valves on vent line,
    (d) Redundant scram discharge volume (SDV) isolation drain valve,
    (e) Repiping of scram instrument lines.

Without these modifications, the contribution to the total CDF by ATWS events would have been significantly increased.

Loss of DC Power: $T_{DC}UV$

The staff/BNL review of the GESSAR II PRA identified one additional accident sequence beyond those identified by GE, the loss of dc power initiated by a common failure of two dc busses followed by failures of the high-pressure systems (U). The low-pressure injection system (V) is unavailable because of the loss-of-dc-power initiation event. This sequence contributes about 1% of the total core-damage frequency. Failure of the two busses is attributable to operational or test and maintenance errors which propagate to system failure. A frequency of occurrence of 6×10^{-5} per year was chosen for this initiator on the basis of information provided in NUREG-0666, "Probabilistic Safety Analysis of DC Power Supply Requirements for NPPS." It was assumed that if the two dc busses were lost, the reactor would scram as a result of loss of feedwater. It should be noted that a number of assumptions were made in the development of the event tree for this analysis. These were: (1) failure probability of two dc busses, (2) the operability of the RCIC system without dc power, (3) the validity of a 19-hour mean time to repair (MTTR) of a dc bus, and (4) the failure probability of 0.01 of the RHR given failure of the dc busses. Since this analysis was not submitted by GE, the staff/BNL made the above estimates in the evaluation. The uncertainty associated with this sequence quantification is assumed to be large because of substantial judgmental input in the above assumed probabilities. Nevertheless, this scoping analysis indicates this sequence would not likely be a significant contributor to core-damage frequency.

Inadvertent Opening of Relief Valve: $T_{I}UX$

This sequence involves an inadvertent opening of a relief valve, followed by failure of the HPCS and RCIC coolant injection (U) and failure of timely ADS operation. This sequence contributes approximately 0.3% to the core-damage frequency for internal events.

Inadvertent opening of a relief valve results in loss of reactor coolant into the suppression pool. If the valve remains open, it is necessary to provide makeup to the RCS to prevent core uncovery and damage. The preferred way of accomplishing this is through the high-pressure systems (RCIC or HPCS). If these systems are not available, injection can be provided through low-pressure systems (core spray or coolant injection) following operation of the ADS to reduce reactor pressure. Failure of the high-pressure systems and failure of timely actuation of the ADS described above results in continued discharge of water from the RCS into the suppression pool, with no means to provide makeup. The results of the staff/BNL review supported the conclusion that this event is a very small element of the total core-damage frequency.

15.6.2.3 External Events

The initial GESSAR II PRA submittal included only internal events. Subsequently, at the staff's request, an external-events analysis was provided incorporating seismic issues, fire, and internal flooding. GE concluded that there were no significant risk impacts from these areas. The staff is still assessing the seismic submittal. The staff findings on external events are summarized below.

(1) Seismic

GE assessed the risk attributable to seismic initiating events and presented this assessment to the staff. GE stated that the purpose of the assessment is: (a) to demonstrate the capability of the GESSAR II design to accommodate a low-probability seismic event beyond the safe shutdown earthquake (SSE) design basis, and (b) to meet the intent of the NRC draft policy statement on severe accidents. GE has submitted a representative hazard function to model the seismic acceleration for a generalized site location. They have also presented component and structure fragility values to model the ability of structures and components to withstand seismic events.

The staff has not yet completed its review of the hazard function or fragility analysis. Additionally, the staff is evaluating the seismic systems analysis to determine if all important seismically induced failure modes have been considered. At this time the staff is not able to validate GE's claim for low seismic risk for GESSAR II. Staff findings in this area will be presented in a future supplement to the SER.

(2) Fire Events

GE performed a fire probabilistic risk analysis of six critical fire locations which were identified by a screening process. For each of these locations, a frequency of occurrence of fires was established using recent data. These fire frequencies serve as initiation probabilities for event trees characterizing the fire sequences. Fire growth times were obtained from experiential data, a deterministic fire model computer code (COMPBRN), and engineering judgment. Suppression times were taken from a distribution of reported fires. Next, the fire-induced accident sequences and the core-damage contribution for each critical location were quantified. The total fire contribution to the frequency of core damage was then determined. The release frequency was then evaluated by linking the fire-event sequences to the internal-event PRA and the fire-induced plant risk was determined. However, it is the staff's judgment that the fire-risk analysis and documentation do not meet the expectation of what a PRA fire-risk assessment should entail. The staff's major concerns are:

(a) All critical areas have not been fully identified.

(b) Equipment affected by fire which may either cause an initiating event or hamper the performance of accident-mitigating systems (even during a so-called stage-1 fire growth) has not been sufficiently addressed.

(c) Fire-frequency estimates still remain questionable.

(d) The fire scenarios analyzed do not, in some respects, correspond to the physical layout of the fire area.

Therefore, by present standards the staff finds the subject document incomplete when compared with other fire-related external-event safety studies.

As a result of these concerns it is not possible to make any definitive quantitative judgments or appraisals on the relative degree of fire risk associated with the GESSAR II design. These concerns, coupled with the prevailing large uncertainties in fire-risk analysis, preclude making any substantive judgments or appraisals of the analysis.

It is difficult to accurately quantify the fire risk for GESSAR II plants. Newer plants and plant designs which comply with current fire-safety standards and which are, perceptibly, more fire safe on a relative basis when compared with older plant designs may not have the more "obvious" fire hazards and potential fire progagation scenarios that have been identified in previous fire-risk studies on the older plants. Accordingly, for these newer plants to assess fire risk would require more detailed analysis and, in some respects, would require analyses that are beyond the state of the art. This is largely due to the fact that major fire contributors have been "designed out" of the plant and, therefore, for one to investigate how fires contribute to the risk-based measures, other nuances or aspects associated with the phenomena of fire would have to be included.

A case in point is barrier effectiveness. In other probabilistic studies dealing with fire risk, this issue was addressed but not pursued further since the change in core-damage frequency would be judgmentally smaller, relative to the fire-related contributors to core damage that had already been calculated for single enclosure fires. The staff believes this to be largely the case also for the GESSAR II design since those other potential contributors to fire and its subsequent propagation and attendant effects on the plant's safety function would also be small.

It is the judgment of staff/BNL that the contribution to core-damage frequency resulting from fires within a GESSAR II plant appears to be small. This perception is primarily due to the GESSAR II design's close adherence to those deterministic fire-protection guidelines and requirements found in Section 9.5.1 of the Standard Review Plan (SRP), especially the design features that can be accommodated in a new plant, such as separation by 3-hour fire barriers.

Thus, in conclusion, the staff finds that:

(a) The GESSAR II risk related to fires appears to be rather low, based on the staff's deterministic analysis.

(b) The GE probabilistic fire-risk analysis in the GESSAR II study does not adequately make the case for this assessment.

(c) In order to appropriately assess the low fire risks at GESSAR II plants, one would be required to perform more detailed analyses such as barrier effectiveness, probability of penetration installation deficiencies, ventilation, and smoke toxicity effects. These analyses, in some respects, are beyond the state of the art or entail a level of analytical detail that has not appeared in other safety studies. Such studies are not needed to assess the licensability of GESSAR II plants.

(3) <u>Internal Flooding</u>

The GE study considered two types of potential flood sources for GESSAR II: cracking or rupture of pipes or water containers, and leakage of seals or leakage past glands. The study evaluated the possible ways that an internal flood can initiate severe accidents. Several locations that are susceptible to internal flooding were identified: the containment building, the drywell, the auxiliary building, the fuel building, the radwaste building, the control building, the diesel generator building, and the turbine building. A screening analysis was performed for these areas based on either the internal flood-initiating frequency of the individual area or the judgmental evaluation of the likelihood of an internal flood leading to a core-damage accident. Flood-initiating frequency for these buildings was evaluated for pipe cracking or rupture of water container. The GE study deemed the leakage from seals and glands to be small.

The result of the screening analysis indicated that two areas may have a greater potential for flood-induced core-damage events. These areas are the diesel generator building and the turbine building. In the GESSAR II analysis, it was conservatively assumed that flooding these buildings would mean all safety-related equipment in them would be unavailable. Two functional event trees were developed to model the progression of the flooding-accident sequences. The frequency of total core damage attributable to internal flooding was calculated in the GESSAR II PRA to be 6.4×10^{-9} which is an insignificant contribution to the frequency of core damage from internal events. There are two criteria that a flooding event has to satisfy before it will eventually develop into a credible flood accident sequence. One of the criteria is that there has to be a large enough water source to cause a potential threat of flooding safety equipment. This is called the flood precursor. Given the occurrence of a flood precursor, a reactor shutdown will be initiated; this may either be in the form of a plant transient or a controlled manual shutdown. At this point, a flood-initiating event has occurred. With a flood-initiating event, various safety systems will then be challenged and combinations of these system failures will result in different core-damage sequences. The other criterion is the time required to arrest a flood. This criterion directly affects whether a flooding precursor would eventually develop into a flood-initiating event. For instance, if flooding occurs in the RCIC room and the flow rate is relatively small, this would imply that ample time is available for the operator to identify and arrest the flood before critical equipment is endangered. This increased probability of a successful operator action would prevent the flood precursor from developing into an initiating event and, hence, from challenging the safety systems

In the review of the GESSAR II flood analysis, these criteria were used to help focus the concerns. In general it was found that the GESSAR II analysis is rather limited in terms of its depth and the effort devoted to address the question of flood-initiating events or their impacts. Part of the reason is believed to be the lack of detailed information on the location of equipment. The analysis relies heavily on what is presented in GESSAR II and what was reported in two other flood PRAs. Owing to the deterministic approach used in GESSAR II, it is limited in scope in analyzing all potentially severe flooding accidents.

In the area of the flood precursor, the staff/BNL finds that the frequency reported in the GESSAR II flood study for the different areas is not consistent with those of other PRAs. For instance, in the evaluation of the containment building, the GESSAR II analysis considered a crack in the 24-in. suppression pool makeup line. A flow rate of 160 gpm was assumed and a 30-hour period was calculated as necessary to drain 232,000 gal of water. The 160-gpm flow rate was small in comparison to what other PRAs have assumed. A higher flow rate reduces the time that is available to the operator to arrest the flood. Furthermore, no distinction was made in the GESSAR II analysis between maintenance-induced floods and rupture-induced floods. An assessment of the precursor frequency by the staff/BNL is presented in Table 15.3. The flood-precursor frequency was calculated using a Markovian model and it is divided into two groups. The first group represents the maintenance-related floods. The calculation considered the frequency of online maintenance practices and the probability of inadvertent opening of isolation valves which results in a flooding precursor. The frequency was found to range from 1.5×10^{-4} to 2.0×10^{-5} for the different systems.

The second group of the precursor frequency relates to rupture-induced floods. The components that were considered in the staff/BNL evaluation included valves, pipes, and pumps. Since a rupture is not normally detected until the system is in demand or in test, the precursor frequency was calculated for three types of plant challenges, namely, turbine trip, MSIV closure, and manual shutdown. The frequency varies between 2×10^{-4} and 2×10^{-5}.

Given the occurrence of a flood precursor, credit should be allowed for the operator to respond to the flood annunciators and to follow procedures to determine the location of the flood and to arrest the flood. The time allowed for these operator actions depends greatly on the flooding rate within the particular building or compartment of interest and the locations of the critical components. These parameters are plant specific and some of them cannot be readily derived based on the GESSAR II design. However, if one assumes that the human failure probability to diagnose and arrest flood is 0.5, then the flood-initiating frequency is on the order of 5×10^{-4}. There are, however, considerable uncertainties in potential flood-initiating frequencies, and the location of essential equipment. It is also not possible to determine with accuracy the conditional probability of core damage given a flood, because of the above limitation. Since the GESSAR II design calls for individual compartments for different safety equipment, the potential for common-mode failure of redundant dependent systems is believed to be low. On these bases, the staff judges that the flood contribution to core damage is probably small as compared with the contribution of other initiating events.

In conclusion, staff review of the GESSAR II flood analysis indicates that the study may not be representative of the GESSAR II design, since the detailed information needed is yet to be developed. The study did not address the different flood-related topics in any depth. However, an estimate of the contribution to core damage from flood events based on reassessed flood-initiating frequency and the conditional core-damage frequency from other internal events suggests that it would not be significant as compared with other initiating events. It is required that the flooding issues be reevaluated when future applicants reference GESSAR II. This would allow a location-specific flooding analysis to be performed, to account for the uncertainties discussed previously in conditional core-damage estimates. The evaluation should consider that rupture of lines to the suppression pool has the potential for removal of water from the pathway of the release of fission products and hence might produce a release category more severe than any considered for internal events. The impact of this evaluation on plant risk should be addressed.

(4) Other External Events

The only external events treated to a quantitative analysis in the PRA were those previously discussed: seismic, fire, and internal flooding. However, there are other external events which would potentially impact the plant such as hurricanes, tornadoes, external floods, aircraft strike, hazardous materials, and snow and ice loading. GE conducted brief qualitative assessment of these other external events to evaluate their potential risk impact. The PRA Procedures Guide (NUREG/CR-2300) provided guidance in the evaluation, suggesting screening criteria for the inclusion of external events into PRA studies. An external event is excluded from PRA studies (a) if it is included in the definition of another event or events, (b) if the event can be shown not to occur close enough to the plant to affect it, (c) if the event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events, or (d) if the event is of equal or less damage potential than the events for which the plant has been designed. These screening criteria have been utilized in these qualitative assessments. Each event is discussed below.

Hurricanes

In Table 10-1 of ASCE Paper No. 3269 (American Society of Civil Engineers, 1961), hurricane events are considered to be included under external flooding and the wind forces are covered under extreme winds and tornadoes. Thus hurricanes are addressed in GESSAR II, Sections 2.2 (Tornadoes) and 2.3 (External Floods). This follows the suggestions of NUREG/CR-2300 to exclude treatment of a specific event that can be included in the definition of another event or events.

Tornadoes

In Section 2 of the GESSAR II SER (Site Characteristics), the discussion of regional meteorological conditions for design and operating bases contains the specific considerations for high winds and tornadoes. These include:

(1) The structures are designed to withstand wind velocities of 130 mph at 30 ft above plant grade with a velocity distribution and gust factor as described in ASCE Paper No. 3269.

(2) The safety-related structures and equipment are designed for the design-basis tornado described in NRC Regulatory Guide 1.76 for Region I. The characteristics of this tornado are:

Maximum wind speed (mph)	360
Rotational speed (mph)	290
Translational speed:	
Maximum (mph)	70
Minimum (mph)	5
Radius of maximum rotational speed (ft)	150
Pressure drop (psi)	3.0
Rate of pressure drop (psi/sec)	2.0

Using these conditions the wind velocity probability for γ = 360 mph is estimated to be $\sim 10^{-5}$ (GE letter, Nov. 7, 1983).

Additional information on wind and tornado loadings is contained in the GESSAR II FSAR, Section 3.3. The design of structures to withstand the design-basis tornado or an operating-basis wind of 130 mph is expected to result in a low-risk contribution from these events. However, the staff is concerned that the failure of nonsafety structures has a potential to damage safety-related components. Utility applicants who reference GESSAR II will be required to demonstrate that the risk resulting from the failure of nonsafety structures is low.

External Floods

Section 2.4.2.1 of GESSAR II and Section 2.2 of the staff's SER (NUREG-0979) commit the applicant to provide site-specific flood data, including the date, level, peak discharge, and related information for major flood events in the region where a GESSAR II plant is to be sited.

Included in the GESSAR II plant design is consideration of the probable maximum flood potential. Seismic Category I structures that may be affected by floods are designed to withstand floods using the "hardened" flood-protection approach. Through the hardened protection approach, structural provisions are incorporated in the plant's design to protect safety-related structures, systems, and components from postulated flooding. Seismic Category I structures required for safe shutdown remain accessible during all flood conditions.

Safety-related systems and components are protected from flooding either because of their location above the design-flood level, or because they are enclosed in reinforced-concrete seismic Category I structures. The flood protection requirements are addressed in Section 2.2 of the staff's SER.

Additional flood protection from external sources is discussed in Subsection 3.4.1 of GESSAR II.

Structures of safety significance are designed for a design-basis flood, as defined in Regulatory Guide 1.59, up to an elevation 1 ft above plant grade including allowance for the effects of coincident waves and the resultant runup as calculated from site-unique parameters.

On the basis of information presented in GESSAR II, and the results obtained from the evaluation of internal flood events (GE letter, Nov. 7, 1983), the probability of core damage from external flood sources is believed to be a small contributor to core-damage risk. However, because of wide uncertainties in site-specific flooding potential, utility applicants referencing GESSAR II will be required to demonstrate that the risk from external flooding is low.

Aircraft Strike

The nuclear island is intended for use at sites where the probability of an aircraft impact is $<10^{-7}$ per year. It is the responsibility of a utility applicant to show compliance with this requirement. If the applicant's plant is located at a site where this probability is not $<10^{-7}$ per year, the applicant will be required to provide an evaluation of the consequences, as they relate to the PRA evaluation, of an aircraft crash considering the frequency and type of flights of aircraft germane to the site.

Hazardous Materials

In assessing the risk from hazardous materials, NUREG/CR-2300 suggests including the risks from industrial or military facilities, pipeline accidents, release of chemicals in onsite storage, and transportation accidents. All of these are site dependent, and require site-specific information to quantify any potential impact. Utility applicants referencing GESSAR II will provide the information to demonstrate that the risk from hazardous materials is low.

Utility applicants will provide a determination of design-basis events (i.e., probability of occurrence >10^{-7} per year and potential consequences serious enough to affect the safety of the plant to the extent that 10 CFR 100 guidelines could be exceeded) for each of the following accident categories: explosions, flammable vapor clouds (delayed ignition), toxic chemicals, fires, collisions with intake structures, and liquid spills.

Snow and Ice Loading

Structural integrity of safety-related buildings can be impacted by snow and ice buildup. However, GE did not address the impact of snow and ice load on structures in the GESSAR II PRA. The GESSAR II design value of snow load (50 lb/ft^2) may be exceeded at some locations in the northern United States; therefore, a utility applicant referencing GESSAR II must assess the risk impact from snow and ice in the plant-specific FSAR.

15.6.2.4 Effects of Uncertainties

Like any other probabilistic analyses, the GESSAR II PRA contains large uncertainties. These can be grouped into four general areas: statistical, modeling (assumptions such as human error, common cause models, and others), omissions, and computational. Each of these types of uncertainties is applicable to the various PRA segments such as the core-damage sequence estimates, the containment analysis, the source term, and the site/consequence analysis. An excellent discussion on the subject of uncertainty which pertains directly to this review can be found in "Evaluation of Risk Estimates," in Section II.A of the Indian Point ASLB "Recommendations to the Commission" (ASLB, October 24, 1983). The Indian Point ASLB points to two major omissions (sabotage and equipment aging) in the Indian Point PRA which may cause the risk estimates to be low. These omissions apply to GESSAR II plants as well. The staff/BNL review of the GESSAR II PRA points out the tendency for modeling assumptions to be conservative in some areas and nonconservative in other areas. The overall significance is not known. Systems-modeling assumptions exist in the values of various input parameters such as hardware failure data, human error data, frequency of accident initiators (especially loss of offsite power), large and medium LOCAs, fires, and seismic events. These sources of uncertainties are still dominant sources because of (1) a relatively sparse data base on severe earthquakes in the Eastern United States, (2) inadequacies in quantifying certain human errors during accident scenarios, and (3) no data on large LOCAs. Since the LOCA contribution is so minor for the GESSAR II design, the impact of uncertainties would be very small. There is also a large uncertainty attributed to varying degrees of systems success/failure modeling assumptions, completeness, and statistical and arithmetic errors. Since GESSAR II is a standard plant design as yet unsited, there are additional uncertainties regarding external event hazards and effects of plant interfaces which could introduce adverse interactions. These are not specifically quantified but must be recognized.

The staff/BNL point estimates of core-damage frequency and their associated uncertainties for the MAAC site and the national average site were calculated. For the MAAC site, the core-damage frequency (CDF) for internal events ranges from 3.7×10^{-6} for the 5% confidence limit to 6.0×10^{-5} for the 95% confidence limit with an error factor of 4. For the national average site, on the other hand, the CDF ranges from 6.8×10^{-6} for the 5% confidence limit to 1.1×10^{-4} for the 95% confidence limit. However, this range only includes a limited consideration of statistical uncertainties. Recognizing the other sources of uncertainties, such as completeness, it is the staff's judgment that uncertainty estimates should not lead to a core-damage frequency much larger than a factor of 10 above the reported point estimate. This is especially true with the inclusion of the ultimate plant protection system in the GESSAR II design.

15.6.2.5 GESSAR II Risk Findings

(1) Risk Overview

Fission products may be released into the containment building during a core-meltdown accident, and a number of systems are available to help contain or mitigate this radionuclide release. If these systems fail or are compromised, a fraction of the radionuclides may be released to the atmosphere with corresponding adverse effects upon the surrounding environment. These effects or consequences can be measured in terms of health effects on the surrounding population and property damage in the surrounding area (damage indices). A number of these damage indices (e.g., early fatalities and injuries) are short term (within months of the accident) and require a dose that exceeds certain thresholds. For example, a dose of 320 rem to the blood-forming organs, with supportive medical treatment (175 rem without treatment), will result in an early fatality in a small fraction of the population receiving such a dose. These damage indices are a strong function of the timing and magnitude of the fission-product release and also of the emergency response of the population. Other damage indices such as latent fatalities and interdiction of crops or land are long term (50 years after the accident) and are measured over large distances (500 miles around the reactor site). These damage indices are a function of the type and amount of radionuclides released and are relatively insensitive to the emergency response of the population. It is, therefore, possible for the short-term and long-term damage indices to be dominated by quite different dominant accident sequences and failure modes. This must, therefore, be taken into account when discussing risk-dominant sequences.

In PRAs previously submitted to the NRC, the less probable but more severe accident categories usually dominated early fatalities (more severe implies early release, short warning time for evacuation, and large fraction of fission products released), whereas the more probable but less severe release categories dominated the long-term damage indices. This was largely because the less severe release categories were usually below the thresholds necessary to result in early fatalities. In this regard, the GESSAR II PRA is different from previously submitted PRAs because none of the release categories were calculated by General Electric to result in early fatalities. In Table 15.4 are reproduced long-term risk (person-rem) for each of the 15 release sequences and failure modes as reported in GESSAR II.

Table 15.5 groups the 15 release sequences into similar "events" and indicates that transients without suppression-pool bypass account for approximately 70% of the long-term damage indices. This "event" grouping covers a wide range of Class I accident (see Table 15.8) sequences and failure modes. However, the general characteristics of the fission product release path for all the failure modes within the "event" are similar. For this "event" the containment building is assumed to fail from a variety of hydrogen phenomena (global burns, local detonations, local burns, or standing flames), but the drywell wall and ceiling are assumed to remain intact. (See Figure 15.1 for a description of the Mark III containment.) Suppression-pool scrubbing results in significant reduction in the aerosol fission products. In general, the staff and its consultants agree that if the containment failure mode results in all of the released fission products passing though the suppression pool, then radioactivity associated with any aerosols carried by the steam/air/hydrogen mixture entering the pool will be substantially reduced. However, the staff considered a wider range of uncertainty associated with this failure mode (Section 15.6.2.5(4)) than did General Electric.

Table 15.4 also indicates that Class I transients with pool bypass to the vaporization release (limited bypass events in Table 15.5) account for 25% of the magnitude of the long-term damage indices. For this "event" GE assumes that a global detonation will occur in the wetwell and the shock wave will fail the drywell ceiling. Hence, a limited fraction of the fission products are assumed to bypass the suppression pool. However, by comparing the GE estimates of person-rem/event in Table 15.5, it will be noted that the limited bypass sequence is not significantly more severe than without bypass (in terms of offsite consequences: 70,000 vs. 52,000 person-rem/event). This is due to a number of factors. It takes time to generate hydrogen and reach global detonation limits in the wetwell. During this time, a significant fraction of the fission products would have been released from the fuel with much of the fission products either permanently retained in the primary system or scrubbed in the suppression pool. The global detonation is assumed to occur close to the time of reactor pressure vessel melt-through and GE assumes that the detonation shock wave fails the drywell head, thus draining water (from above the drywell head) into the drywell. The water floods the region under the reactor vessel and terminates any further release of fission products from the fuel ex-vessel (core-concrete interaction). Therefore, although this "event" results in suppression pool bypass, the fraction of fission products actually bypassing the pool is very limited.

dThe staff/BNL do not entirely agree with the way GE has analyzed the above "event" or fission-product-release path. These differences are discussed in greater detail in Section 15.6.2.5(4), but in summary, the staff/BNL are concerned with GE's assumption that the drywell head is the only possible failure location. Extensive analysis by the staff/BNL has lead to the consideration of the possibility of other failure locations in the drywell wall (BNL, May 1984; May 15, 1984). These other failure locations would allow fission products to bypass the suppression pool and also would not result in quenching of the core debris ex-vessel. Thus, fission-product release during core/concrete interactions would continue for these alternative failure locations and would result in a significantly greater fission-product release than GE calculated.

The GE PRA includes predictions of best-estimate plant risk levels for the 15 release sequences analyzed (Table 15.4). The staff has reported consequences for three sequences based on source terms that resulted from a parametric study. The reasons for the difference in procedure and the methods used in conducting the study are discussed in Section 15.6.2.5(6).

General Electric has calculated a best estimate of the risk to the public from internal events at what may be characterized as a "composite site," expressed as fatal latent cancers, to be 1.7 x 10^{-5} per reactor year. GE's 15 release categories have contributed to the total, each weighted by its appropriate probability. Because the staff is of the opinion that the current state of the art of source-term methodology allows only a range of source terms to be calculated at this time, the staff has not calculated a corresponding value. Staff calculations, however, currently indicated an overall core-damage frequency that is about a decade higher than the GE estimate. As far as consequences are concerned, since the lower range of release fractions calculated by the staff is comparable to GE's best estimate, the lower range of consequences would also be comparable. However, the high range of conditional consequences calculated by the staff is one to two decades higher than those calculated by GE. Therefore, the risk of latent fatality may range from 1 to 3 orders of magnitude larger than that calculated by GE. It is important to note that both estimates are significantly smaller than those shown in the Reactor Safety Study (NUREG-75/014, formerly WASH-1400), due both to the plant design and to changes in source-term methodology. The staff has calculated that there is an exceedingly small probability that early fatalities might result from releases at the high range for early containment failure times (short warning times for protective actions) and adverse meteorological conditions. These calculated values, again, are negligibly small compared with values in the Reactor Safety Study.

(2) Risk Insights

A major conclusion of the GESSAR II PRA is that for a wide range of potential severe accident sequences initiated by internal events, General Electric predicts that no early fatalities would occur, and the staff/BNL independent calculations predict an exceedingly small number for early fatalities only when a high value is used for the release. This is a very significant risk insight and depends on a number of important factors. These factors can be subdivided into two broad groups related to methodology and containment design. Each group is discussed in detail below. Because the wetwell is predicted to fail as a result of hydrogen phenomena, the focus of both the GE analysis and the staff/BNL review has been on the integrity of the drywell and fission-product evolution and transport within the plant. This emphasis is different from the emphasis in previous PRA analyses which focused on containment integrity.

The determination of fission product release and transport in the GESSAR II PRA is a significant departure from other recently published "industry" PRAs. In the past, PRAs have generally used the prescriptions in the Reactor Safety Study to determine the release of fission products from the fuel and the movement of these fission products throughout the primary system and containment building. However, there has been significant research activity in this area since the publication of the Reactor Safety Study in 1975. A basis for estimating fission product behavior (NUREG-0772) was published in 1981 by the NRC's Office of Research (RES). In addition, updated fission-product source-term prediction methods are currently being developed by the Accident Source Term Program Office (ASTPO, 1983) and are receiving extensive peer review. PRAs have generally recognized the potental influence of the new source-term methods, but the impact has been expressed only in the form of uncertainty (e.g., the level 2 risk curves (see PECO, 1982, CECO, 1981; PASNY, 1982)). The point estimate risk curves for the PRAs (PECO, 1982; CECO, 1981; PASNY, 1982) are based on the Reactor Safety Study prescription with regard to fission-product release from the fuel and transport in containment. The GESSAR II PRA departs from this practice and incorporates new factors to determine the source terms. These changes relative to the Reactor Safety Study are significant and some of the more important are:

- Fission product release rates are calculated as a function of core temperature.

- Permanent retention of some fission products in the primary system is presumed.

- High suppression pool decontamination factors are used, which are stated by GE to be based on experiments and modeling.

Taking credit for the above mechanisms tends to reduce the fission-product source terms relative to the releases that one would calculate using Reactor Safety Study methods. The use by General Electric of the above mechanisms in "best-estimate" calculations should, therefore, be clearly understood when comparing the GESSAR II PRA with PRAs previously submitted to the NRC in which Reactor Safety Study source terms were used to generate the best-estimate risk curves.

However, even allowing for differences in methodology, there are unique design features associated with the GESSAR II Mark III containment that help to mitigate the consequences of severe core-damage accidents. The staff agrees with General Electric that for most of the core-damage-accident sequences, drywell integrity will be maintained. For these sequences, the vast majority of the fission products released from the fuel will be directed through the suppression pool. In addition, for the highest frequency sequences (Class I), the suppression pool will be subcooled, resulting in the fission products being subjected to significant pool scrubbing. Even when the suppression pool is saturated (Class IV sequences) the NRC staff/consultant calculations indicate significant pool scrubbing. In summary, the combination of the ability of the Mark III containment to maintain drywell integrity and the fission-product scrubbing effectiveness of the suppression pool is a significant mitigation feature in terms of reducing offsite consequences of severe core-damage accidents.

(3) <u>Effects of Uncertainties on Risk Results</u>

When the GESSAR II PRA was originally submitted to the NRC on March 19, 1982, it presented best-estimate calculations for core-damage frequency, containment-failure-mode conditional probabilities, release fractions, and offsite consequences. Several subsequent submittals by General Electric presented the effects of uncertainties relative to the original best-estimate calculations. However, GE treated uncertainties associated with determining the frequency of a particular release category differently from uncertainties associated with the magnitude of the fission products released and, hence, offsite consequences. GE calculated uncertainties in the core-damage frequencies by propagating uncertainties in reliability data through fault and event trees. Uncertainties in accident phenomenology were considered in the context of a range study. These range studies are discussed further throughout this report. Further, there are uncertainties in the calculation of consequences, given a release. These uncertainties include those that result from simplified calculations of radionuclide transport from the plant to the receptor, lack of precise dosimetry, and statistical variations of health effects.

(4) Differences in GE Results and Staff Results

Containment Event Trees

The staff/BNL containment analysis is described in detail in Section 15.6.2.5(5) of this supplement and in BNL's draft NUREG/CR (May 1984). In this section major differences between the NRC/BNL assessment and the GE approach will be discussed. Although the GE containment event trees appear to be comprehensive in the selection of potential containment failure modes, the staff is not convinced that the more severe failure modes have been given sufficient importance. The appropriateness of the branch point split fractions is discussed in Section 15.6.2.5(5). However, the most obvious failure modes that have been omitted by GE are steam explosions, basemat penetration, and long-term bypass of the suppression pool. At this stage, the staff is reluctant to give these failure modes zero probability, as was done by GE. However, steam-explosion-induced failure of containment is now considered less probable (Section 15.6.2.5(5)) than it was in the Reactor Safety Study. The approach taken by GE in assuming containment failure via overpressurization before basemat penetration (and hence eliminating the latter as a potential failure mode) appears to be a conservative assumption in terms of health consequences. Finally, sequences that bypass containment have been given very low probability in the GE PRA. The failure to isolate containment is considered improbable, as is the potential for excessive leakage through the main steam isolation valves (MSIVs). These potential failure modes are addressed in greater detail in Section 15.6.2.5(5).

Loss-of-Offsite-Power Transients and Effects of Hydrogen

The most frequent accident sequences are the Class I transients. In the GESSAR II PRA, Class I transients initiated by a LOOP including diesel failure were represented by two separate containment event trees. The total probability associated with the CT1-P class was subdivided into the probability of restoring power within 60 minutes of the start of core damage ($CT1\text{-}P_a$) and the probability of restoring power after 60 minutes ($CT1\text{-}P_b$). Each of these event trees is discussed below. Differences between the staff/BNL and General Electric assessments of the $CT1\text{-}P_a$ tree relate primarily to differences in the assigned conditional probabilities of various hydrogen phenomena (global combustion or detonations vs. local combustion or detonations). General Electric allocated approximately 10% of the frequency of the $CT1\text{-}P_a$ to global detonations, which were assumed to fail the drywell head. The NRC/BNL assessment did not predict global detonations at all during the time frame associated with the $CT1\text{-}P_a$ tree and the staff, therefore, gave global detonations zero probability within 1 hour. General Electric allocated approximately 30% of the frequency of the $CT1\text{-}P_a$ to local detonations but concluded that such an event would not fail the drywell head or wall. The staff/BNL assessment predicted a lower conditional probability of a local detonation (approximately 7% of the total frequency) but concluded that such an event might fail the drywell head and possibly also the drywell wall. If the drywell head fails, water will drain into the drywell and GE assumes that this water will then flood and quench the core debris ex-vessel. This assumption results in the termination of the ex-vessel vaporization release. A failure in the drywell wall allows fission products to bypass the suppression pool; also, the core debris would not be flooded and quenched ex-vessel. The impact on risk of the staff/BNL assumption was described in Section 15.6.2.5(1). Leakage from the drywell via previously existing small leak paths, as well as MSIV leakage, has a high probability. These leak paths by themselves, however, do not constitute a major contribution to severe-accident risk because of the relatively small mass which can be released via these pathways. Therefore, these leakage pathways are not treated as separate branches on the containment event trees, but are included in the assessment of the drywell/suppression-pool fission-product attenuation effectiveness.

Fission-Product Release

In the following sections, staff/BNL predictions of fission-product release during severe accidents are compared with GE predictions. The comparisons are made for various phases of accident progression, the classes having been chosen from the dominant core-damage sequences. Staff predictions span a range of values for fission-product releases. GE has performed sensitivity studies and provided a range and "best estimate" release (GE letter, May 17, 1984). The staff/BNL calculations are described in detail in Section 15.6.2.5(7) of this supplement.

In-Vessel Release Phase

Both General Electric and the staff considered two release mechanisms during in-vessel core degradation. The first release mechanism includes core heatup and degradation prior to core slump. This mechanism was traditionally referred to as the "melt" and "gap" release phases. The second release phase occurs after core slump begins, as the core is melting through the vessel head. The first release mechanism is terminated at the time of core slump because the core is assumed to slump into water present in the bottom of the vessel. This water evaporates and cools the core. The core heats up for a second time prior to head failure and during this time additional fission products are assumed to be released from the fuel.

There is close agreement between the staff and GE fission-product-release factors for the first in-vessel release mechanism (prior to core slump). There is even better agreement for the more volatile species. The staff and GE do not agree as well for the second release phase (in-vessel heatup following core slump). However, the total in-vessel fission-product release is dominated by the first mechanism and thus differences during the second in-vessel heatup are not crucial. Consequently, the staff and GE values of total in-vessel fission-product releases are near agreement.

In-Vessel Retention

In the GE sensitivity study, the fraction of fission products retained in vessel was assumed to be either 0.0 or 0.95. The same retention factors were applied to all the fission-product species except noble gases and organic iodine, which were not retained. In the staff study, different ranges were assigned to the iodine, cesium, tellurium, and aerosol groups (the noble gases and organic iodine were not retained). See Section 15.6.2.5(6).

Re-Emission of Retained Fission Products

In the staff analysis, an attempt was made to gauge the effect of re-emitting a fraction of the fission products initially held up in the primary system. The re-emission would be caused by the post-release heating by fission products deposited in the primary system. GE did not consider this effect in its analysis. In the staff analysis iodine, cesium, and tellurium were assumed to be re-emitted during the same time as the ex-vessel release. The remaining fission-product species were assumed not to be re-emitted. See Section 15.6.2.5(6).

Ex-Vessel Vaporization Release

The factors used by the staff to describe the ex-vessel vaporization release phase were based on the QUEST study (Quantitative Uncertainty Estimation of Source Terms), a program to estimate the uncertainty in selected specific radiological source terms being performed by Sandia National Laboratory for the NRC (Sandia, March 1984). QUEST is the only study that estimates uncertainties in a comprehensive manner. For the less-volatile species of fission products, the range of releases estimated in the QUEST study spans many orders of magnitude; for the volatile species, any remaining inventory of fission products which was not emitted during the in-vessel release phase is simply assumed to be emitted during the ex-vessel phase. Thus, in the staff's analysis a unit release fraction was assumed for the noble gases, iodine and cesium, but the high release fraction from the QUEST range was used for the other fission-product species. In the GE analysis, a much narrower range was considered for this phase of the accident. GE started with the Reactor Safety Study vaporization source term and varied it by a factor of 2. The implication of varying the release during this phase of the accident has a very direct bearing on all sequences with suppression-pool bypass.

Suppression-Pool Decontamination Factors

In the staff study it was assumed that the maximum suppression pool DFs are given by the GE values. GE reduced the suppression-pool DFs by one order of magnitude in its sensitivity study.

The NRC DFs were based, in part, on an evaluation of differing conditions for different times in the accident sequences, and, in part, account for pre-existing bypass paths as discussed in Section 15.6.2.5(7).

See Table 15.6 for a comparison of staff and General Electric DF values.

(5) Containment Analysis

The staff/BNL analysis of GESSAR II containment design is presented in the following sections.

Drywell and Containment Failure Modes

Containment event trees are used to relate a given accident class to a number of potential containment failure modes or release paths. Before proceeding, it is useful to first describe the process of "binning" as it relates to the staff/BNL containment failure analysis.

The process of binning is a way of reducing a large number of accident sequences into a smaller number of "representative" sequences or classes that can be analyzed in detail to determine potential containment building failure modes. Each of these failure modes will have unique fission-product-release characteristics. It is intended that the failure modes and fission-product-release characteristics associated with a particular accident class will be representative of the many individual accident sequences binned into the class. The accident classes consist of accident sequences with similar parameters.

In Table 15.7 the frequencies of the various accident classes as reported in GESSAR II are compared with the values suggested by the NRC staff and their contractors at Brookhaven National Laboratory. From an inspection of Table 15.7, it is apparent that ten accident classes were identified in GESSAR II (e.g., considering CT1-P_a and CT1-P_b to be a single class). However, containment event trees, used to relate a given accident class to a number of potential containment failure modes or release paths, were developed for only eight of the accident classes (Table 15.8). The frequencies associated with the other two accident classes were calculated to be relatively low in GESSAR II so that individual trees for these classes were not necessary.

Six potential containment failure modes were identified in GESSAR II and they are reproduced in Table 15.9. Of these six modes, two (i.e., the γ and δ modes) are caused by long-term gradual overpressurization and represent 14% of the containment failure probability in the PRA. The other failure modes (γ', γ'', μ, and μ') result from various hydrogen-related phenomena and comprise the remaining 86% of the containment failure probability. Clearly, hydrogen phenomena are potentially very important contributors to containment failure in the GESSAR II design.

On the basis of the staff/BNL analyses of these phenomena, hydrogen events such as combustion and detonation become important once the effective zirconium (metal)-water reaction reaches a level of about 30% of the effective fuel cladding in the active core region. GE considers other potential failure modes, such as steam explosions, isolation failure, or penetration of the basemat by the core debris, to be not mechanistically possible for GESSAR II design. The subjects of leakage of the containment large enough to prevent overpressure failure (leak before break) and purposeful venting of the containment to accomplish the same end are under review by the staff and have not been taken into account as yet in the staff analyses. Current emergency procedure guidelines incorporate venting instructions. These instructions are meant to enable the operators to mitigate Class II type events (see Table 15.8) which represent a small fraction of the total core-damage frequency. (See Table 15.4.)

From an inspection of the reference Mark III containment, and on the basis of current understanding of potential core debris/concrete interactions, the staff is unable to confirm that the core debris would be permanently retained in the containment building after a core-meltdown accident. However, it is likely that containment failure will occur either by overpressurization or hydrogen-related phenomena before basemat penetration.

The rationale that justified General Electric's eliminating in-vessel and ex-vessel steam explosions as potential failure modes for GESSAR II is presented in Appendix H of the PRA. The phenomena associated with steam explosions have been under extensive investigation at Sandia National Laboratories under the sponsorship of the NRC. Application of this research by the NRC to an evaluation of severe accidents in the Zion and Indian Point facilities has resulted in the conclusion that the probability of a steam-explosion-induced failure of containment is much lower than assumed in the Reactor Safety Study. However, both of the Zion and Indian Point facilities are pressurized-water reactors (PWRs) with large dry containments. From staff review of the GESSAR II containment and of the probability of steam-explosion-failure modes, it appears that the probability of such failures in the GESSAR II design will also be significantly lower than assumed in the Reactor Safety Study. Although the staff has concluded that steam-explosion-containment-failure modes have a lower probability than previously thought, research on the subject is continuing and the influence of smaller explosions on the source term is being reviewed as part of the development of mechanistic codes for ASTPO. In addition, sequences that bypass containment have been given very low probability in the PRA. General Electric considers that the containment-isolation-failure mode is improbable, as is the potential for excessive leakage through the MSIVs. These potential failure modes will be addressed in greater detail in the following sections.

The containment event trees relate the accident classes in Table 15.8 to the six failure modes in Table 15.9. Clearly, there is a potential for a very large number of fission-product-release paths and, hence, source terms. GE reduced the large number of potential release paths to a more manageable number of release categories using a computer code that GE developed. The various release sequences in the containment event trees were binned into a smaller number of consolidated sequences (release categories). The conditional probabilities for each of these consolidated sequences can be obtained from the event trees and they relate the accident class to the release description. The accident-class frequencies can be combined with the conditional probabilities for the consolidated release categories to calculate the frequencies of each of the 15 release categories. Each of the 15 release categories has the potential to result in a variety of offsite consequences (or damage indices). Two potential damage indices (namely latent fatalities and person-rem) are given in Table 15.4 for each release category. GE calculated the health consequences shown in Table 15.4, and no early fatalities were predicted for the 15 GESSAR II release categories. The mean-damage indices are multiplied by the frequencies of the release category to give a measure of the overall risk for the GESSAR II standard plant. The risk is expressed in terms of accident descriptions in Table 15.5.

From an inspection of Table 15.7 it is clear that Class I transients initiated by loss of offsite power (CT1-P) are important contributors to the frequency of all the accident classes. The staff and its contractor, BNL, have, therefore, reviewed the containment event trees for this class in more detail than the event trees for classes with lower probabilities. In GESSAR II, Class I transients initiated by loss of offsite power (LOOP) and the failure of the diesel generators were represented by two separate containment event trees. The total probability associated with the CT1-P class is subdivided into the probability of restoring power within 60 minutes of the start of core damage ($CT1\text{-}P_a$) and the probability of restoring power after 60 minutes ($CT1\text{-}P_b$). As a result of this review process, the two event trees shown in Figures 15.2 and 15.3 were constructed. Differences between the event trees developed by the staff/BNL and those developed by General Electric are discussed in the following sections

Suppression-Pool Bypass

Suppression-pool bypass pathways may be pre-existing or may develop through failure of the containment isolation system, failure or leakage of drywell penetrations, or failure or leakage of drywell hatches. In addition to the consideration of the formalism of failure evaluations in probabilistic assessments, the staff is also considering operating experience. In regard to pre-existing leakage, Weinstein (1980) reports review results of containment leakage at or below technical specification limits, based on operating experience through 1978. The author concluded that the achievement of containment integrity was low (no higher than 0.92) and had remained constant over time, but the severity of the leakage above technical specification limits had been decreasing. Although no BWR plants with Mark III containments are included in the operating data, it is clear that most plants experience the personnel, design, maintenance, or operating errors that contribute to leakage in excess of technical specifications. GE has not considered this pathway in the PRA; GE's "unidentified drywell leakage" consists of leakage at a typical Mark III technical specification limit. GE has considered failures in lines from either the reactor pressure vessel or the drywell which could bypass the suppression pool by opening to either the containment area above the pool or the secondary containment. Table 15.11 gives the coding used to define different degrees of pool bypass and scrubbing and Figure 15.4 provides a graphical description of the three cases used in the risk assessment of the GESSAR II design. Table 15.12 lists these lines and also lists the barriers to fission-product release for each pathway. GE assumes no suppression pool bypass paths resulting from isolation system failures consisting of multiple isolation valves failing to close concurrent with pipe breaks. Valve failure rates are assumed to be unaffected by the severe accident because: (1) valves inside the drywell generally close before the time when severe accident conditions develop and are consequently qualified for drywell LOCA environments, and (2) valves which are outside the containment are generally unaffected by the severe-accident environment before their closing. The staff is currently investigating containment isolation valves in operating plants to establish a more extensive data base covering actual leakage and reliability under operating conditions. Results of this effort will enable the staff in the future to understand better the basic operability and failure data concerning containment isolation systems. The staff does not expect the results of this investigation to change conclusions that are relevant to GESSAR II.

Some of the largest potential bypass paths listed in Table 15.12 are the 26-in. main steam lines. The main steam isolation valves (MSIVs) are designed to close automatically before a severe accident has progressed to the core-damage stage. Automatic closure should occur, at the latest, on the low reactor water level signal. The environment for the MSIVs located inside containment is bounded by the environmental qualification conditions (NUREG-0588, Rev. 1) which is based upon drywell LOCA conditions. The outboard MSIVs are located in the steam tunnel and will be exposed to much lower temperatures. Consequently, the containment isolation system can be expected to function, and these valves are expected to close initially. However, once closed they will be subjected to the severe accident environment.

Leakage through MSIVs has been observed to exceed current technical specification limits (11.5 scfh) at several operating plants. Values of several thousand cubic feet per hour have been observed occasionally. For these high-leakage valves, the leakage is currently postulated to be occurring because of a combination of oxidation buildup on the valve seats and inadequate guidance of the valve stem as it strokes toward the valve seat. Although leakage through such paths is likely to be very small, it should be noted that this type of bypass path may provide the effective limit on fission-product removal if high suppression pool scrubbing is estimated.

In addition to containment isolation system and MSIV failure, another potential source of pool bypass is through failure of disabled reactor-pressure-vessel instrument lines following hydrogen combustion in the containment. It is assumed that three assemblies must fail in order for significant leakage to occur. This failure is listed in the containment event tree, CT1-P_a, under the columnar heading "no breach of RPV [reactor pressure vessel] pipes," with failure symbol "δ'." This conditional failure probability of three elements is believed to be small.

Examples of RPV instrument lines considered are 1-inch lines (including control rod drive lines), 1½-inch standby liquid control lines, and 3/8-inch sample lines.

Pool bypass can also result if drywell penetrations are breached. These types of failures are assigned the failure symbol "δ" in the event trees. This is postulated by GE to result from either the failure via continuous burn of hydrogen near a large penetration or guard pipe, or failure of a vacuum breaker set. For the penetrations and guard pipes, the failure is assumed to have a probability of 0.01 in the containment event tree, CT1-P_a, and is determined by taking the product of the conditional probability of a continuous burn given ignition, 0.31, times the conditional probability of a failure of the penetration/low-pressure coolant-injection guard pipe as a result of the continuous burn, 0.03, i.e., 0.31 x 0.03 = 0.01. GE did not consider pathways through small electrical penetrations and hatches in its event tree because of the assumed plugging effect of aerosols (Morewitz plugging) inside tortuous paths which are expected to limit releases. The staff has assumed in its assessment that the Morewitz plugging model may not apply for certain leakage paths (e.g., the guard pipe) so that a bypass pathway may remain.

Breach of the drywell may also result from failure of the vacuum breakers. The GESSAR II design employs redundant air-operated vacuum breakers in series. The design flow direction is from the containment to the drywell. A potential bypass flow must involve failures of both the normally closed, fail-closed vacuum breaker and the associated check valve. GE predicts the probability of both the power-operated valve and the mechanical (check) valve failing to be $(2 \times 10^{-3}) \times (1 \times 10^{-4}) = 2 \times 10^{-7}$. This value is lower than the value of 0.01 previously mentioned resulting from failure of a penetration or guard pipe from continuous burn.

For event trees not involving hydrogen ignition, the failure probability "δ," is governed by the GE estimate of 10^{-4} of a sudden change of the drywell leakage rate from about 200 cfm to 1,500 cfm.

In general, the staff/BNL agree with the General Electric evaluation that implies a relatively low probability of significant suppression-pool bypass. However, the staff/BNL have been unable to confirm the applicability of the

Morewitz plugging model, which was developed for dry, inert aerosols, to highly radioactive fission products under light-water-reactor (LWR) conditions, and, therefore, has not explicitly considered the effects of leak-path plugging by aerosols.

(6) Fission Product Transport and Consequence Analysis

In this section the staff provides an assessment of the fission-product release from the damaged reactor core, the transport of fission products in containment, the ultimate release of these fission products to the environment, and the potential health consequences that ensue. This portion of the staff's review is, therefore, specifically related to the following sections of the GESSAR II PRA:

Section 5:	Magnitude of Radioactive Release
Section 6:	Consequences of Radioactive Release
Appendix F.3:	Fission Product Release and Transport
Appendix F.4:	Consequence Analysis

In addition, the information provided by General Electric at the Third and Fourth General Electric-NRC Technical Update Meetings (BNL letters, Aug. 20 and Oct. 27, 1982) is relevant to this section and supplements the descriptions in the PRA. Because this rapidly evolving subject is so important, the staff's formal questions to General Electric addressing source-term behavior were extensive, and numerous meetings were held to discuss this part of the PRA.

The NRC staff took cognizance of the methodology currently being developed by the Accident Source Term Program Office that is receiving extensive peer review in performing a parametric study to reflect many of the phenomena of interest. The flexibility of the MARCH 1.1 and CORRAL codes was utilized to perform this parametric study in: primary system hold-up, re-emission, suppression pool decontamination, and release from core/concrete interactions. The following sections discuss the rationale for the range in each parameter as utilized in the study. It should be recognized that this does not represent a rigorous application of the most recent methodology but, rather, a method of approximating the major factors believed to be significant to source terms based on a review of the literature. Both the staff's consultants and General Electric performed such an exercise.

Range of Parameters Used in Source-Term Sensitivity Study

In-Vessel Release From Fuel

The staff/BNL model for in-vessel fission-product release during core degradation is based on the release rate model described in NUREG-0772. In this model, the fractional rates of fission products released from the fuel are a function of both time at a certain temperature and the rate of temperature change. This implies that the fission products released during core degradation will be a function of the sequence being studied. In the calculations for GESSAR II, the NUREG-0772 fission-product model is included directly in the MARCH code, and thus, as the core heatup progresses, the fission-product emission is calculated simultaneously. Because of recently available data on tellurium behavior, two sets of release fractions were used for tellurium release. The data of Lorenz, Beahm, and Wichner (1983) account for the possibility of zirconium-tellurium compound formation which will inhibit the release of tellurium; this was not included in NUREG-0772.

Besides the core-degradation phase of the accident, there is a second in-vessel release phase, during which fission products could be emitted. This phase corresponds to the time during which the core is reheating after it has slumped into the inlet plenum and is starting to attack the bottom head. This release was estimated by using the same fission-product release model described above and limiting the fuel temperature rise to the temperature corresponding to that of melting steel.

In summary, the staff/BNL parametric study is based on the NUREG-0772 release-rate model included directly in the MARCH code. The release of fission products is thus both a function of time at temperature rise and in the case of Te, the amount of Zr metal available. Results using this method are shown in Table 15.13. The corresponding General Electric results are shown for comparison. Also shown are the corresponding results from WASH-1400, and preliminary results from a study by Battelle Memorial Institute (BMI-2104), the latter using the methodology being developed for ASTPO. The results determined by General Electric and BNL are the total in-vessel release (core melt and second in-vessel heatup). It can be seen that with the exception of Te, the agreement among the release fractions is quite good. The volatile components (Xe-Kr, I, and Cs) are largely emitted (90%-100%); the less-volatile groups (Ba, Ru, and La) are emitted to a lesser degree. The release of Ba-Sr and La in the WASH-1400 model are respectively low and high when compared to the methods based on NUREG-0772. In the case of Te, the WASH-1400 release is particularly low; General Electric and BNL are close if the NUREG-0772 release rates are used, and BMI and BNL are close if the modification of Lorenz, Beahm, and Wichner (1983) (assuming Zr oxidation) is used.

The in-vessel release fractions presented in Table 15.13 are in acceptable agreement, with the exception of Te. The Te release fraction is a function of the availability of Zr metal and varies by a factor of approximately 2. This factor is not significant for the present state of the art of source-term methodology development but may become important in future assessments using more-established mechanistic models.

Primary System Retention

The staff's estimates of the range of primary system holdup are based on TRAP-MELT calculations reported in a draft copy of BMI-2104, Volume 3 (ASTPO, 1983). This analysis was based on the Grand Gulf facility, which has a primary system similar to the GESSAR II design. In order to estimate an appropriate range of primary system retention, the retention factors as a function of sequence and time within a sequence were inspected. The high and low values for various nuclides were found by taking the highest and the lowest values from the BMI study and these are shown in Table 15.14. In contrast, GE assumed the same retention for all fission products (assumed to be either 0.0 or 0.95) except for noble gases and organic iodine which were not retained.

Primary System Re-Emission

The Xe-Kr and organic iodine groups are not retained in the primary system, so this issue does not affect these groups. The re-emission of fission products was assumed to affect only the more-volatile aerosol groups, namely, I, Cs-Rb, and Te-Sb. The less-volatile aerosols were not assumed to be re-emitted. The range assumed in this analysis varied from 75% re-emission (for the highly volatile nuclides) to no re-emission for the aerosols. The values were based on engineering judgment, using limited experience gained in a single iteration of an in-vessel transport calculation. In the General Electric sensitivity study, no re-emission was assumed.

Suppression Pool Decontamination

While the primary purpose of the pressure-suppression pool is to condense steam which would otherwise pass into the containment atmosphere, the water in the pool can also be expected to absorb fission products and other debris which may be carried into it during an accident. With the exception of the noble gases and organic forms of iodine, all other fission-product elements can exist only as vapors which are soluble in or react with water, or as materials which are solids at or below the boiling temperature of water. The fission-product retention in the pool is based on a model that allows gases injected rapidly into the pool to disperse into small bubbles which quickly approach thermal and chemical equilibrium with the pool water as they rise to the surface. The absorption by the water of vapors and entrained particles from these bubbles can occur through several processes, the overall effect of which is referred to as "scrubbing."

Critical parameters in estimating scrubbing efficiency are the particle-size distribution of entrained material swept into the pool, the relative amount of noncondensible gases entering the pool, the size and shape of the bubbles, and the temperature and chemical properties of the pool water.

Additional factors limiting the overall effectiveness of the pool are bypass of the suppression pool and re-evolution of dissolved fission products from the pool surface. The latter effect includes vaporization of volatile fission-product compounds (e.g., volatile iodine forms) and entrainment of small water droplets during vigorous bubble bursting and flashing of pool water. Although these phenomena are considered to be of secondary importance, they could place an effective limit on fission product removal when very high pool DFs (e.g., the value of 10,000 used by General Electric) are estimated.

Examples of the variability in the critical parameters that can be obtained at various times in different accident sequences are given in Table 15.6. Given in the table are also the high and low DF values used in the staff's parametric analysis for fission-product removal by the pool, and the corresponding DF values used by GE.

Core/Concrete Interaction Release

The release of fission products during core/concrete interactions is highly uncertain and the range of uncertainty was obtained from a series of VANESA calculations published as part of the QUEST study (Sandia, 1984). The fraction of fission products released from the melt during core/concrete interactions is shown in Table 15.15. As can be seen from this table, the fraction released for any given nuclear species is expressed as a range. Essentially, no Ru or Mo are released during this phase of the accident and thus they do not appear on the table. Furthermore, Ce acts as a volatility-level surrogate for Pu and Np whereas La acts as a surrogate for Y, Pr, Nd, and Am. The release of fission products during core/concrete interaction is scenario dependent since it has to be consistent with the in-vessel release, which is scenario dependent. The high and low release values are shown on Table 15.15. It is clear that essentially all the noble gases and volatile fission-product groups are emitted. The GE values for core/concrete release were obtained from the Reactor Safety Study, from which they varied by a factor of 2.

Transport Within Containment

The aerosol transport within the containment, including such processes as plateout and settling, is determined by the CORRAL code. This model is known to have shortcomings, such as neglect of aerosol agglomeration and overly simplified modeling, which tend to overestimate the airborne aerosol concentration. This results in an enhanced leakage of aerosols at the time of containment failure. However, this effect would be lessened for the design described in GESSAR II because of the sizes of particles. Additional calculations with the methodology currently under development by ASTPO will be performed and if the results differ from those presented here, they will be reported in a future supplement to the SER.

(7) Releases to the Environment

Release Categories

To render risk computations tractable, a comparatively small number of specific release descriptions and avenues of escape to the environment are chosen to represent the continuum of all possible releases. These release categories, consist of specification of the time after the accident, initiation of the start of release, its duration and location, the heat energy discharged during the release, the relative time at which the operators would be aware that a severe accident was evolving, and the quantities of each of the fission products released to the environment.

The staff evaluation of release categories is based on a parametric study of severe accident phenomenology. This parametric study gives high and low estimates of the release fractions for important nuclides which differ by several orders of magnitude. Therefore, at the present state of the art of the staff review, there is no reliable information on differences in release fractions among various sequences; any such actual differences are masked by the large range of the parameters used in the study. The staff's low estimates are comparable to the General Electric baseline estimates (see Table 15.16). The staff's high estimates of non-noble-gas-release fractions are markedly lower than those employed in the Reactor Safety Study which modeled a Mark I containment. Much of this difference in fission-product release is due to differences in the design between the Mark I and Mark III containments.

Element Groups

The radiologically important elements have been assembled into groups in the staff's computation. These element groups are used in Table 15.16 and are as follows:

(a) krypton and xenon
(b) iodine
(c) cesium and rubidium
(d) tellurium and antimony
(e) strontium
(f) barium
(g) technetium
(h) molybdenum, ruthenium, and rhodium
(i) lanthanum, yttruim, praeseodymium, and neodymium
(j) cerium, plutonium, neptunium, and americium
(k) cobalt, curium, zirconium, and niobium

(8) Meteorology

The offsite doses computed to result from the accidental release of radioactivity into the environment can span a very large range because of normal variations in the wind direction and other weather conditions at the time of the accident. Releases from a coastal site, for example, may be blown off shore by strong winds, resulting in virtually no radiological impact to the surrounding population. It is also possible that the release could occur during a period of heavy precipitation, causing virtually all of the radioactive aerosols in the release to be deposited on or near the site and permitting only the noble gas and organic iodine radioisotopes to be carried to the surrounding population. At other times, however, the wind may blow in the direction of a heavily populated area, and precipitation may not occur until the release has been transported to the area, thus depositing a large fraction of the release among the greatest density of people.

During a representative year, weather conditions approaching the most and least favorable for each category of potential offsite risk will occur. In general, extreme weather conditions by normal standards (hurricanes, droughts, etc.) do not greatly affect the distribution of conditions during an entire year for purposes of modeling the dispersal of radioactivity in the environment, and such extreme observable conditions do not lead to more extreme offsite radiological consequences than those sampled during a typical year. In brief, meteorological conditions conducive to both the most and least favorable dispersals might easily be seen as comfortable weather conditions by local inhabitants.

Hourly measurements at Shippingport, Pennsylvania, during the calendar year 1979 were used in the consequence calculations. GE used the data for site 6 of the Reactor Safety Study.

(9) Population Density

The average population densities within 20 miles of existing U.S. nuclear power plants vary from a few people per square mile to several hundred people per square mile. These plants include many situated in sparsely inhabited regions and some in the far outskirts of large metropolitan areas. In most cases these average population densities describe very nonuniform population distributions consisting of towns and small cities having local densities of thousands of people per square mile along with large areas with little or no population. For virtually all plant sites, the number of people who could be affected by an atmospheric release varies greatly with the direction in which the wind carries the release.

For purposes of calculating risks of severe accidents at a plant of the GESSAR II design, the staff has selected a site intended to represent a reasonable upper bound of a U.S. population that could be at risk. By several measures of local and regional population density, this site possesses a greater surrounding population than 90% or more of the existing nuclear plant sites in the United States. It is also comparatively uniformly surrounded by inhabitants, without large uninhabited areas nearby, minimizing the dependence of risk upon wind direction frequency. The site selected is that of the U.S. government-owned Shippingport power reactor, located 25 miles from Pittsburgh, Pennsylvania, which was recently decommissioned after 25 years of operation.

Increasing population and other demographic changes will affect the population density of all sites, both those that are currently occupied by nuclear power plants and those that may be selected in the future. To account for such changes, census data have been used to extrapolate the population densities of towns and cities surrounding Shippingport to the densities expected in the year 2010.

General Electric performed consequence computations using a hypothetical constructed-site population approximating methods used in the Reactor Safety Study. The staff, using the criteria outlined above, has selected a site that could be described as typical of existing reactor sites in the more densely inhabited portions of the United States.

(10) Projected Accident Consequences

CRAC Code

"Release Categories" (Section 15.6.2.5(7)), the Shippingport site meteorogical data (Section 15.6.2.5(8)), and the population distribution (Section 15.6.2.5(9)) were used in the CRAC (Computation of Reactor Accident Consequences) code to estimate radiological effects of severe accidents. For each release category, 1,456 dispersals into the surrounding environment were computed, corresponding to each of 91 representative weather sequences with wind direction assumed to be into each of 16 compass headings from the site. The probability of the wind blowing into each of the compass headings was taken from the observed wind direction distributions at Shippingport during 1979.

Potential radiological consequences are dominated by doses received from radioactivity deposited during the passage of the released plume. It would be unreasonable to assume that heavily contaminated areas would remain populated for long periods of time following the accident. The CRAC model permits evacuation of selected areas following an accident and also the later relocation of population from heavily contaminated areas after the plume passes.

For slowly evolving accidents, offsite protective measures can significantly reduce radiological consequences to the nearby population in computing dose consequences. Therefore, the inhabitants within 10 miles downwind of the plant were assumed to begin evacuation 2 hours after plant operators warned of an impending severe accident. These evacuees were assumed to move downwind at an effective speed of 2.5 mph, escaping from the plume's path 6 hours later after having traveled an effective downwind distance of 15 miles. For those sequences in which more than 8 hours would elapse between warning and the beginning of the release, the population within 10 miles was assumed to have evacuated before the release. For all the remaining sequences, the warning time has been estimated to be less than 2 hours, and the evacuation of the nearest inhabitants may occur within the released plume.

In real evacuations, it is likely that some fraction of those who should have been informed of the need to evacuate will either decline to evacuate or will not have received the warning, while most will leave at speeds greater than 2.5 mph or with less delay. It is possible, therefore, that risk to evacuees has been either overestimated or underestimated.

The health-effects calculations used the same methods as the Reactor Safety Study, but with minor improvements in data. The thyroid doses received by iodine inhalation depend upon the chemical form of iodine in the environment, and the thyroid dose calculations used data consistent with either soluble iodide adsorbed on one 1-μ-diameter aerosol particles or elemental iodine vapor.

For areas so heavily contaminated that a dose of 200 rem to bone marrow could be received in fewer than 7 days, it was assumed that relocation of the area population would occur 12 hours after the plume passed. For areas of lesser contamination, ground exposure was limited to 7 days in calculating all non-chronic dose effects.

<u>CRAC Input for GESSAR II</u>

In order to characterize the consequences of the release of radioactive material from the GESSAR II design, three of the release categories were chosen for further study. Since a parametric study was used to estimate source terms for the different release categories, rather than a detailed phenomenologic assessment, conclusions based upon a simple inspection of differences are unwarranted. Table 15.16 lists the plume characteristics and the high and low release fractions determined in the parametric study for one of the categories to illustrate the range of release fractions (1-SB-E1 sequence). The timing of the release is also important in calculating early consequences. Table 15.16 also gives the high release fractions and the plume characteristics for two categories where the release fractions are virtually identical, but the timing of the release is greatly different (ATWS and 2-T-L3). The values of the release fractions and plume characteristics calculated for the three release categories by General Electric are given in Table 15.16 for comparison.

The release categories in Table 15.16 are specified as fractions of the core inventory of the 54 nuclides used in the Reactor Safety Study to evaluate accident consequences and the inventory has been normalized to account for the GESSAR II power level. The nuclides are further modified by CRAC to account for radioactive decay during the times until release listed for each release category.

Computed Accident Consequences

Accident consequences include a large number of human, environmental, and economic impacts. To represent the impacts related to GESSAR II, the staff has chosen several important categories to reflect both early and chronic impacts as indices of all accident consequences.

Selected damage indices (calculated as discussed in Appendix VI of the Reactor Safety Study) for the three release categories in Table 15.16 are listed as conditional means in Table 15.17. Conditional means are average consequences, assuming the release to have taken place, of the 91 meteorological sequences for each of the 16 wind directions, each weighted by wind directional frequency Damage indices for the low-release fractions are not shown, as they are comparable to those calculated by General Electric (see Table 15.4).

Only the 1-SB-E1 (high) release category gave doses greater than 200 rem to bone marrow for 5 of the 91 meteorological sequences sampled. Further, in one of the 5 sequences, calculations gave bone-marrow doses above 320 rem. The resulting small probability of death shared by a small number of people as calculated by CRAC is shown in Table 15.17 as a small fraction of a fatality.

To illustrate the importance of plume characteristics, the timing of the release and evacuation assumptions, the damage index of early injuries (55-rem dose to the bone marrow) in Table 15.17 may be compared for the two release categories with comparable release fractions but different timing (Sequences 2-T-L3 and ATWS in Table 15.16). The release category with short warning time produced estimated exposures high enough to be considered early injuries for some meteorological sequences, but no such doses were computed for the category with a long warning time. The meteorological sequences for which early damage indices are shown in Table 15.17 were dominated by those with precipitation which concentrated the release of radioactivity in the vicinity of the site.

Also shown in Table 15.17 are "person-rem" and total cancer fatalities.

Conclusions Regarding Risk

General Electric calculates no risk of early fatality and the staff, using a high estimate of release fractions, calculates an exceedingly small average early fatality. This is attributable both to improvements in source-term methodology compared to the Reactor Safety Study and to the GESSAR II design where the majority of fission products released from the reactor vessel (or the drywell) to the wetwell are directed through the suppression pool. Since the release fractions used in the staff's consequence calculations are characterized as high-range estimates, and the low-range estimates are comparable to General Electric's estimates, the staff has concluded that the risk of an early fatality is negligibly small.

General Electric has not presented calculations of early injuries. The staff calculated bone-marrow doses above 55 rem for the cases where little warning time is available for the population in the vicinity of the site to take protective action. This emphasizes the importance of the containment and the improvement that could be obtained by delay of containment failure.

15.6.3 Consideration of Overall Potential Design Improvements

15.6.3.1 Compliance With Severe Accident Policy

The Commission's proposed policy (draft NUREG-1070) regarding severe accidents requires that an application for a standard plant design comply with the requirements of 10 CFR 50.34(f) (CP/ML rule). Paragraph (1)(i) of the rule requires the applicant to assess improvements in the plant design that have potential for significant risk reduction that are practical and do not impose an excessive economic impact on the plant.

In accordance with the severe-accident policy statement, the decision process for certifying the GESSAR II design will include a multifaceted approach. Deterministic calculations, engineering judgment, rulemaking on various specific issues, design principles such as defense in depth, as well as PRA methods, will all be used.

With respect to severe accidents caused by internal events, the methodology available today as described in this supplement allows the staff to predict with reasonable confidence that the consequences of severe accidents are likely to be significantly less than those predicted in the Reactor Safety Study. This is due to several factors including differences in plant systems and containment design between GESSAR II and the Mark I Reactor Safety Study plant, in updated fission-product source-term methods, and increased confidence in application of those methods. On the basis of the results of current staff analyses using currently available methodology, the staff believes that very costly preventive or mitigative plant design modifications to the GESSAR II design cannot be justified on a risk-reduction basis.

Cost/benefit analysis has been used as a screening tool for the potential design improvements discussed more fully below. As stated in the proposed Commission policy statement relating to severe accidents (draft NUREG-1070) final staff decisions will be based on engineering judgment, insight and experience, understanding of phenomenology, and on the staff's confidence in its understanding, and attributes such as reduction in core-damage frequency, cost and ease of modifications, and maturity of the technology.

15.6.3.2 Potential Design Improvement

The staff prepared a list of potential design improvements and guidance regarding methods of assessing the relative benefits of the improvements (NRC letter, April 13, 1984). The list consisted of 14 different groups by subject and included such things as human factors (accident diagnostics, maintenance, and emergency procedures), augmented decay heat removal, combustible gas control, venting systems, ac/dc power supplies and other system improvements (see Section 15.6.1, this supplement). The staff's guidance for General Electric's assessment of the potential design stated that the assessment should include:

(1) A discussion of each potential design improvement included on the list (and any other considered appropriate by GE) with a qualitative assessment of the relative advantages and disadvantages of each item.

(2) A quantitative ranking of each item considering its potential relative impact on overall plant risk using an acceptable ranking method such as is described in NUREG/CR-3385, "Measures of Risk Importance and Their Applications" or NUREG/CR-3568, "A Handbook for Value Impact Assessment."

(3) An identification of the most promising means of risk reduction with preliminary cost estimates for a selected set of improvement schemes, this set to be chosen after discussion with the staff.

(4) Detailed risk, incremental risk, and cost-benefit analyses for some selected subset of potential design improvements after discussion with the staff.

In response to the staff's request, GE provided a preliminary assessment to be used for an initial ranking of each of the potential design improvements in the staff's suggested list (NRC letter, April 13, 1984). Seventy-three concepts (potential improvements) were reviewed for which estimated costs, risk reduction, and cost/benefit ratios were reported.

The staff met with GE on May 1, 1984 to discuss the preliminary assessment and to provide GE with comments and guidance for followup work in this area. On the basis of staff comments, GE submitted a more-detailed assessment (NEDE-30640). This assessment included a description of an ultimate plant protection system (UPPS) intended to accomplish a number of preventive/mitigative actions including inventory makeup, depressurization, and heat removal - all without dependence on electrical power. The UPPS is described further below.

On the basis of the cost/benefit assessment performed by GE, a ranked design improvement list has been prepared and is shown in Table 15.18. Table 15.18 gives design modifications for which GE calculated a cost/benefit ratio of about 150 or less and indicates that there are a number of diverse systems falling within this range of cost/benefit.

In NEDE-30640, GE concludes that since none of the design modifications analyzed were shown to have a cost/benefit ratio of 1 or less, there are none that are cost beneficial for the GESSAR II design. It is also concluded that if any modification is to be implemented, the addition of the UPPS would reduce risk the most and would lessen the importance of generic and unresolved safety issues.

It is the staff's opinion that if uncertainties in core-damage frequency and source terms are accounted for, cost/benefit ratios lower than those presented in Table 15.18 by as much as 3 orders of magnitude could result. For this reason, the staff is continuing to consider the items in Table 15.18 as well as other items, and will require that GE perform more detailed studies of a selected subset of improvements before the staff reaches conclusions about the need for improvements in the GESSAR II design.

On the basis of a preliminary review of UPPS, the staff agrees that the UPPS appears to have merit in providing a backup low-pressure coolant makeup supply and in mitigating containment overpressurization. Figure 15.5 is a simplified schematic drawing of the UPPS. This system is designed to provide a connection allowing hookup of the fire protection system or a fire truck to the low-pressure core spray (LPCS) system injection line, thus supplying makeup to the vessel. Containment heat removal capability is provided by venting the containment atmosphere through the air-operated containment high-flow valves. RPV depressurization is to be accomplished in the absence of electric power by actuating selected SRVs using air from the pneumatic system air supply (bottles). GE states that none of the above actions are dependent on the availability of electric power.

In summary, initial screening of potential design improvements to eliminate from further consideration plant modifications that are not believed to be cost beneficial is nearing completion. The staff is considering a set of design options (including UPPS) for further study. The resulting set is to be evaluated in detail independently by GE and the staff, utilizing all of the analytical tools addressed for in the proposed severe-accident policy statement.

The completed results of the staff's review of the design improvements will be reported in a future supplement to the SER.

15.6.4 Staff Recommendations and Conclusions

Section 15.6.1 identified the actions necessary to satisfy the proposed severe-accident policy statement requirements for standard plants. The staff's conclusions regarding these requirements are discussed below.

(1) Topic A of the proposed policy statement requires that plant design improvements be considered and that cost-beneficial improvements be incorporated into the design. The staff directed GE to evaluate the desirability of more than 70 design improvements. Details of this process are described in Section 15.6.3.

After a preliminary assessment the staff finds certain improvements that are worth further study. The staff will continue its evaluation and will determine which modifications should be incorporated into the GESSAR II design. Staff findings will be reported in a future supplement to the SER. Completion of the staff's evaluation will demonstrate satisfactory treatment of Topic A from NUREG-1070.

(2) Topic B of the proposed policy statement, requires the technical resolution of all applicable Unresolved Safety Issues and high- and medium-priority Generic Safety Issues. All applicable USIs and GSIs are discussed in Appendix C. Where sufficient detail in plant design is available, resolution has been demonstrated except for USI-48 which remains open. Where design information is insufficient, actions necessary for utility applicant resolution are documented. Issues requiring applicant evaluation are summarized in Appendix C. The staff judges that Item B of NUREG-1070 has been satisfactorily treated when the hydrogen-control issue is resolved.

(3) Topic C requires a probabilistic risk assessment and consideration of severe accident vulnerabilities. The GE risk assessment was reviewed by BNL and the staff. Minor elements of review are still ongoing. Details of staff evaluation of the PRA are found in Section 15.6.2.

The results of the staff/BNL review indicate that the core-damage frequency from internal events is relatively low, at 3.8×10^{-5} per reactor year. This frequency is further reduced with implementation of the UPPS. Station blackout events contributed the major portion of this frequency. The PRA also evaluated the external events of seismic, fire, and internal flooding. Fire and internal flooding were determined to contribute insignificantly to the GESSAR II core-damage frequency. The results of the staff's review related to GE seismic analysis will be reported in a future supplement to the SER.

The staff is utilizing results from the PRA to draw insights regarding plant vulnerability to severe accidents, and to assist in the evaluation of design improvements. Some risk insights for internal events have been gained. The risk of latent cancer is significantly smaller than that shown in the Reactor Safety Study, owing both to changes in the source-term methodology and to plant design. The risk of early fatalities is negligibly small.

Two issues related to the PRA remain open, seismic analysis and containment structural analysis.

The resolution of these issues will be addressed in a future supplement to the SER. At this time, the staff believes that the basic PRA conclusions will not be significantly changed. With the resolution of the above issues, Topic C of draft NUREG-1070 will have been satisfactorily treated.

Topic D of draft NUREG-1070 requires that the staff complete its review of the design and conclusion of safety acceptability for the design. Topics A, B, and C above constitute staff review of the GESSAR II design. With completion of the areas discussed in those topics and resolution of the open issues listed in Section 1.8, the staff expects to be able to confirm the safety acceptability of the GESSAR II nuclear island, and recommend rulemaking for design certification.

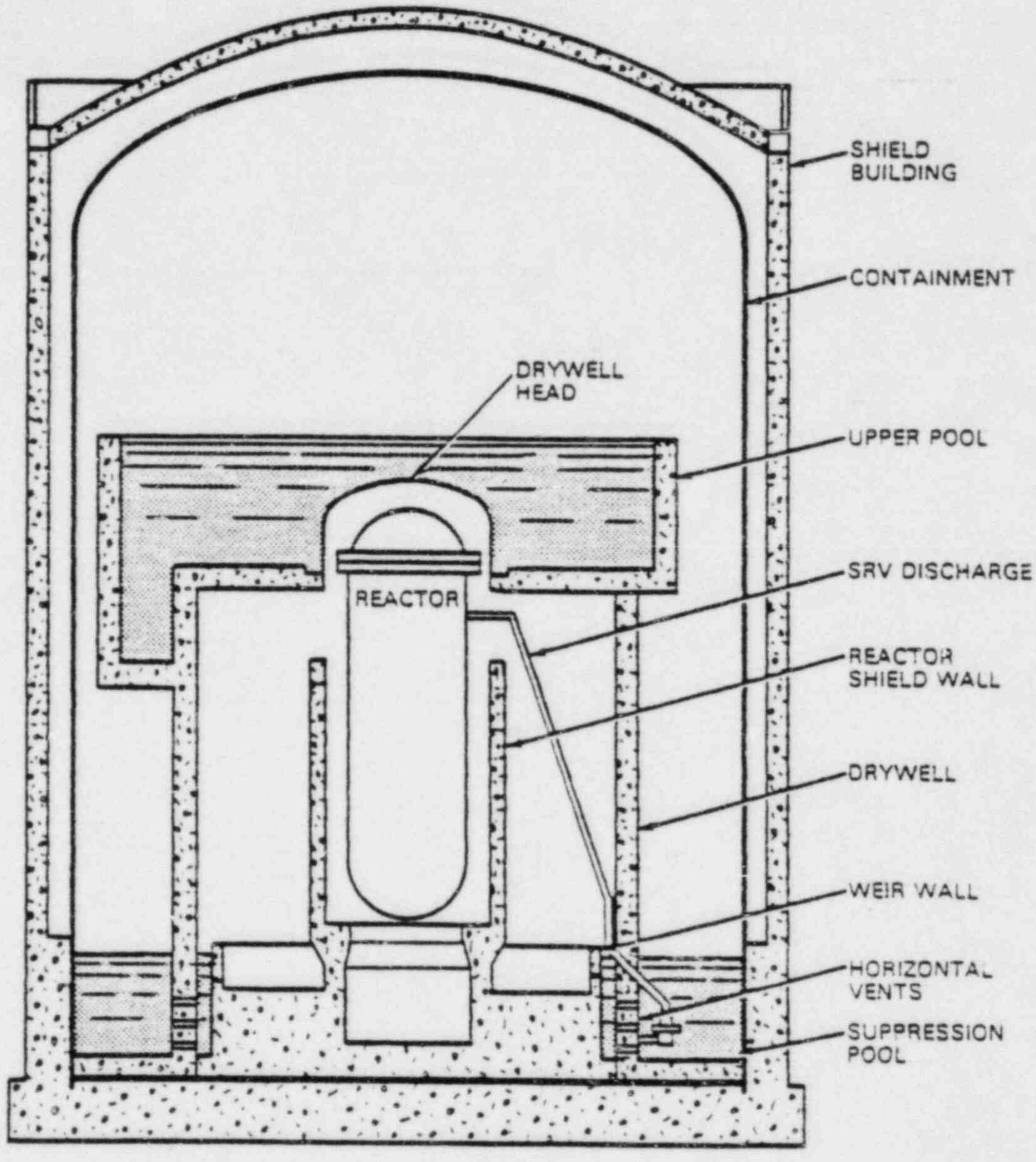

Figure 15.1 Principal features of MARK III containment

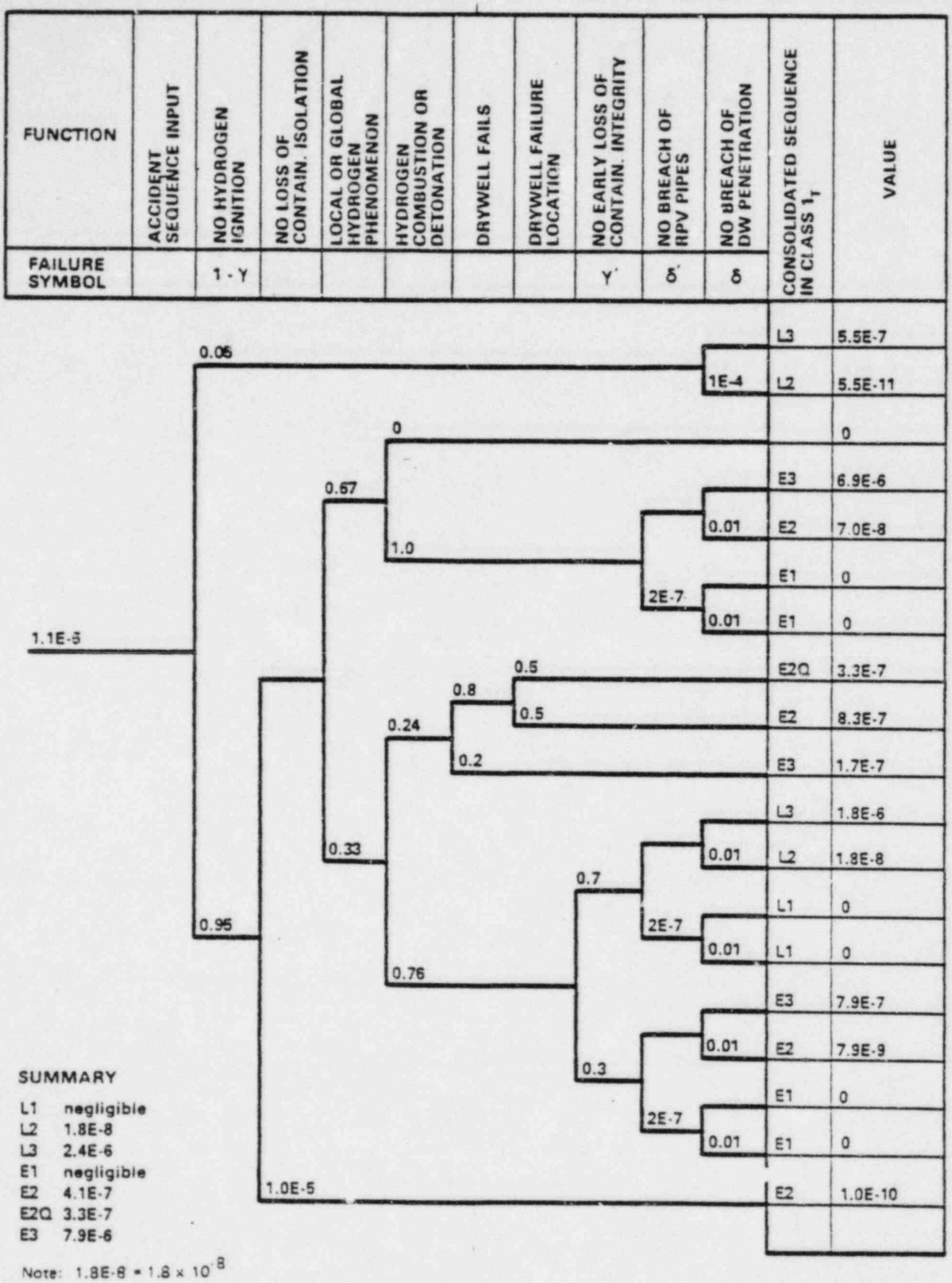

Figure 15.2 CT1-P_a best estimate containment event tree

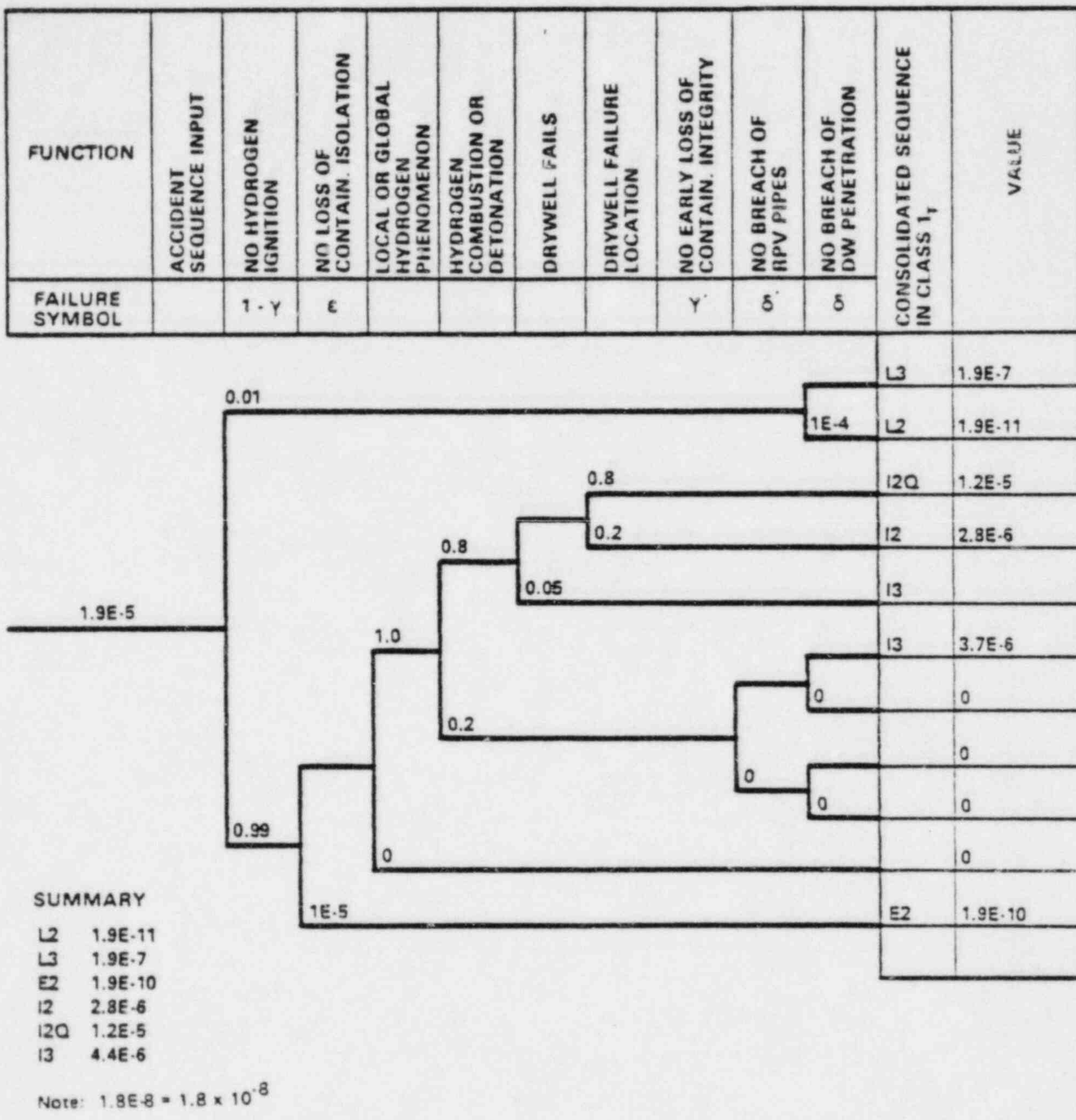

Figure 15.3 CT1-P_b best estimate containment event tree

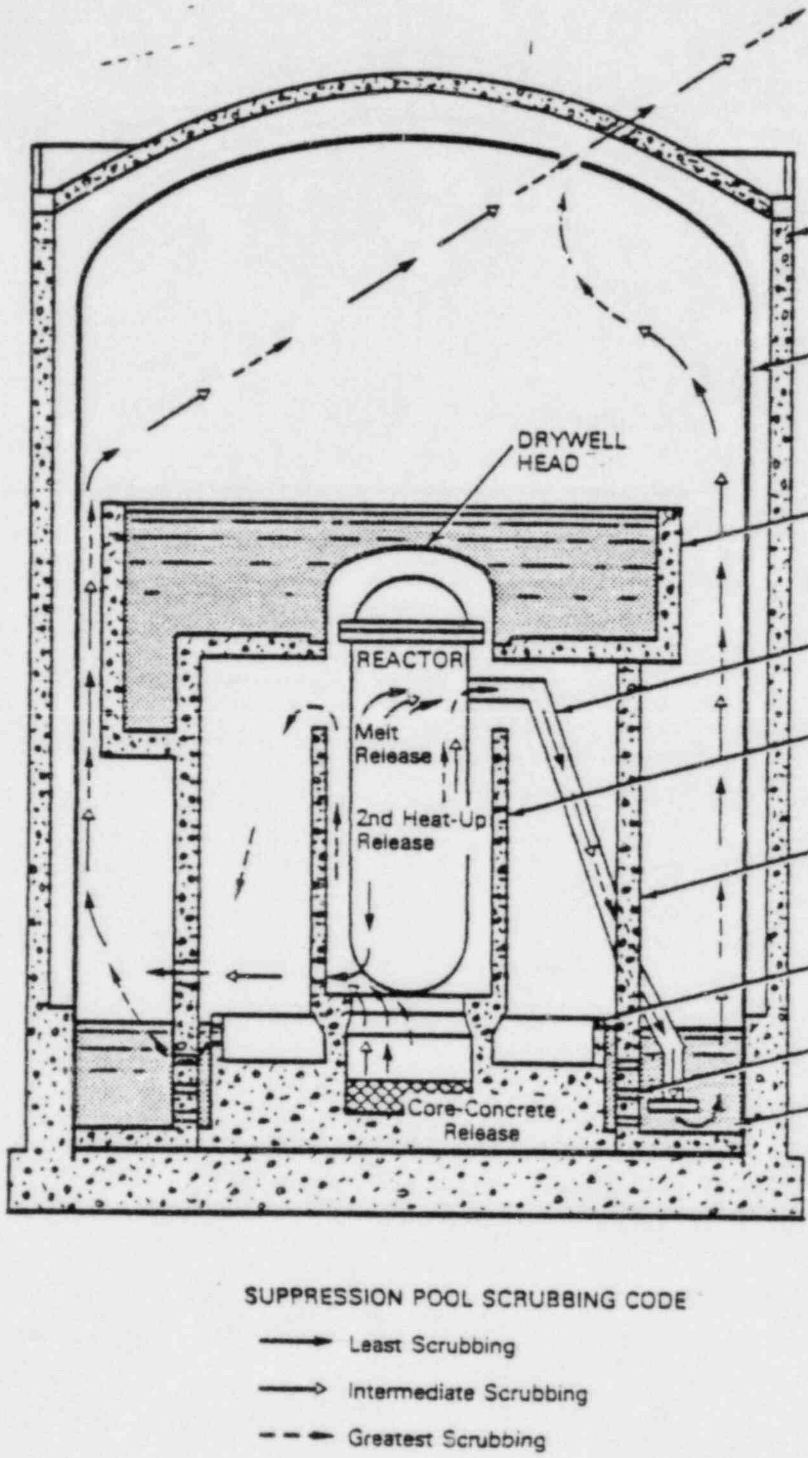

Figure 15.4 Relative degree of fission-product scrubbing by the suppression pool

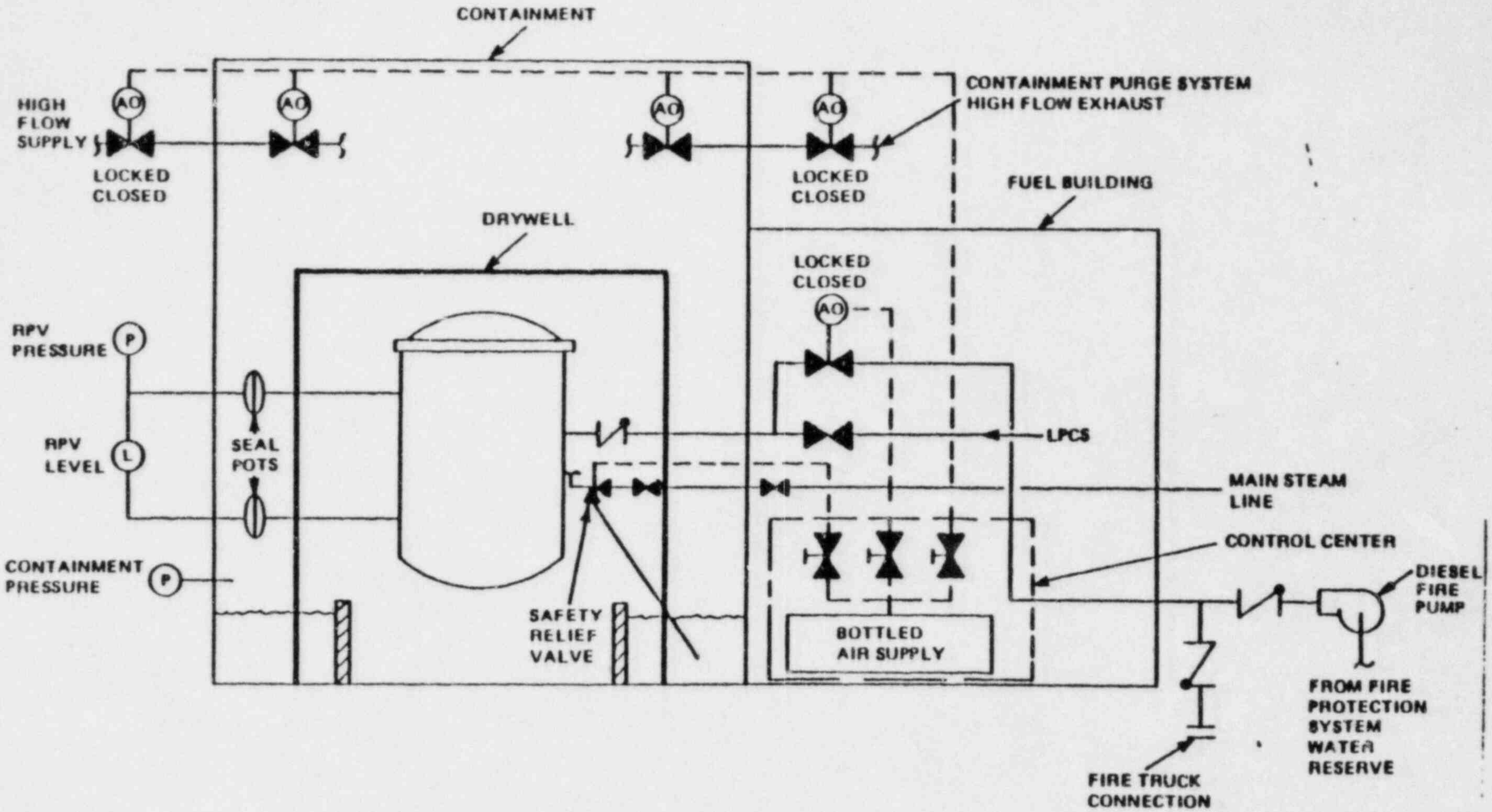

Figure 15.5 GESSAR II ultimate plant protection system

Table 15.1a Ranking of BNL and GESSAR II PRA sequences by core-damage frequency

Sequence	Staff/BNL estimates of the frequency of dominant accident sequences*			
	MAAC site		National average site	
1. $T_E UV$	1.5E-5	(68%)	3.0E-5	(79%)
2. $T_F QW$	1.9E-6	(8.6%)	1.9E-6	(5.0%)
3. $T_E W$	7.8E-7	(3.5%)	1.6E-6	(4.2%)
4. $T_F C_M U_H P_A$	8.7E-7	(3.9%)	8.7E-7	(2.3%)
5. $T_F C_M L_H$	7.6E-7	(3.4%)	7.6E-7	(2.0%)
6. $T_F QUX$	5.3E-7	(2.4%)	5.3E-7	(1.4%)
7. $T_{DC} UV$	4.3E-7	(1.9%)	4.3E-7	(1.1%)
8. $T_F C_M C_1 P_A$	3.9E-7	(1.8%)	3.9E-7	(1.0%)
9. $T_F C_M UP_A$	1.8E-7	(0.8%)	1.8E-7	(0.5%)
10. $T_F C_M P_1 U_M P_A$	1.1E-7	(0.5%)	1.1E-7	(0.3%)
11. $T_F C_M C_1 C_{21}$	1.1E-7	(0.5%)	1.1E-7	(0.3%)
12. $T_F C_M U$	1.0E-7	(0.4%)	1.0E-7	(0.3%)
13. $T_I UX$	1.0E-7	(0.4%)	1.0E-7	(0.3%)
Total**	2.2E-5		3.8E-5	

*1.5E-5 = 1.5×10^{-5}.
**Total of all sequences considered, not just the dominant sequences listed.

Table 15.1b: Ranking of GESSAR II sequences by core-damage frequency

Sequence	GESSAR II original GE estimates of the frequency of dominant accident sequences*
1. T_EUV	4.0E-6 (90.0%)
2. T_EP_1UV	1.2E-7 (2.6%)
3. T_EUX	1.3E-7 (2.1%)
4. T_FP_1UV	1.6E-8 (0.4%)
5. T_FQUX	1.3E-8 (0.3%)
6. T_IUV	1.2E-8
Total	4.3E-6

*1.5E-5 = 1.5×10^{-5}.

DEFINITION OF SEQUENCE TERMS

T_{DC}	Loss of two dc buses (Divisions 1 and 2)
T_E	Loss of offsite power
T_F	Isolation
T_I	Inadvertent open relief valve
U_H	High-pressure core spray system
U_R	Reactor core isolation cooling system
U	Failure of HPCS and RCIC coolant injection
X	Failure of timely ADS actuation
V	Low-pressure ECCS unavailable
C_M	Mechanical failure to scram
C_E	Electrical failure to scram
C_{21}	Two standby liquid control loops
C_1	One standby liquid control loop
L_H	Level control
P_1	One stuck-open relief valve (SORV)
P_A	ADS inhibit
Q	Feedwater system
W	Containment heat-removal function (including residual heat removal system and power conversion system)

Table 15.2 Class 2 transient containment analysis*

	First stage		Second stage		Reduction factor	
Transient	GESSAR II	Staff/BNL	GESSAR II	Staff/BNL	GESSAR II	Staff/BNL
CT2T		1.8E-5		3.8E-6		4.7
CT2L		1.4E-7		1.4E-8		10.0
CT2A		7.4E-7		1.2E-7		11.7

*1.0E-6 = 1.0×10^{-6}.

Proprietary Information Withheld From Public Disclosure Pursuant to 10 CFR 2.790(d)

Table 15.3 Internal flood precursors

Flood	Frequency*
Maintenance-induced floods	
1. RCIC system	1.5E-4
2. HPCI system	1.5E-4
3. Core spray	2.1E-5
4. LPCI system	4.1E-5
5. Service water system	2.1E-5
Rupture-induced floods	
1. HPCI discharge	4.8E-5
2. Core spray discharge	9.4E-5
3. LPCI discharge	2.0E-5
4. Service water	8.0E-5
5. Fire protection system	2.3E-5
6. RCIC suction	6.5E-5
7. HPCI suction	3.4E-5
8. Core spray suction	4.6E-5
9. LPCI suction	4.8E-5

*1.5E-4 = 1.5×10^{-4}.

Table 15.4 GESSAR II risk results by release category[1]

Release category	Event[2]	Mean latent fatalities	Mean person-rem	Probability[3]
Total	-	-	-	4.3E-6

Proprietary Information
Withheld From Public Disclosure
Pursuant to 10 CFR 2.790(d)

[1]This table is reproduced from GESSAR II.

[2] T = transcients
SB = small breaks
LB = large breaks
A = anticipated transients without scram

(See Tables 15.10 and 15.11 for containment release time and pool scrubbing coding shown as last two characters in event coding.)

[3]5.6E-7 = 5.6×10^{-7}.

Table 15.5 GESSAR II risk by event tree[1]

Type of event[2]	Core damage event frequency (event/yr)[3]	Public exposure[3,4] (person rem/event)	Annualized public risk (person-rem/yr)
Transients w/o bypass	3.7E-6	52E3	0.193
Transients with limited bypass[5]	9.5E-7	70E3	0.068
Breaks w/o bypass	1.6E-9	60E3	0.0001
Breaks with limited bypass[6]	7E-10	1300E3	0.0009
Containment cooling loss	2E-8	10E3	0.0002
ATWS	5E-8	40E3	0.002
Totals	4.7E-6	-	0.265

[1]From GESSAR II.
[2]Includes core damage and containment failure.
[3]3.7E-6 = 3.7×10^{-6}; 52E3 = 52×10^{3}.
[4]Assumes WASH-1400 site 6 with 81.6 million people within a 500-mile radius.
[5]Includes drywell bypass leakage after RPV failure only.
[6]Includes breaks outside the drywell and drywell bypass leakage.

Table 15.6 Representative parameters critical to determination of pool decontamination factor

Parameter	In-vessel release		Ex-vessel release, early ATWS
	Early Class I	Late ATWS	
Particle density (gm/cm)	2.9	2.8	2.8
Hydrogen flow (gm/s)	60	8.7	55
Steam flow (gm/s)	3600	310	350
CO flow (gm/s)	0	0	1400
CO_2 flow (gm/s)	0	0	1500
Pool temperature (°C)	55	97	100
Pool depth (cm)	610	610	150
Bubble diameter (cm)	.75	.75	.75
Aspect ratio	1.5	1.5	1.5
Decontamination Factors (DFs)			
Pool DF min (staff)	90	20	6
Pool DF max (staff)	10,000	10,000	600
Pool DF (GE)	10,000	10,000	600

Table 15.7 Frequency of core damage for various accident classes

Class[1]	GESSAR II PRA[2]	BNL initiators[2]	
		MAAC loop	National average loop
CT1-T	5.8E-8	1.1E-6	1.1E-6
CT1-P_a	1.7E-6	2.5E-6	5.0E-6
CT1-P_b	2.5E-6	1.25E-5	2.5E-5
CT2-T	3.3E-8	2.9E-6	3.8E-6
CT3	7.5E-10	1.3E-7	1.3E-7
CT4	1.1E-8	3.0E-6	3.1E-6
CTA	2.4E-10	1.2E-7	1.2E-7
CT1-L	3.1E-9	3.0E-9	3.0E-9
CT2-L	3.4E-9	1.4E-8	1.4E-8
CT5	1.9E-11	2.3E-11	2.3E-11
CT6	3.4E-10	1.2E-9	1.2E-9
Total accident sequence	4.4E-6	2.2E-5	3.8E-5

[1]See Table 15.8 for class definitions.
[2]5.8E-8 = 5.8×10^{-8}.

Table 15.8 Containment event trees[1]

Class	Tree name	Description

**Proprietary Information
Withheld From Public Disclosure
Pursuant to 10 CFR 2.790(d)**

[1]Reproduced from Table C.16-3 of the GESSAR II PRA.

[2]The frequency associated with this event is relatively small and does not justify an individual tree. This sequence was processed by other trees.

Table 15.9 Containment failure modes identified in GESSAR II

Failure mode	Description
γ	Slew containment static overpressurization (on the order of hours) caused by either noncondensible gas generation during core-concrete interaction, or steam generation following loss of containment heat removal
γ'	Loss of containment integrity caused by a continuous burn
γ''	Fast containment static overpressurization (within seconds to minutes) caused by global hydrogen combustion
μ	Containment dynamic overpressurization (within a fraction of a second) caused by local hydrogen detonation
μ'	Containment dynamic overpressurization caused by global hydrogen detonation
δ,δ'	Loss of drywell integrity caused by continuous ourn of piping or guard pipes

Table 15.10 Containment release time coding

Accident class	Release time	Description
I and III	E	Early release, where containment integrity is lost shortly after core damage
	I	Interim release, when loss of containment integrity occurs sometime after core damage
	L	Late release, when containment integrity is lost after a slow overpressurization by noncondensible gases
II	B	When lengthy loss of heat removal causes loss of containment integrity followed by core damage
	C	Core damage leading to loss of containment integrity by overpressure
IV	F	Fast loss of containment integrity followed by core damage
	S	Slower loss of containment integrity followed by core damage

Table 15.11 Suppression-pool scrubbing code

Code	Relative degree of release scrubbing
1	Suppression pool scrubbing of the in-vessel melt release; all other releases (second in-vessel heatup and ex-vessel core/concrete interaction release) bypass the pool
2	Scrubbing of in-vessel melt and second heatup releases until RPV melt through. Ex-vessel core/concrete release bypasses the pool
3	Continuous scrubbing of all releases - in-vessel melt and second heatup, and ex-vessel core concrete

Table 15.12 Potential suppression-pool-bypass paths

Line	Isolation barrier
From RPV to containment:	
1" instrument line	Orifice
1½" SLC line	Check valves (2)
1" CRD lines	Ball valve (1), drives
3/8" sample lines	Ball valves (2)
From RPV to secondary containment:	
20" RHR shutdown cooling	Check valves (2), motor-operated globe valve (1)
10" RCIC steam line	Motor-operated gate valves (2)
6" RCIC pump discharge	Air-operated stop check (2)
12" HPCS pump discharge	Air-operated stop check (1), motor-operated gate valve (1)
14" LPCI/LPCS discharge	Air-operated stop check valve (1), motor-operated gate valve (2)
6" RWCW lines	Motor-operated gate valves (4)
From RPV to outside secondary containment:	
26" main steam	Air-operated gate valves (3), motor-operated gate valve (1)
20" feedwater	Check valves (2), motor-operated gate valve (1)
3" main steam drain	Motor-operated gate valves (3)
4" RWCU to main condenser	Motor-operated gate valves (3), regulator, check valves (2)
From drywell to containment:	
10" vacuum relief	Air-operated butterfly valve (1), check valve (1)
TIP guide tubes	Drives
Airlock/equipment hatch	Seals
Guard pipe failure	Piping
From drywell to secondary containment:	
2" drywell bleedoff	Motor-operated gate valve (1)
From drywell to outside secondary containment:	
6" drywell cooling water	Motor-operated gate valves (2)
Postaccident gas sample	Motor-operated gate valves (2), orifice, solenoid-operated gate valve (1)

Table 15.13 In-vessel release fractions for Class I transients

Nuclide	WASH-1400[1]	GE[1]	BMI[1]	BNL[1]
Xe-Kr	9.0E-1	Proprietary Information Withheld From Public Disclosure Pursuant to 10 CFR 2.790(d)	1.0	9.09E-1
OI	6.3E-3			3.0E-4
I	8.94E-1		1.0	9.09E-1
Cs	8.1E-1		1.0	9.09E-1
Te	1.5E-1		4.3E-1	5.06E-1[3] 8.41E-1[4]
Ba	1.0E-1		3.7E-1	3.75E-1
Ru	3.0E-2		2.0E-2	3.66E-2
La	3.0E-3			1.9E-5

[1]9.0E-1 = 9.0×10^{-1}.
[2]Organic (penetrating) forms of iodine.
[3]Using Te release data from Lorenz, Beahm, and Wichner (1983).
[4]Using Te release data from NUREG-0772.

Table 15.14 Fraction of fission products retained in primary system after vessel release

Nuclide category	Held-up fraction	
	High	Low
Xe-Kr	0.0	0.0
OI	0.0	0.0
I	.25	.01
Cs	.55	.25
Te	.95	.85
Aerosols	.8	.4

Table 15.15 Percentage of fission products leaving primary system released during core/concrete interaction

Nuclide[1]	High (percent)	Low (percent)
Te	50	20
Ba	20	10
Sr	50	10
Tc	40	10
Ce[2]	1	
La[3]	2	
Xe-Kr	100	100
Cs	100	100
I	100	100

[1]Release of Ru, Mo, and Rh is essentially neglible.

[2]Ce represents Ce, Pu, and Np

[3]La represents La, Y, Sm, Pr, and Nd

Table 15.16 GESSAR II release fractions

Release characteristic and groups	GE			BNL			
				1-SB-E1			
	1-SB-E1	ATWS	2-T-L3	HIGH	LOW	ATWS	2-T-B3
Characteristics							
Release time (hr)				.56	.56	1.17	31.5
Release duration (hr)				11.64	11.64	10.0	15.8
Warning time (hr)		Proprietary Information		.3	.3	0.6	21.6
Release energy (10^6 cal/s)		Withheld From Public Disclosure Pursuant to 10 CFR 2.790(d)		1.1	1.1	15.0	4.4
Radionuclide groups							
(1) Kr, Xe				1.0	1.0	1.0	1.0
(2) I				3E-2	1E-2	3E-2	2E-2
(3) Cs, Rb				3E-2	1E-2	2E-2	2E-2
(4) Te, Sb		Proprietary Information		4E-2	5E-3	9E-3	9E-3
(5) Sr		Withheld From Public Disclosure		1E-2	6E-3	4E-3	4E-3
(6) Ba		Pursuant to 10 CFR 2.790(d)		3E-2	6E-3	7E-3	7E-3
(7) Tc				2E-2	6E-3	4E 3	6E-3
(8) Mo, Ru, Rh, Co*				7E-6	3E-7	3E-4	2E-4
(9) La, Y, Pr, Nd				1E-3	6E-7	2E-4	3E-4
(10) Ce, Pu, Np, Am				6E-4	6E-6	1E-4	1E-4
(11) Co**, Cm, Zr, Nb				6E-7	6E-9	1E-7	1E-7

* In GE analysis.
**In staff analysis.
NOTE: 7.3E-3 = 7.3×10^{-3}.

Table 15.17 Damage indices[1] for GESSAR II

Release category	Early fatality (persons)	Early injury (persons)	Latent fatality (persons)	Person-rem[2]
1-SB-E1	0.006	10	600	9E6
2-T-B3	0	0	300	5E6
ATWS	0	1	400	6E6

[1]The damage indices shown are the conditional mean values for upper-range source-term consequence calculations.
[2]9E6 = 9×10^6.

Table 15.18 GESSAR II potential design improvements - cost-benefit ranked listing (General Electric assessment, NEDE-30640)

Rank	Design modification	Calculated cost-benefit
1	Larger battery capacity for 10-hr blackout (3.10.a)*	< 10
2	Ultimate plant protection system (UPPS) (App. A)	< 10
3	Improved or additional low-pressure system (3.2.e)	< 10
4	AC bus cross-ties (3.9.c)	< 10
5	Improved maintenance procedures/manuals (3.1.c)	< 10
6	Computer-aided instrumentation (3.1.b)	< 10
7	Alternate pump power source (3.8.j)	< 10
8	Batteries for dc pump power (3.10.c)	< 10
9	Larger battery capacity for 16-hr blackout (3.10.1)	< 10
10	Simulator training for severe accidents (3.1.h)	< 10
11	Improved high-pressure system (3.2.a)	< 20
12	DC bus cross-ties (3.10.d)	< 20
13	Additional active HP system (3.2.b)	< 50
14	Uninterruptable power supplies (3.9.b)	< 50
15	Fuel cells for diverse dc pump power (3.10.c)	< 50
16	Additional diesel generator (3.9.a.1)	< 50
17	Gas turbine (3.9.d)	< 50
18	Passive HP system (3.2.c)	< 50
19	Steam-driven turbine generator (3.9.f)	< 50
20	Increased electrical divisions/diesels (3.9.a.2)	<100
21	Increased design margin (3.12.b)	<100
22	Jockey pump system (3.2.g.1)	<100
23	Reduction in common cause dependencies (3.8.c)	<150
24	Passive ultimate heat sink (3.4.b)	<150
25	Improved operating response (3.8.b)	<150

*Section in GE report NEDE-30640, where design option is discussed.

APPENDIX A

CONTINUATION OF CHRONOLOGY

July 18, 1983	Letter from GE transmitting Amendment 18 to GESSAR II regarding postaccident monitoring instrumentation and stress-corrosion cracking.
July 20, 1983	Letter to GE transmitting Supplement 1 to SER (NUREG-0979).
July 22, 1983	Letter from GE transmitting proprietary portion of Amendment 19 to GESSAR II regarding human factors.
July 22, 1983	Letter from GE transmitting nonproprietary portion of Amendment 19 updating Section 3.11 on environmental qualification and responding to Outstanding Issue 3 to GESSAR II SER (NUREG-0979).
July 26, 1983	Letter from GE transmitting responses to action plan for resolving containment design issues.
July 27, 1983	Letter to GE transmitting Final Design Approval FDA-1 for GESSAR II BWR/6 nuclear island design.
August 1, 1983	Letter from GE transmitting responses to BNL/NRC questions.
August 17, 1983	Letter from GE forwarding Impell Corporation report on confirmatory soil-structure interaction analysis.
August 22, 1983	Letter from GE informing that drywell structure drawings will be used in review of PRA regarding containment structural system pressure-carrying capacity.
September 1, 1983	Letter to GE transmitting Supplement 1 to NUREG-0979 regarding GESSAR II.
September 2, 1983	Letter to GE requesting additional information on severe-accident review.
September 9, 1983	Letter from GE transmitting supplemental containment vessel dimensions needed in review of GESSAR II PRA regarding containment structural system pressure-carrying assessment.
September 14, 1983	Letter from GE transmitting proprietary response to request for additional information on severe accident portion of GESSAR II.
September 21, 1983	Letter from GE transmitting GESSAR II Seismic Event Analysis.

October 4, 1983	Letter from GE transmitting proprietary portion of Amendment 20 to GESSAR II regarding environmental qualification of safety-related equipment.
October 4, 1983	Letter from GE transmitting Amendment 20 responding to Confirmatory Issue 18 of SER, Supplement 1, revising responses to letter on communication and lighting systems and updating interface tables.
November 7, 1984	Letter from GE transmitting proprietary information regarding "GESSAR II Fire and Flood External Event Analysis" in response to draft policy statement.
November 17, 1983	Letter from GE transmitting "GESSAR II Internal Event PRA Uncertainty Analysis" in support of the severe accident review.
December 5, 1983	Letter from GE transmitting marked-up draft Appendix 15E pertaining to "Station Blackout Capability."
December 29, 1983	Letter from GE transmitting proprietary draft "GESSAR II Seismic Event Uncertainty Analysis."
January 19, 1984	Letter from GE transmitting proprietary information regarding seismic fragility analysis.
January 26, 1984	Letter from GE transmitting proprietary response to request for additional information on severe accident review.
January 31, 1984	Letter from GE transmitting proprietary draft amendment regarding leak-before-break approach.
February 1, 1984	Letter from GE transmitting proprietary response to request for additional information on rationale for treatment of fire and flood event uncertainty analysis.
March 13, 1984	Letter to GE regarding comments on "Severe Accident Program for Nuclear Plant Regulation" (NUREG-1070).
April 13, 1984	Letter to GE regarding requirement to assess improvements in the reliability of core and containment heat removal systems. Potential design improvements included for consideration.
April 20, 1984	Letter from GE responding to April 13, 1984, request for additional information. Design modifications will be evaluated per "Measures of Risk Importance and Their Applications" (NUREG/CR-3385).
April 20, 1984	Letter from GE transmitting response to requests for additional information on seismic PRA review and internal flood PRA review.

May 30, 1984	Letter from GE transmitting response to GE commitment in Section 7.2.2.2 of SER regarding testing of optical isolators.
June 21, 1984	Letter from GE transmitting "Evaluation of Proposed Modifications to GESSAR II Design" (NEDE-30640) in response to request for additional information on specific potential design improvements including consideration of risk reduction capability and cost/benefit assessment.
July 11, 1984	Letter from GE transmitting containment structural analysis plan to support conclusions reached in Appendix G to PRA on containment failure mode and pressure.
July 12, 1984	Letter from GE transmitting "GESSAR II External Event Risk" providing qualitative assessment of risk from hurricanes, tornados, external floods, aircraft strike, and hazardous materials.
July 13, 1984	Letter from GE transmitting "Resolution of Applicable Unresolved Safety Issues and Generic Issues for GESSAR II" (NEDO-30670), providing technical resolution of issues.
July 13, 1984	Letter from GE transmitting information regarding nuclear island/balance-of-plant interfaces, including interface assumptions in PRA and documentation of design evolution.

APPENDIX B

REFERENCES*

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Letter, December 9, 1980, from NRC to all BWR licensees, "BWR Scram Discharge System."

---, February 25, 1983, from J. S. Charnley (GE) to F. J. Miraglia (NRC), "Proposed Revision to GE Licensing Topical Report NEDE-24011-P-A."

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*The entries in this appendix represent a continuation of Appendix B in the GESSAR II SER (NUREG-0979) and SSER 1 (NUREG-0979, Suppl. No. 1).

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---, November 23, 1983, from J.S. Charnley (GE) to C. H. Berlinger (NRC), "Post-Irradiation Fuel Surveillance Program."

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APPENDIX C

UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES

Resolution of unresolved safety issues (USIs) and high/medium priority generic issues (GSIs) is required by draft NUREG-1070. The status of the USIs is found in NUREG-0606, "Unresolved Safety Issues Summary." The status of generic safety issues is discussed in NUREG-0933, "A Prioritization of Generic Safety Issues," along with the proposed resolution schedule.

It must be acknowledged that staff evaluations of the USIs and GSI is ongoing. Rather than await generic resolution, severe-accident certification for GESSAR II requires plant-specific findings at this time, to the extent possible. Since staff findings for many USIs and GSIs are, at most, preliminary, it is not possible to evaluate the GESSAR II design against established staff criteria. Therefore, resolution of the issues will be demonstrated through engineering evaluations and demonstration that: (1) the subject USI or GSI is not applicable to the GESSAR II design, (2) GE risk assessment (or engineering analysis) shows insignificant societal risk arising from the issue, or (3) where the risk assessment or engineering analysis cannot demonstrate insignificant societal risk, the design incorporates features which adequately respond to all concerns inherent in the issue.

Unresolved safety issues and generic safety issues which were outstanding when the GESSAR II severe-accident evaluation was undertaken, are discussed below, along with staff conclusions regarding their resolution.

UNRESOLVED SAFETY ISSUES

USI A-1: Water Hammer

Since 1969 more than 150 incidents involving water hammers in BWRs and PWRs have been reported. The water hammers (or steam hammers) have involved steam generator feedrings and piping, the RHR system, ECC systems, and containment spray, service water, feedwater, and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, there have been several incidents which have resulted in piping and valve damage. USI A-1 deals with the technical resolution of safety concerns related to the occurrence of water hammer in nuclear power plants.

The staff has completed its evaluation of USI A-1 (NUREG-0927, Rev. 1, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants, Technical Findings"). The staff concluded that the frequency and severity of waterhammer occurrences has been significantly reduced through (1) incorporation of design features such as keep-full systems, vacuum breakers, J-tubes, void detection systems, and improved venting procedures, (2) proper design of feedwater valves and control systems, and (3) increased operator awareness and training. Thus the water hammer issue at the present is less significant than was suggested by the water hammer occurrences in the early and mid-1970s.

The revised Standard Review Plan sections resulting from USI A-1 represent proven design concepts and operation of considerations for avoidance of water hammer and will be used only for review of "custom plant" CP applications, and for standard plant applications docketed after the issuance of these Standard Review Plan section revisions, which are intended for referencing in CP applications. These revisions represent current staff review practices (already used in current case reviews).

The GESSAR II design was subjected to such a review during its licensing evaluation. A summary of the GESSAR II design capability for preventing or mitigating water hammer is described below.

In order to protect the GESSAR II emergency core cooling systems (ECCSs) (Section 6.3.2.2.5 of the GESSAR II) against the effects of water hammer, the ECCSs are provided with jockey pumps. These jockey pumps keep the ECCS lines full of water up to the motor-operated injection valves so that the ECCS pumps will not start pumping into voided lines. In addition, to ensure that the ECCS lines remain full, vents have been installed and filling procedures have been established. Further assurance for filled discharge piping is provided by pressure instrumentation that is used to initiate an alarm that sounds in the main control room if the pressure falls below a predetermined setpoint indicating difficulty maintaining a filled discharge line. Should this occur, or if an instrument becomes inoperable, the required action is identified in the technical specification.

To provide additional protection against potential water hammer events in GESSAR II plants, piping design codes require consideration of impact loads. Approaches used at the design stage include: (1) avoiding rapid valve operation; (2) piping layout to preclude water slugs in steam-filled lines; (3) using snubbers and pipe hangers; and (4) using vents and drains. The use of snubbers and pipe hangers are a byproduct of protection from seismic loads however, their use helps to mitigate the effects of waterhammer events.

In addition, a preoperational vibration- and dynamic-effects test program will be conducted by the applicant, in conjunction with GE, in accordance with Standard OM-3 of the American Society of Mechanical Engineers for all Class 1, Class 2, Class 3, and other piping systems and piping restraints.

These tests will provide adequate assurance that the piping restraints have been designed to withstand dynamic effects of valve closures, pump trips, and other operating modes.

Nonetheless, in the unlikely event that a pipe break did result from a severe waterhammer event, core cooling is assured by the emergency core cooling systems and protection is provided against the dynamic effects of such pipe breaks inside and outside of containment. Any applicants referencing GESSAR II will be committed to the design concepts and operational procedures required by the revised Standard Review Plan sections for those areas of plant design outside the scope of GESSAR II. Therefore the staff considers USI A-1 resolved for GESSAR II.

USI A-17: Systems Interaction

The design, analysis, and installation of systems in a nuclear power plant are frequently the responsibility of teams of engineers with functional specialties--such as civil, electrical, mechanical, or nuclear. Experience at operating plants has led to questions of whether the work of these functional specialists is sufficiently integrated to enable them to minimize adverse interactions among systems. Some adverse events that occurred in the past might have been prevented if the teams had ensured the necessary independence of safety systems under all conditions of operation.

GE has not described a complete or comprehensive program that separately evaluates all safety-related structures, systems, and components for adverse systems interactions. The GESSAR II nuclear island was reviewed against the Standard Review Plan (NUREG-0800) which contains the regulatory criteria for the interdisciplinary reviews. The staff's evaluations of those areas as per the SRP are provided in the SER for GESSAR II (NUREG-0979).

While GE has not described a separate program addressing systems interactions, GE states that provisions are included in the PRA methodology to identify commonalities and dependencies that could result in adverse systems interactions. These provisions included using the minimal cutsets derived from system-level fault trees that were linked through event trees developed for the PRA event sequences. The procedure calls for the use of a consistent nomenclature for basic components and events for all systems throughout the plant and to identify commonalities and dependencies whenever the same basic item occurred as an element in cutsets of different systemic fault trees.

The GE effort to identify common-cause events, common-mode failures, and inter-system dependencies has gone beyond the licensing basis to address the systems interaction issue for the GESSAR II design and is being done in advance of the issuance of any formal NRC guidance or requirements. In the absence of criteria and requirements, no conclusions can be made concerning the adequacy and completeness of GE's additional work.

On the basis of experience with the systems interaction issue, the staff identified the following concerns:

(1) The system-level failure modes and effects analyses considered only the failure effects within a system.

(2) The RPS, RCIC, RHR, remote shutdown, SBGT, and some HVAC systems were excluded from the failure modes and effects analyses.

(3) The balance-of-plant systems upon which the GESSAR II systems depend were not within the scope of the GE efforts.

(4) Spatially coupled systems interactions could not be analyzed because the GESSAR II design is yet to be constructed.

GESSAR II has been evaluated against current licensing requirements that are founded on the principle of defense in depth. Adherence to this principle results in requirements such as physical separation and functional independence of redundant safety equipment.

Considering GE's PRA analysis and GE's compliance with current SRP guidelines, the staff finds that some assurance exists that adverse systems interactions that pertain to GESSAR II design will be minimized; however since systems interaction is an issue that applies to complete plant designs, the staff will require that the systems interaction and PRA studies be completed by applicant-performed programs that supplement the work that GE has done on the nuclear island. The final assurance must be deferred until an applicant makes reference to the GESSAR II design. The applicant must either address the above concerns or comply with any requirements produced from the resolution of USI A-17.

USI A-40: Seismic Design Criteria

NRC regulations require that nuclear power plant structures, systems, and components important to safety withstand the effects of seismic events. Detailed requirements and guidance regarding the seismic design of the plants are provided in NRC regulations and regulatory guides. However, there are a number of plants with licenses that were issued before NRC's current regulations and guides were in place. Task A-40 is an effort to reevaluate the older plants to assure no undue public risk is involved and to make revisions to the Standard Review Plan (SRP) and regulatory guides to bring them in line with the state of the art in seismic design requirements. A-40 basically consists of a number of seismic design criteria changes which upgrade the SRP to reflect advanced technical knowledge and in some cases to reflect current industry practice. All changes but one are proposed to be applied to new CP and PDA applications. The exception is above-ground free-standing tanks where backfit to operating plants is proposed to ensure that proper design loads were used in existing tank designs.

GESSAR II has been evaluated against the latest seismic design criteria and all balance-of-plant designs used in the application referencing the GESSAR II design will be evaluated to the latest criteria as a result of USI A-40 resolution; therefore, this issue is resolved for GESSAR II.

USI A-43: Containment Emergency Sump Reliability

Following a postulated loss-of-coolant accident, that is, a break in the reactor coolant system piping, the water flowing from the break would be collected in the suppression pool. This water would be recirculated through the reactor system by the emergency core cooling pumps to maintain core cooling. This water might also be circulated through the containment spray system to remove heat and fission products from the drywell and wetwell atmosphere. Loss of the ability to draw water from the suppression pool could disable the emergency cooling and containment spray systems.

The principal concerns are somewhat interrelated but are best discussed separately. The first concern deals with the various kinds of insulation used on piping and components inside of containment. The safety concern is that the LOCA would destroy insulation and that this insulation debris could block the

RHR suction strainers or otherwise adversely affect the net positive suction head (NPSH) requirement of the pumps, block spray nozzles, and degrade the safety systems performance.

The second concern deals with the hydraulic performance of the RHR intakes as related to the hydraulic performance of safety systems supplied therefrom. Extensive full-scale experiments have been performed to assess air ingestion and other adverse hydraulic conditions. The results for BWRs (NUREG/CR-2772) show that air ingestion is generally less than 0.5% when the Froude number at the suction intake is less than 0.8. These test data can be used instead of requiring in-plant preoperational tests relying on vortex observations.

The staff is presently reviewing the GESSAR II containment design with regard to USI A-43 and will report on its resolution in a future supplement to the SER.

USI A-44: Station Blackout

Electrical power for safety systems at nuclear power plants must be supplied by at least two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical division for safety systems includes two offsite alternating current (ac) power connections, a standby emergency diesel generator ac power supply, and direct current (dc) sources.

The term "station blackout" refers to the complete loss of ac electric power to the essential and nonessential buses in a nuclear power plant. Station blackout therefore involves the loss of offsite power concurrent with the failure of the onsite emergency ac power system. Because many safety systems required for core decay heat removal and containment heat removal are dependent on ac power, the consequences of station blackout could be severe.

USI A-44 involves a study of whether or not nuclear power plants should be designed to withstand an extended station blackout. This issue arose because of the accumulated experience regarding the reliability of ac power supplies. There have been numerous instances of emergency diesel generators failing to start and run in response to tests conducted at operating plants. In addition, a number of operating plants have experienced a total loss of offsite electrical power, and more occurrences are expected in the future. In almost every one of these loss-of-offsite-power events, the onsite emergency ac power supplies were available immediately to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power souces has been unavailable. In a few cases there has been a complete loss of ac power, but during these events, ac power was restored in a short time without any serious consequences.

The major areas of study in A-44 included the likelihood and duration of the loss of offsite power, the reliability of onsite emergency ac power sources, and the potential for severe accident sequences after a loss of all ac power. Significant factors that contribute to risk from station blackout events were identified and evaluated. On the basis of this evaluation, the staff has proposed recommendations to resolve this issue, but the resolution is not yet final.

The proposed resolution of A-44 would require nuclear power plants to be capable of coping with a station blackout for a specified duration. The duration would be determined on the basis of the following site-specific characteristics: (1) the redundancy of onsite emergency ac power sources (number of diesel generators available for decay heat removal minus the number needed for decay heat removal), (2) the reliability of onsite emergency ac power sources (e.g., diesel generator), (3) the frequency of loss of offsite power, and (4) the probable time to restore offsite power.

For generic resolution of A-44, the capability and capacity of all systems necessary to provide core cooling and decay heat removal for the duration of the station blackout should be assured. The following items should be included in this evaluation

- dc battery capacity
- condensate storage tank capacity
- compressed air capacity
- leakage from pump seals that could result in loss of reactor coolant inventory needed to maintain core cooling
- operability of necessary equipment in an environment resulting from a station blackout (i.e., without HVAC)

In addition to the above, the proposed resolution includes recommendations to improve and maintain the reliability of onsite emergency ac power sources at or above specified minimum levels.

A loss of all ac power was not a design-basis event for the GESSAR II nuclear island. If both offsite and onsite ac power are lost however, the plant does have the capability to respond successfully, for a limited time, by relying on various backup systems. GESSAR II can utilize a combination of safety/relief valves, dc power systems, and the reactor core isolation cooling (RCIC) system to remove core decay heat without reliance on ac power. These systems have the capability to ensure that adequate cooling could be maintained for at least 2 hours.

The loss of ac power for a period of time exceeding 2 hours has been analyzed in the GESSAR II PRA. This event was found to be a dominant contribution to core-damage frequency. This accident was found to contribute approximately 79% of the total core-damage frequency (as modified by BNL review). Although the relative frequency was still quite low (approximately 3×10^{-5} per reactor year), station blackout events were identified as fruitful areas for risk reduction efforts.

Further work by GE indicated a station blackout capability exceeding 10 hours is possible assuming credit for straightforward operator actions and potential design improvements. A preliminary assessment by BNL indicated that this would reduce core damage from internal events by a factor of approximately 2.

In addition to extended station battery capacity, GE has proposed an ultimate plant protection system (UPPS) which significantly improves the plant's capability to respond successfully to total station blackout events. Details of this system and of the proposed battery extended capability are discussed in Section 15.6.3 on design improvements. This modification, considered together with the ability to withstand a station blackout for 10 hours, gives the staff confidence that the resolution of USI A-44 has been achieved in a manner that will result in low public risk from the issue. This conclusion is confirmatory subject to the completion of the staff review of the UPPS and extended station battery capacity.

USI A-45: Shutdown Decay Heat Removal

The primary objectives of the USI A-45 program are to evaluate the safety adequacy of decay-heat removal (DHR) systems in existing light-water reactor (LWR) power plants and to assess the value and impact (or cost-benefit) of alternate measures for improving the overall reliability of the DHR function. The A-45 program is conducting probabilistic risk assessments and deterministic evaluation of those DHR systems and support systems required to achieve hot-shutdown and cold-shutdown conditions in both pressurized- and boiling-water reactors. Integrated systems analysis techniques are being used to assess the vulnerability of DHR systems to various internal and external events, including transients, small-break loss-of-coolant accidents, and special emergency challenges, such as fires, floods, earthquakes, and sabotage. State-of-the-art cost-benefit analysis techniques are being utilized to assess the net safety benefit of alternative measures to improve the overall reliability of the DHR system.

At this time, the staff in its safety assessment for generic resolution of A-45 considers the following alternative measures for improving the overall reliability of the DHR function:

(1) Improved operating and/or procedural changes that would strengthen the availability of decay-heat removal.

(2) In conjunction with (1) above, the staff will search for alternate paths for decay-heat removal wherein existing equipment is used in atypical modes of DHR (e.g., bleed and feed in PWRs).

(3) Add on dedicated shutdown decay-heat-removal systems.

The GESSAR II PRA indicated that shutdown cooling system failures (following a transient) accounted for less than 1% of the original PRA core-damage frequency from internal events. However, staff reassessment indicated core-damage contributions attributable to failures of the DHR systems is nearer to 7% of the total frequency.

Additionally, the PRA did not consider DHR system failures when the plant is in extended shutdown mode. Additional core-damage frequency contribution from this failure mode may exist; however, it probably would not exceed the contribution from the previous effect. Actual core-damage contribution because of RHR failures may, therefore, be a few percent of total core damage.

GE has also proposed an alternate diverse DHR system called the ultimate plant protection system (UPPS). The staff has not fully evaluated the capabilities of this system. However, it would appear to significantly enhance the ability to mantain decay-heat removal following extensive system failures from internal and external events. Since the staff seismic review is incomplete, these preliminary conclusions may be impacted. The staff will report on its UPPS evaluation in a future supplement to the SER. Therefore, because of the low contribution to the core-damage frequency attributable to DHR system failures, a favorable finding on the UPPS may demonstrate satisfactory resolution of USI A-45. The staff's conclusion on UPPS will be reported in a future supplement to the SER.

USI A-47: Safety Implication of Control Systems

This issue concerns the potential for accidents or transients being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration and would be in addition to any control system failure that may have initiated the event. It is generally believed that control system failures are not likely to result in loss of safety functions which could lead to serious events or result in conditions that safety systems are not able to cope with.

In-depth studies for all the nonsafety-grade control systems have not been performed, however, and there exists some potential for accidents or transients being made more severe than previously analyzed, as a result of some of these control system failures or malfunctions.

Two potential concerns have already been identified in which a failure or malfunction of the nonsafety-grade control system can (1) potentially cause a steam generator or reactor vessel overfill, or (2) can lead to a transient in which the vessel could be subjected to severe overcooling. In addition, there is the potential for an independent event like a single failure, or a common-mode event, to cause a malfunction of one or several control systems which would lead to an undesirable control action, or provide misleading information to the plant operator.

The staff is presently reviewing the GESSAR II design as it relates to USI A-47 and will address its resolution in a future supplement to the SER.

USI A-48: Hydrogen Control Measures

Postulated reactor accidents that result in a degraded or melted core can result in generation and release to the containment of large quantities of hydrogen. The hydrogen is formed from the reaction of the zirconium fuel cladding with steam at high temperatures and/or by radiolysis of water. Experience gained from the TMI-2 accident indicates a potential need to require more specific design provisions for handling larger hydrogen releases than are currently required by the regulations, particularly for smaller, low-pressure containment design.

The staff's current position which is proposed in the draft Commission Paper, "Amendments to 10 CFR Part 50 Related to Hydrogen Control," is to require hydrogen-control system, e.g., ignitor system to control a large quantity of hydrogen (75% metal-water reaction) for PWR ice condenser and Mark III BWR containments.

By letter dated August 20, 1984, GE submitted a draft amendment to GESSAR II, Sections 1G.12 and 1G.21. This draft amendment requires utility applicants referencing GESSAR II to provide an igniter hydrogen control system capable of handling hydrogen as required by the proposed Interim Requirements Related to Hydrogen Control (46 FRG 2281). The hydrogen control system will be based on the NRC staff approved results of the Hydrogen Control Owners Group tests and analyses. GE has also provided UPPS which will reduce the overall risk of core damage and the overall probability the hydrogen will be generated. The staff is reviewing GE's commitment on hydrogen control and UPPS and will provide its evaluation in a future supplement to the SER.

GENERIC SAFETY ISSUES

GSI A-29: Nuclear Power Plant Design for the Reduction of Vulnerability to Sabotage

Generic Safety Issue A-29 deals with the effectiveness of various nuclear power plant system designs to reduce vulnerability to sabotage. Although present reactor designs do provide some inherent protection against sabotage, extensive physical security measures are currently thought to be necessary to provide an acceptable level of protection. An alternate approach would be to more fully consider and integrate other possible means for reducing reactor vulnerabilities to sabotage and tampering, and their effects on plant safety, operability, reliability, maintainability, and physical security. The staff is currently considering such issues in its effort for generic resolution of A-29.

In the course of the A-29 review program, the staff did evaluate the vulnerability of the GESSAR II plant design to sabotage and tampering, by considering plant features that inhibit sabotage, the plant's capability to mitigate sabotage, and the balance between safety and safeguards. GE provided this assessment in Amendment 16 to GESSAR II.

The staff concluded that the GESSAR II design contains a number of features that limit vulnerability to sabotage. The combination of multiple and diverse means of providing makeup water to the reactor vessel along with the inherent natural circulation capability of the BWR reactor design and the suppression pool significantly inhibits sabotage. The system separation and the self-test and status monitoring system provide further inhibitors to sabotage. Redundancy of safety systems, system separation and the status monitoring limit the adverse impact of tampering. It is important to note that with the exception of the access control features, no plant system features were designed specifically to inhibit or mitigate sabotage. Thus, application of present regulatory requirements in other areas (e.g., flood, missile and fire protection, and system monitoring requirements) resulted in the significant level of sabotage protection provided by the GESSAR II design.

The above considerations lead the staff to judge that the risk from sabotage for the GESSAR II design is low and considers A-29 resolved for GESSAR II.

GSI A-30: Adequacy of Safety-Related DC Power Supplies

The dc power system in a nuclear power plant provides control and motive power to valves, instrumentation, emergency diesel generators, and many other components and systems during all phases of plant operation, including abnormal shutdowns and accident situations.

Assurance of a reliable dc power supply is subject to two concerns: (1) that the batteries and other system elements should remain in full operation-ready (not degraded) condition, and (2) that independence of the two redundant power divisions should be assured. An aspect of the potential significance of the issue is that failure of one power division would generally cause a reactor scram which could result in a demand for dc power to remove decay heat and prevent core damage.

An estimate of dc-initiated core-damage frequency for GESSAR II was given in BNL's report. BNL estimated the impact of a loss of two dc power divisions on the GESSAR II core-damage frequency to be approximately 4 x 10^{-7} per reactor year. Since there are four separate dc power divisions on the GESSAR II design, dc power failures are a low contributor to the frequency of core damage. This issue is resolved for the GESSAR II design.

GSI B-5: Ductility of Two-Way Slabs and Shells, and Buckling Behavior of Steel Containments

This issue is divided into two parts. One part involves the concern over the lack of information related to the behavior of two-way reinforced concrete slabs loaded dynamically in biaxial membrane tension, flexure and shear, when subjected to a postulated LOCA or high-energy-line break (HELB). If structures (concrete slabs) were to fail (floor or wall collapse) because of loadings caused by LOCA or HELB, there would be a possibility that other portions of the reactor coolant system or safety-related systems could be damaged.

The second part involves the concern over the lack of a uniform, well-defined approach for design evaluation of steel containments subjected to unsymmetrical dynamic loading. Section III of the ASME Code does not provide detailed guidance on the treatment of buckling of steel containment vessels for such loading conditions (i.e., earthquake, postulated LOCA, or HELB). If steel containment shells were to fail as a result of these loads which may cause buckling, one of the plant's levels of defense would be lost and could result in a radioactive release to the environment. A large LOCA or HELB near the containment wall could possibly provide such a load.

The staff concluded that there is sufficient information pertaining to the design of two-way slabs subjected to dynamic loads and biaxial tension to enable a reasonably accurate analysis and, therefore, a solution has been identified for this part of the overall issue. The staff's Safety Evaluation Report (SER) relating to the Final Design Approval of the GESSAR II nuclear island design (NUREG-0979) stated that the GESSAR II design meets the requirements of SRP Section 3.5.3. This SRP section provides the limits, criteria, and exceptions as appropriate to provide a conservative basis for engineering design to ensure that the structures or barriers are adequately resistant to and will withstand the effects of such forces. Therefore, this part of the issue is considered resolved for GESSAR II.

For the second part of the issue, that pertaining to the buckling behavior of steel containments, the staff has developed and is using an interim set of criteria for evaluating containment buckling for plants undergoing operating license review. GE commits to using the interim criteria for evaluating steel containment buckling. The loading needed to cause buckling would have to be due to a high-energy source such as a large LOCA or HELB near the containment wall as well as to earthquakes. GE proposed the resolution of this issue for GESSAR II to be forthcoming with GSI B-6 that follows.

GSI B-6: Loads, Load Combinations, Stress Limits

This issue concerns the design of structures, systems, and components which must accommodate individual loads and combinations of loads that can result from natural phenomena, normal operating conditions, and postulated accidents. Part of this issue has been resolved--the part which concluded that seismic loads and LOCA and SRV (safety/relief valve) loads on containment structures should continue to be combined. The only remaining work on this issue is research into decoupling the LOCA and safe shutdown earthquake (SSE) events for mechanical systems. Recently, combined loads were increased to further account for phenomena such as asymmetric blowdowns in PWRs because improved techniques have been developed for defining loading. These changes have raised questions concerning implementation of new regulations, increased construction costs, increased radiation exposure of maintenance crews performing increased inspections and maintenance, and reduced reliability of stiffer systems under normal operating transients.

The staff, in addressing the probability of an earthquake-induced large LOCA, published Research Information Letter No. 117 (NRC memorandum, April 10, 1981) that identified the following results:

(1) Through-wall cracks are about a million times more likely to occur than double-ended guillotine breaks, thus supporting the leak-before-break hypothesis.

(2) Fatigue crack growth due to all transients, including earthquakes, is an extremely unlikely mechanism for inducing a large LOCA. The contribution of earthquakes to the occurrence of this event is a small percentage of the total probability.

Although the above results are identified for PWRs, it is assumed that the results for BWRs are similar for this analysis. The proposed resolution for this issue is to decouple the SSE-LOCA load requirements which will permit the removal of some snubbers and pipe whip restraints. The removal of pipe restraints will improve access to many equipment areas and, as a result, will reduce the time plant personnel need to spend in high radiation areas, thus reducing occupational exposure. Removing snubbers will reduce the stiffness during normal operation resulting in a reduction in the probability of pipe rupture during normal operating transients.

It is expected that this issue will be resolved in early 1985 when the double-ended guillotine-break (DEGB) studies for Mark I plants by Lawrence Livermore National Laboratory (LLNL) and parallel studies by GE on Mark II and Mark III plants (GE letter, May 31, 1984) that support the leak-before-break approach will be complete. It is expected that these studies may support decoupling of LOCA and SSE events which will yield reductions in public risk and occupational exposure.

The staff is presently reviewing GESSAR II as it relates to GSI B-6 and will report on its resolution in a future supplement to the SER.

GSI B-10: Behavior of BWR Mark III Containments

The concern of this issue is that pool loads following a postulated LOCA may damage structures and components located within the wetwell. Although many of these structures (i.e., walkways) are not safety related, various ECCSs take suction from the wetwell and, therefore, damage in the wetwell may affect the performance of the ECCS.

The staff has reviewed GE's pool-dynamic-load definitions and has arrived at a hydrodynamic load definition that can be utilized by Mark III containment applicants for operating licenses. NUREG-0978 (August 1984) contains the staff acceptance criteria on this issue. In Section 6.2.1.8.3 of the SER the staff evaluated Appendix 3B of GESSAR II to the acceptance criteria now deferred in NUREG-0978. The staff found the procedures described in GESSAR II acceptable with certain exceptions. These exceptions are discussed in SER Section 6.2.1.8.3.

Resolution for GESSAR II is a commitment that a utility applicant referencing the GESSAR II design will address staff acceptance criteria for LOCA-related Mark III containment pool dynamic loads.

GSI B-17: Criteria for Safety-Related Operator Actions (SROAs)

This issue involves the development of a time criterion for SROAs, including whether or not automatic actuation will be required. Development and implementation of criteria for SROAs may result in the automation of some actions currently performed by operators if such actions are shown to be burdensome or to result in a high likelihood of error because of the short time available to accomplish them. Automation of these actions may reduce the expected frequency of core-damaging events and, therefore, public risk.

The GESSAR II PRA includes estimates of human error probability based on the time available to accomplish the actions in accordance with the guidance in NUREG/CR-1298.

GE stated that during the control room design review for GESSAR II, a BWR/6 simulator was used to ascertain whether there were any safety-related operator actions for which limited time was available. The emergency procedure guidelines also consider the time available for SROAs in their development, on the basis of the BWR operating experience.

As a result of the GE review, modifications have been made to the GESSAR II design to lessen the burden on the operator during abnormal transients. For example, the automatic depressurization system (ADS) logic has been modified to eliminate the need for manual actuation during events that do not cause high containment pressure. A time delay was also added to the ADS inhibit logic to give the operator more time to stabilize water level during anticipated transient without scram (ATWS) events. These improvements/modifications have lessened the effect of operator actions on public risk.

Failure of the operator to depressurize the primary system for transients with failure of high-pressure systems were dominant core-damage sequences in other BWR risk studies; however the ADS modifications have lessened the impact of these sequences on core damage for GESSAR II.

Since ATWS events have also been dominant contributors to BWR core-damage frequency, any improvements to ATWS mitigation systems, such as lessening the burden on the operator during these events, will lessen the effect of operator action on risk. Implementation of Alternate 3A modifications on the GESSAR II design have reduced the reliance on operator action and thus risk from ATWS events. The GESSAR II PRA results, which include systematic consideration of operator actions, show low public risk. Even recognizing that uncertainties associated with the probability of human error may be quite large, substantially higher error rates would have to apply for the risk to be significant. Therefore this issue is resolved for GESSAR II.

GSI B-26: Structural Integrity of Containment Penetrations

This issue involves staff evaluations to assess the adequacy of specific containment penetration designs of high-energy-fluid systems from the point of view of structural integrity, inservice inspection (ISI) requirements, and new surveillance or analysis methods applicable to containment penetrations which are identified as inaccessible. For generic resolution, the staff should determine whether or not the configuration and assessability of the welds in the proposed design and the procedures proposed for performing the examination permit the inservice examination requirements of Section XI of the ASME Code at an augmented frequency in break exclusion regions, as required by SRP Section 3.6.2. Upon satisfactory resolution of inspectability concerns, this issue should not affect public risk.

For those plants that meet the SRP Section 3.6.2 requirements there should be no significant contribution to core-damage frequency for this issue. Since GESSAR II has committed to meet Section XI of the ASME Code and SRP Section 3.6.2, this issue is resolved.

GSI B-55: Improved Reliability of Target Rock Safety-Relief Valves

Safety concerns regarding the pressure-relief system of BWRs have emerged based on the operating experience gained with and the associated failures of Target Rock valves. The concerns include failure of the SRV to open on demand, and failure to reclose after demand or spurious opening.

A significant number of failures have occurred on Target Rock valves for various reasons. Studies and testing of these valves have resulted in design changes in the valves and the issuance of several generic installation, operating, and maintenance instructions.

The GESSAR II design utilizes the direct-acting spring-loaded-type SRV, not the pilot-operated Target Rock type; thus, minimizing the concern for GESSAR II. The staff, as well as GE, recognizes the importance of SRV performance and the limited operational data for the type of valve utilized in the GESSAR II design. Therefore, GE will require utility applicants referencing GESSAR II to participate in an SRV surveillance program developed by the BWR Licensing Review Group. This surveillance program specifies more detailed information than required for Licensee Event Reports on the Nuclear Plant Data Reliability System.

By not using the Target Rock valves on GESSAR II and the commitment to have an applicant referencing GESSAR II participate in a comprehensive SRV surveillance program, this issue is resolved for GESSAR II.

GSI B-56: Diesel Reliability

Events (offsite and onsite) that result in loss of offsite power necessitate reliance on the onsite emergency diesel generators for successful accident mitigation. Improvement in starting reliability of onsite emergency diesel generators will reduce the probability of events that could escalate into a core-damage accident and, thus, could effect an overall reduction in public risk.

This issue is closely related to USI A-44, Station Blackout and much of the significance to the GESSAR II design has been discussed with the LOOP sequences that dominate the core-damage frequency for GESSAR II. The contribution to core-damage frequency is dominated by common-mode failure of the three diesel generators and it exceeds the contribution resulting from the random failure frequencies of individual diesel generators. The GESSAR II PRA included an assessment of the reliability of the three diesel generators which supply emergency onsite power. The assessed individual diesel generator failure to start and keep running is 2×10^{-2}. The common-mode diesel generator failure probability for three diesel generators is estimated to be about 4×10^{-4}. Resolution of this issue will ensure some basic reliability goals for each diesel generator. An applicant referencing the GESSAR II design will be required to meet the reliability criteria. However, improvement in individual diesel generator reliability may not necessarily improve the common-mode failure probability and, therefore, may not have an appreciable impact on core-damage frequency. As such, the risk reduction achievable may be very little or none. Design improvements for GESSAR II such as systems that need no ac or dc power to provide reactor coolant system inventory and provide containment heat removal are being considered for GESSAR II to further reduce public risk. See Section 15.6.3 of this SER for the staff evaluation of potential design improvements to the GESSAR II design that are both cost beneficial and reduce public risk. The staff considers B-56 resolved provided that utility applicants referencing GESSAR II meet the applicable reliability criteria for diesel generators.

GSI B-58: Passive Mechanical Failures

Safety-related systems contain many valves; therefore, passive failures present a potentially significant safety concern because the effects on safety-related systems can be widespread. GSI B-58 is concerned with passive mechanical valve failures; GSI C-11 is concerned with active pump and valve failures. Active failures typically occur during valve operation; passive failures occur over a period of time, going unnoticed as the valve is rendered inoperable with the failure occurring after valve operation is demanded.

The staff is presently reviewing the GESSAR II design with regard to GSI B-58 and will address this in a future supplement to the SER.

GSI B-61: Allowable ECCS Equipment Outage Periods

This issue concerns establishing surveillance-test intervals and allowable equipment-outage periods, using analytically based criteria and methods for the technical specifications. The present technical specification allowable equipment outage times were determined primarily on the basis of engineering judgment. Optimization of the allowed outage period and the test and maintenance interval can reduce equipment unavailability and in turn reduce public risk. This optimization balances the decreased maintenance unavailability of equipment with reduced outage time with the costs of potential shutdowns due to exceeding limiting conditions of operation. The surveillance test interval affects the system reliability by possibly decreasing the failure probability of the equipment as the test interval is decreased, but at the expense of increased test unavailability of the equipment if the equipment cannot respond to a demand while under test. The overall safety significance will depend on the combination of all the changes to the technical specifications which are based on analysis rather than engineering judgment.

The GESSAR II PRA considers equipment unavailability due to maintenance probabilities based on outage periods allowed by the Standard Technical Specifications. On the basis of this assumption, the risk from this issue is addressed by the results of the PRA. Since it is concluded that the PRA shows there is low risk with regard to outage periods for the GESSAR II design, the staff considers this issue resolved for GESSAR II.

GSI C-8: Main Steamline Leakage Control Systems

The staff performed offsite dose estimates which indicate that operation of the main steam isolation valve leakage control system (MSIVLCS) required for many BWRs may result in higher offsite accident doses than if the system is not used. For the proposed generic resolution of GSI C-8 the dose calculations will assume non-operation of the MSIVLCS and will take credit for cold trapping of iodine and volatile gases. Leakage paths, other than through the MSIVLCS, will be considered. These alternate paths will include such components as the main steam piping, condenser, and the off-gas system. Leakage from these components would be small because normal operation requires that leakages be maintained at a low level; however, integrity of these systems is not assured during earthquakes since they are not designed for SSE. However, the probability of a design-basis LOCA and an earthquake is small. The MSIVLCS collects leakage past the valves and discharges it into a compartment serviced by the standby gas treatment system. Holdup time or cold trapping is not considered in the classical analysis of a design-basis LOCA with the MSIVLCS. Therefore, the calculated doses are expected to be greater through the MSIVLCS than through the steam system, unless the steam system integrity is lost. The resolution of this issue will include investigation of alternate means to handle leakage past the MSIVs and the desirability of the MSIVLCS. Little staff effort was originally devoted to resolve this issue because the issue had low priority. However, new concerns have arisen because operational experience has shown a relatively high failure rate for the MSIVLCS and because of recent data (NRC memorandum, July 1, 1982) on the magnitude and frequency of MSIV leakage at BWRs in excess of the technical specification limit (typically 11.5 scfh) by 2 orders of magnitude have renewed concerns for the viability of the MSIVLCS design. Excessive MSIV leakage, may exceed the design capacity of the MSIVLCS and render the MSIVLCS ineffective.

The BWR Owners Group has reviewed the current MSIV testing and maintenance activities and has made recommendations about testing and repairing MSIVs to improve their sealing capability. At least one plant that has implemented these recommendations has met the technical specification limit for MSIV leakage after one cycle of operation without refurbishment (7 out of 8 valves, the 8th valve was 14 scfh). The staff will be following other efforts at other plants to verify that the BWR Owners Group recommendations will consistently reduce the MSIV leakage rates.

With regard to the GESSAR II design (a positive-pressure LCS), leakage will occur past all four valves (2 MSIVs, leakage control valve, and turbine stop valve) into the condenser and to the turbine building through the gland seals, assuming the positive-pressure LCS is not operating. The large surface area available for fission-product plateout, cold trapping of iodine and small leakage pathways should provide fission-product retention. For the first 20 minutes following a LOCA, the LCS will not be operating, and in this regard, GESSAR II will have the same effect as a BWR with a negative-pressure LCS.

The suppression-pool bypass study documented in the GESSAR II Source Term Sensitivity Report evaluated the probability and consequence of fission-product release through MSIVs. The results show that the potential fission-product release through the main steam lines is negligible compared to fission products which had received pool scrubbing and were released after containment failure. Therefore, this leakage pathway has been evaluated and shown to be a negligible risk contributor in the GESSAR II design. Staff evaluation also showed that only sequences with leakage rates of greater than 100 scfh were dominant contributors to offsite consequences. With the BWR/6 revised test and maintenance procedures, the resulting leakage rates should be in the order of 20 scfh, before considering the cascading effect of four valves in series. Therefore, the staff considers this issue resolved for GESSAR II providing the MSIV leakage rates are within those specified by the Technical Specifications.

GSI C-11: Assessment of Failures and Reliability of Pumps and Valves

Operating experience at nuclear power plants indicates that a number of valves, valve operators, and pumps fail to operate as required in the technical specifications either under testing conditions or when they are demanded to operate. The unreliability of active valves and pumps in nuclear power plant safety systems contributes to the risk associated with postulated core-damage-accident sequences.

The failures of active pumps and valves leading to core damage have been evaluated in the GESSAR II PRA. The core-damage frequency is dominated by station blackout with eventual failure of the RCIC system. This sequence contributes about 80% to the core-damage frequency. Therefore, improved reliability of pumps and valves will not substantially decrease risk unless the improvement decreases the frequency of $CT1-P_a$ (LOOP for < 60 min.), $CT1-P_b$ (LOOP for > 60 min) accident sequences. As shown in the GESSAR II PRA, the CT1-P sequences are dominated by the common-mode failure of the three diesel generators.

The reliability of RCIC is about 90%, it has a steam-driven turbine pump and all control valves are dc powered. If the RCIC were assumed to be perfectly reliable, GE estimates the risk reduction to be a factor of 1.2. Therefore, any improvement in the reliability of active valves and pumps over the values assumed in the GESSAR II PRA would not achieve any significant reduction in core-melt frequency.

Improvements in the SRVs and in the testing and maintenance of MSIVs for GESSAR II are expected to show a decrease in the significance of these valves to risk, but further increases in the reliability of the remaining pumps and valves will have a negligible impact on GESSAR II risk. This issue is resolved for the GESSAR II design.

GSI 12: BWR Jet Pump Integrity

A jet pump failure, caused by progressive stress-corrosion cracking of the pump's hold-down beam, drew attention to the issue of jet pump integrity. Failure of a jet pump is a concern during normal operation because of the jet pump's contribution to proper water flow distribution; and during LOCA conditions for maintaining a water level (reflooding) following a recirculation line break. A damaged jet pump could also permit increased coolant loss in a LOCA since the jet pump nozzle area is the limiting break flow area.

The issue of jet pump integrity associated with progressive stress-corrosion cracking of the pump's hold-down beam should be solved by changing the hold-down beam material and by improving the heat treatment process. These improved hold-down beams, which are to be used in the GESSAR II design, were installed in a foreign BWR and have been operating for about 2 years with no indication of stress-corrosion cracking. The staff considers this issue resolved for the GESSAR II design with the requirement that applicants referencing the GESSAR II design get an early indication of possible hold-down beam damage by monitoring the rates of jet-pump-driven flow to driving flow during normal operation and by performing ultrasonic inspection of the beams for incipient cracking at refueling (at approximately 18-month intervals). The value of the ultrasonic inspection at refueling is based on the slow, progressive nature of stress-corrosion cracking (GE estimates it takes 1½ years for cracks to propagate to failure).

GSI 23: Reactor Coolant Pump Seal Failures

This issue deals with the high rate of reactor coolant pump (RCP) seal failures that challenge the makeup capacity of the ECCS in PWRs. RCP failures occur at approximately the same frequency in BWRs. However, operating experience shows that the leakage rate for RCP seal failures in BWRs is less. The smaller leak rate, larger high-pressure ECCS and feedwater makeup capabilities, and isolation valves on the RCP loops lessen the potential problem in BWRs. Therefore, the safety significance in BWRs is minimal. In response to the requirement of TMI Action Plan Item II.K.3.25, the staff concluded that no modifications to the seal cooling for BWR recirculation pumps is required (NRC, December 1, 1982).

Since the same recirculation pumps currently used in BWRs will be in the GESSAR II design, this issue is resolved.

GSI 29: Bolting Degradation or Failure in Nuclear Power Plants

In recent years, the number of bolting-related incidents reported by licensees of operating reactors and reactors under construction has increased. A large number of these reported incidents are related to primary pressure-boundary applications and major component support structures. Therefore, there is an increasing concern regarding the integrity of the reactor coolant pressure boundary and the reliability of component-support structures following a LOCA or earthquake.

The concern is compounded by the fact that there is currently no reliable nondestructive examination (NDE) method to detect cracking or degradation of such bolts or studs from the failure modes of stress corrosion, fatigue, erosion corrosion, and boric acid corrosion. Visual inspection is currently the only reliable method to discover degradation by boric acid corrosion or erosion corrosion, which requires disassembly of the component for inspection. Under present inservice inspection requirements, visual inspection of bolts is not mandatory, nor is ultrasonic inspection on less than 2-inch-diameter pressure-retaining bolts. A major accident such as a LOCA could occur because of undetected extensive bolting failure of the primary pressure boundary.

There have been 44 bolting incidents reported in PWRs (NUREG-0943, January 1983); 2 failures have been reported in BWRs. The principal modes of bolting failure were classified as stress corrosion, fatigue, boric acid corrosion, and erosion corrosion. Nineteen of the incidents resulted from stress corrosion, the most common type of bolting failure (one in a BWR).

Twelve failures were identified as boric acid corrosion (second most common type of failure). The remaining 13 failures were either fatigue, erosion corrosion, or other types of failure (one fatigue failure in a BWR).

The bolting specified in the GESSAR II design is not subject to boric acid corrosion, and by not using high-strength bolts the GESSAR II design guards against the stress-corrosion failure mechanism.

Since there have been only 2 reported bolting failure incidents occurring in BWRs, and two of the most common types of failure modes are unlikely by virtue of the GESSAR II design, this issue is resolved for GESSAR II.

GSI 40: Safety Concerns Associated With Pipe Breaks in the BWR Scram System

This issue concerns failure of the scram discharge volume (SDV) piping or associated piping which make up the BWR scram system. A rupture of the SDV piping could result in an un-isolatable break outside the primary containment, which may threaten ECCS equipment by flooding or causing environmental conditions for which the ECCS equipment is not qualified. The GESSAR II design locates the SDV piping and associated piping within the primary containment; therefore, any leakage would return to the suppression pool and not flood the ECCS equipment which is located outside the primary containment. The GESSAR II design also has containment sprays which make it possible to mitigate the effects of scram systems breaks.

The staff has provided guidance (NUREG-0803) to ensure pipe integrity, detection capability and mitigation capability, and qualification of emergency equipment to the expected environment, GE has committed to the guidance of NUREG-0803 for GESSAR II; therefore, the staff considers this issue resolved.

GSI 41: BWR Scram Discharge Volume Systems

This issue concerns deficiencies in the BWR scram discharge systems that were highlighted at Browns Ferry Unit 3 in 1980 when about 40% of the control rods failed to fully insert during a normal shutdown. The significance of the issue lies in the potential for failure to scram.

The GESSAR II design incorporates modifications required by the December 9, 1980, NRC letter to all BWR licensees, as well as other modifications to address ATWS events.

With the staff recommendations required for resolution of this issue incorporated in the GESSAR II design and the risk from ATWS events reduced by the modifications discussed in Section 15.6.2 of this supplement, this issue is resolved.

GSI 50: Reactor Vessel Water-Level Instrumentation in BWRs

BWRs use water-level instrumentation to perform a number of functions including control functions such as feedwater control, and protective functions such as automatic scram and automatic initiation of emergency core cooling systems. This issue considers that there could be a potential adverse control systems interaction with protection systems. For example, interactions may lead to a loss of reactor water level caused by automatic termination of normal feedwater (control) with failure to automatically start the emergency water source (protection). This scenario may occur with a break in the water level instrument reference leg in one of the two channels of water level instrumentation and a failure in the other channel of water level instrumentation.

In response to RG 1.97 requirements, the GESSAR II design has an enhanced water level instrumentation system. This design includes indication of instrument line breaks or leaking equalizer valves so that the operator is alerted when one channel of the water level instrumentation is not functioning correctly. Normally, administrative procedures require operators to switch the level control to the correctly functioning channel to avoid adverse control functions. In addition, the GESSAR II design contains ECCS actuated from one of two divisions in such a way as to ensure ECCS actuation in the event of an instrument-line break together with a single failure in the remaining channels. For the BWR/6, the vertical drop in the drywell has been minimized to ensure that temperature and flashing concerns are alleviated.

For these reasons, the staff considers GSI-50 resolved for GESSAR II.

GSI 51: Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems

The service water system (SWS) is the ultimate heat sink that, during an accident or transient, cools the intermediate cooling loops that in turn cool safety-related equipment and area-cooling coils. Experience has shown a number of incidents where fouling (from mud, silt, corrosion products, or aquatic bivalves) of the safety-related SWS has led to plant shutdowns, reduced power operation for repairs and modifications, and degraded modes of operation.

A possible solution to this issue is improvements in surveillance and preventive maintenance programs at all sites, especially those where fouling elements such as aquatic bivalves are known to exist. These programs should improve the SWS reliability.

Much of the essential service water system for the GESSAR II nuclear island is outside the scope of design. This issue will be addressed by utility applicants that reference the GESSAR II design.

GSI 61: SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments

This issue postulates a break in the SRV discharge line in the wetwell airspace above the suppression pool of Mark I and II plants. Coupled with the line break is the failure of the relief valve to close after its actuation in response to the transient. The relief valve must remain open for approximately 10 minutes for a significant amount of steam to escape, bypass the pool, and overpressurize the containment vessel. This postulated scenario would result in a direct release of coolant and effluents to the environment, should containment failure occur as a result of steam bypassing the pool. If core damage were to occur, large offsite releases of radioactivity would be experienced.

In the GESSAR II design the SRV discharge line is routed through the drywell wall and enters the wetwell below the water level. There is also a sleeve that surrounds the discharge line and terminates in the pool at the level of the first row of horizontal vents. The GESSAR II design eliminates this problem and the staff consider this issue resolved for GESSAR II.

GSI 65: Probability of Core Melt Due to Component Cooling Water System

This issue is concerned with failure of the component cooling water system which has the consequence of rendering ECCS pumps or containment cooling inoperable, causing a small LOCA (RCP seal failure), or otherwise affecting the ability of the plant to prevent a core-damage event. In the system fault trees, the GESSAR II PRA considered the support systems required for continued operation. The failure probabilities for room coolers and other cooling provided by the essential service water system were assessed for the evaluation. The failure of component cooling water to the recirculation pump seals has been discussed under Generic Safety Issue 23. These evaluations show that the significance of component cooling water failure is accounted for in the GESSAR II PRA.

The GESSAR II design uses self-cooling for the RCIC system pumps and essential service water for the other ECCS components. Component cooling with a closed cooling water system is only used for functions such as sample cooling, drywell cooling, and recirculation pump seals which do not pose a significant risk if they fail to function. The major portion of the essential service water system is outside the GESSAR II design and its potential to contribute to core damage is addressed by interface requirements proposed by GE in a letter dated June 1984. An applicant referencing the GESSAR II design will be committed to meeting these interface requirements during the licensing process. The actions required to satisfy interface risk assumption are discussed in Section 15.6.2 of this supplement.

GSI 77: Flooding of Safety Equipment Compartments by Back-Flow Through Floor Drains

In 1981, a licensee notified the staff that the watertight integrity of the service water pump rooms in both units could not be assured because check valves had not been installed in the floor drain system which drains by gravity to the turbine condenser pit in the turbine building. Without these check valves, the operability of the service water pumps for both units could not be assured in the event of a condenser circulating water conduit break in one unit. An evaluation (NRC memorandum, March 11, 1983) was performed on the generic implications of this concern and it was concluded that the matter of protection from backflow flooding through the drain system had not been addressed adequately. The safety significance of this issue does not apply to plants designed in accordance with the provisions of SRP Section 9.3.3, "Equipment and Floor Drainage Systems," and SRP Section 10.4.5, "Circulating Water System," since the criteria in these SRP sections adequately deal with this issue.

Safety-related components other than service water pumps may also be affected and flooding may come from sources other than circulating water conduits. The GESSAR II design was reviewed in accordance with the above SRP section criteria and found acceptable. The staff considered those systems needed to provide safe plant shutdown and the physical location of these systems with regard to potential in-plant flooding. Each train of ECCS equipment is located in individual watertight rooms that contain floor drain sumps to collect leakage. Each floor drain sump is equipped with safety-grade level instrumentation and each sump has two sump pumps to remove the water to the radwaste system. There are check valves located at the discharge of each pump to prevent backflow from the radwaste drain collecting system. Therefore, flooding between individual ECCS rooms is prevented since each room has its own sump, and flooding from areas other than the ECCS room via the drain system is prevented by check valves in the sump pump discharge lines.

Because the GESSAR II design meets the requirements of SRP Section 9.3.3 and the applicants referencing GESSAR II will have to meet the criteria of SRP Section 10.4.5 for balance-of-plant design, this issue is resolved for GESSAR II.

GSI 82: Beyond-Design-Basis Accident in Spent Fuel Pool

The risks associated with beyond-design-basis accidents in the spent fuel storage pool were examined in the Reactor Safety Study (NUREG-75/014) and were considered to be orders of magnitude below those involving the reactor core. The reason for this item is the simplicity of the pool; i.e., coolant is at low pressure, spent fuel is subcritical, heat source is low, no anticipated transients could intercept cooling or cause criticality. The reasons for re-examination of this issue are twofold. First, more spent fuel is being stored instead of being reprocessed, thus adding a larger inventory of fission products in the pool, increasing the heat load on the pool cooling system, and decreasing the distance between fuel assemblies. Second, some laboratory studies (NRC memorandum, August 10, 1983; NUREG/CR-0649) have provided evidence of the possibility of fire propagation between assemblies in an air-cooled environment. These two reasons together provide the basis for an accident scenario not previously considered.

A typical spent fuel pool with high density storage racks can hold about five times the fuel in the core; however, since typical reloads discharge one-third the core, much of the spent fuel in the pool will have considerable decay time (this reduces the radioactive inventory). After about 3 years of storage, most of the spent fuel stored in the pool may be air coolable (i.e., need not be submerged to prevent melting even though submersion may be desirable for shielding and reduction of airborne activity). If the pool were drained, the last two discharged fuel loads would still be "fresh" enough to melt under decay heat. The Zircaloy cladding of this fuel could be ignited during the heatup with the resulting fire spreading to most of the fuel in the pool. The heat of combustion in combination with the decay heat would probably drive "borderline aged" fuel to melt. The local decay-heat-generation rates necessary for ignition are now under consideration in Generic Safety Issue 82. Melting and/or production of airborne particulates by combustion could cause a release of fission products from the spent fuel pool to the environment, since most spent fuel pools are located outside the primary containment. This direct release may be more likely than for comparable accidents involving the reactor core. The safety significance and medium-priority status of this issue are based on a seismic event capable of draining the pool concurrent with the conditional failure probability of loss of pool makeup (approximate accident frequency 2×10^{-6}/year).

The analysis is based on the Reactor Safety Study assumption of a fuel pool at a 10-story elevation above grade which may not be applicable to GESSAR II because the GESSAR II fuel pool is below grade in the seismic Category I fuel building and sits on the basemat. The seismic analysis for this design is being reviewed and the seismic capacity of the fuel building is being evaluated.

The staff is presently reviewing the GESSAR II design as it relates to GSI-82 and will address its resolution in a future supplement to the SER.

APPENDIX E

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APPENDIX G

COMPLIANCE WITH CP/ML RULE (10 CFR 50.34(f))

Item (1)(i)

Perform a plant/site-specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant (II.B.8).

Discussion

The PRA was submitted as part of GESSAR II in March 1982. The staff's evaluation of the PRA is discussed in Section 15.6 of this supplement.

Item (1)(ii)

Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (applicable to PWRs only) (II.E.1.1):

(a) A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques.

(b) A design review of AFWS.

(c) An evaluation of AFWS flow design bases and criteria.

Discussion

This requirement is not applicable to BWRs.

Item (1)(iii)

Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage (II.K.2.16 and II.K.3.25).

Discussion

See page 15-7 of the SER.

Item (1)(iv)

Perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCAs from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened (applicable to PWRs only) (II.K.3.2).

Discussion

This requirement is not applicable to BWRs.

Item (1)(v)

Perform an evaluation of the safety effectiveness of providing for separation of high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level (applicable to BWRs only) (II.K.3.13). (For plants with high-pressure core spray systems in lieu of high-pressure coolant injection systems, substitute the words high-pressure core spray for high-pressure coolant injection and HPCS for HPCI.)

Discussion

See page 5-22 of the SER.

Item (1)(vi)

Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems (applicable to BWRs only) (II.K.3.16).

Discussion

See page 5-7 of the SER.

Item (1)(vii)

Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling (applicable to BWRs only) (II.K.3.18).

Discussion

See page 6-41 of the SER.

Item (1)(viii)

Perform a study of the effect on all core-cooling modes under accident conditions of designing the core-spray and low-pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present (applicable to BWRs only) (II.K.3.21).

Discussion

See page 7-30 of the SER.

Item (1)(ix)

Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least 2 hours (applicable to BWRs only) (II.K.3.24). (For plants with high-pressure core spray systems in lieu of high-pressure coolant injection systems, substitute the words high-pressure core spray for high-pressure coolant injection and HPCS for HPCI.

Discussion

The GESSAR II design does not have HPIC, however, it does have HPCS. This matter is discussed in SER Section 9.4.3.

Item (1)(x)

Perform a study to ensure that the automatic depressurization system, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for nonsafety-related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves (applicable to BWRs only) (II.K.3.28).

Discussion

GE has defined and elaborated on the number of times the ADS valves are capable of cycling using only the accumulator inventory and the length of time that the accumulators are capable of performing their function following an accident. A backup system is also provided. During normal operations, the operators are responsible for ensuring that the ADS accumulators are fully charged. The accumulators are instrumented for this purpose. The ADS accumulator system is environmentally and seismically qualified and no credit has been taken for nonsafety-related equipment when establishing the short- and long-term capability of the ADS accumulator system.

A discussion regarding the allowable leakage and the margins incorporated into the criteria are required for the as-built system. The requirement for leak testing will be part of the technical specifications.

The staff concludes that the requirements of item (1)(x) are satisfied for the GESSAR II design.

Item (1)(xi)

Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown (applicable to BWRs only) (II.K.3.45).

Discussion

See SER Section 6.3.

Item (1)(xii)

Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of 10 CFR 50.34(f). As a minimum include consideration of a hydrogen ignition and postaccident inerting system. The evaluation shall include:

(a) A comparison of costs and benefits of the alternative systems considered.

(b) For the selected system, analyses and test data to verify compliance with the requirements of paragraph (f)(2)(ix) of 10 CFR 50.34.

(c) For the selected system, preliminary design descriptions of equipment, function, and layout.

Discussion

Item (1)(xii) is being evaluated concurrently with the other hydrogen-related item (item (2)(ix)), and will be reported on in a future supplement to the SER.

Item (2)(i)

Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs (applicable to construction permit applicants only) (I.A.4.2).

Discussion

Item (2)(i) is outside the scope of GESSAR II and will be provided by utility applicants that reference GESSAR II.

Item (2)(ii)

Establish a program, to begin during construction and continue into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO (Institute of Nuclear Power Operations) and other industry efforts (applicable to construction permit applicants only) (I.C.9).

Discussion

Item (2)(ii) is outside the scope of GESSAR II and will be provided by utility applicants that reference GESSAR II.

Item (2)(iii)

Provide, for Commission review, a control room design that reflects state-of-the-art human factors principles before committing to fabrication or revision of fabricated control room panels and layouts (I.D.1).

Discussion

See SER Section 18.

Item (2)(iv)

Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded (I.D.2).

Discussion

The staff is currently in the process of reviewing the GE safety parameter display system for GESSAR II, GE's Emergency Response Information System (ERIS). The results of the staff's evaluation will be discussed in a future supplement to the SER.

Item (2)(v)

Provide for automatic indication of the bypassed and operable status of safety systems (I.D.3).

Discussion

SER Section 7.2.2.9, Bypassed and Inoperable Status Indication, discusses, in part, the staff's evaluation of item (2)(v). However, it will be the responsibility of a utility applicant referencing GESSAR II to demonstrate that the system monitoring function uses actual status information of monitored components and not demand signal information.

Item (2)(vi)

Provide the capability of high point venting of noncondensible gases from the reactor coolant system and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity (II.B.1).

Discussion

See SER Section 5.2.3.

Item (2)(vii)

Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain Technical Information Document (TID) 14844 source-term radioactive materials, and design as necessary to permit adequate access to important areas to protect safety equipment from the radiation environment (II.B.2).

Discussion

See SER Section 12.3.

Item (2)(viii)

Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844 source-term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations (II.B.3).

Discussion

See SER Section 9.3.2.2.

Item (2)(ix)

Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of 100% fuel-clad metal-water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (1)(xii) of 10 CFR 50.34(f) is sufficient at the construction permit stage. The hydrogen-control system and associated systems shall provide, with reasonable assurance, that (II.B.8):

(a) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel-clad metal-water reaction, or that the postaccident atmosphere will not support hydrogen combustion.

(b) Combustible concentrations of hydrogen will not collect in areas where unintended combustion of detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(c) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen-control system.

(d) If the method chosen for hydrogen control is a postaccident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

Discussion

By letter dated August 20, 1984, GE submitted a draft amendment to GESSAR II Sections 1G.12 and 1G.21. This draft amendment requires utility applicants referencing GESSAR II to provide an igniter hydrogen control system capable of handling hydrogen as required by the Proposed Interim Requirements Related to Hydrogen Control (46 FR 62281). The hydrogen control system will be based on the NRC staff· roved results of the Hydrogen Control Owners Group's tests and analyses. GE has also provided UPPS which is designed to reduce the overall risk of core damage and, therefore, the overall probability that hydrogen will be generated.

The staff is reviewing GE's commitment on hydrogen control and UPPS and will provide its evaluation in a future supplement to the SER.

Item (2)(x)

Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWRs, PORV block valves, for all fluid conditions expected under operating conditions, transients, and accidents. Consideration of anticipated transient without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed (II.D.1).

Discussion

See SER Sections 3.9.3 and 5.2.3.

Item (2)(xi)

Provide direct indication of relief and safety valve position (open or closed) in the control room (II.D.3).

Discussion

See SER Section 7.3.2.3.

Item (2)(xii)

Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide AFW system flow indication in the control room (applicable to PWRs only) (II.E.1.2).

Discussion

This requirement is not applicable to BWRs.

Item (2)(xiii)

Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available (applicable to PWRs only) (II.E.3.1).

Discussion

This requirement is not applicable to BWRs. It applies only to PWR-type reactors.

Item (2)(xiv)

Provide containment isolation systems that (II.E.4.2):

(a) Ensure all nonessential systems are isolated automatically by the containment isolation system.

(b) For each nonessential penetration (except instrument lines) have two isolation barriers in series.

(c) Do not result in reopening of the containment isolation valves on resetting of the isolation signal.

(d) Utilize a containment setpoint pressure for initiating containment isolation as low as is compatible with normal operation.

(e) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

Discussion

See SER Sections 3.9, 3.10, and 6.2.7.

Item (2)(xv)

Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions (II.E.4.4).

Discussion

The present purge system design at GESSAR II provides for continuous purging for the containment during power operation at 5,000 cfm through a 9-inch line to reduce airborne radionuclide concentrations to a level that permits continuous access. The staff has reviewed the present design which incorporates the use of the 9-inch line and has found it acceptable as documented in SSER 1, Section 6.2.4.1.

Item (2)(xv) also requires the applicant to demonstrate "high assurance that the purge system will reliably isolate under accident conditions." The applicant has indicated that performance of prototype 6-inch purge isolation valves has been evaluated and, in GE's opinion, meets the requirements of BTP CSB 6-4 for isolation dependability under accident pressures. As an interface item, the future utility applicant should provide the staff with details of the performance evaluations for the specific 9-inch purge isolation valves for review by the staff.

Item (2)(xvi)

Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents) (applicable to B&W designs only) (II.E.5.1).

Discussion

This requirement is not applicable to BWRs.

Item (2)(xvii)

Provide instrumentation to measure, record, and readout in the control room: (a) containment pressure, (b) containment water level, (c) containment hydrogen concentration, (d) containment radiation intensity (high level), and (e) noble gas effluents at all potential, accident-release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident-release points, and for onsite capability to analyze and measure these samples (II.F.1).

Discussion

See SER Sections 7.1.4, 7.5.2, 11.5.2, and 12.3.

Item (2)(xviii)

Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWRs and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and BWRs (II.F.2).

Discussion

See SER Section 4.4.9.

Item (2)(xix)

Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage (II.F.3).

Discussion

See SER Section 7.5.2.2 and SSER 1 Section 7.5.2.2.

Item (2)(xx)

Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (a) Level indicators are powered from vital buses, (b) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety, and (c) electric power is provided from emergency power sources (applicable to PWRs only) (II.G.1).

Discussion

This requirement is not applicable to BWRs.

Item (2)(xxi)

Design auxiliary heat-removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable (applicable to BWRs only) (II.K.1.22).

Discussion

See SER Section 5.4.2.

Item (2)(xxii)

Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS (applicable to B&W-designed plants only) (II.K.2.9).

Discussion

This requirement is not applicable to BWRs.

Item (2)(xxiii)

Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip (applicable to B&W-designed plants only) (II.K.2.10).

Discussion

This requirement is not applicable to BWRs.

Item (2)(xxiv)

Provide the capability to record reactor vessel water level in one location on recorders that meet normal postaccident recording requirements (applicable to BWRs only (II.K.3.23).

Discussion

See SER Section 7.5.2.1.

Item (2)(xxv)

Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a nearsite Emergency Operations Facility (III.A.1.2).

Discussion

Item (2)(xxv) is outside the scope of GESSAR II; therefore, the response to this requirement will be provided by utility applicants that reference GESSAR II.

Item (2)(xxvi)

Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source-term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency (III.D.1.1).

Discussion

See SER Section 9.3.4.

Item (2)(xxvii)

Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions (III.D.3.3).

Discussion

Item (2)(xxvii) is outside the scope of GESSAR II and will be provided by utility applicants that reference GESSAR II.

Item (2)(xxviii)

Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID 14844 source-term release, and make necessary design provisions to preclude such problems (III.D.3.4).

Discussion

See SER Section 6.4.

Item (3)(i)

Provide administrative procedures for evaluating operating, design, and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant (I.C.5).

Discussion

Item (3)(i) is outside the scope of GESSAR II and will be provided by utility applicants that reference GESSAR II.

Item (3)(ii)

Ensure that the quality assurance (QA) list required by Criterion II, Appendix B, 10 CFR 50 includes all structures, systems, and components important to safety (I.F.1).

Discussion

The QA program described in GESSAR II and GE's response relative to compliance with Item (3)(ii) (GESSAR II Item 1.G.42) has been reversed by the staff and found acceptable for the items important to safety that are controlled under the GE QA program (meeting Criterion II, Appendix B, 10 CFR 50) and to be identified by the utility applicants referencing GESSAR II.

Item (3)(iii)

Establish a quality assurance (QA) program based on consideration of (I.F.2):

(a) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions.

(b) Performing quality assurance/quality control functioning at construction sites to the maximum feasible extent.

(c) Including QA personnel in the documented review of and concurrence in quality-related procedures associated with design, construction, and installation.

(d) Establishing criteria for determining QA programmatic requirements.

(e) Establishing qualification requirements for QA and QC personnel.

(f) Sizing the QA staff commensurate with its duties and responsibilities.

(g) Establishing procedures for maintenance of "as-built" documentation.

(h) Providing a QA role in design and analysis activities.

Discussion

The staff has reviewed GE's response to Item (3)(iii) discussed in GESSAR II, Appendix 1G, and concludes that GE has confirmed compliance, through its QA program described in NEDO-11209-04A and through the QA program to be submitted by utility applicants referencing GESSAR II, that Item (3)(iii) will be properly controlled and carried out.

Item (3)(iv)

Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot-diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system (II.B.8).

Discussion

GE has agreed, as previously mentioned, to provide a separate 9-inch line for continuous purging and to lock closed the 42-inch refueling purge penetration during operating modes other than reactor shutdown and refueling. GE has proposed to dedicate this 42-inch penetration as the equivalent 3-foot-diameter opening required by this item. This will allow for the future installation (if necessary) of a filtered venting system. The staff finds this acceptable.

Item (3)(v)

Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that (II.B.8):

(A) (1) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone; for concrete containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Factored and Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel-clad metal-water reaction accompanied by either hydrogen burning or the added pressure from postaccident inerting, assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above as appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(2) Subarticle NE-3220, Division 1, and Subarticle CC-3720, Division 2, of Section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraphs (f)(3)(v)(A)(1) and (f)(3)(v)(B)(1) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 1717 H St., NW., Washington, D.C.

(B) (1) Containment structure loadings produced by an inadvertent full actuation of a postaccident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design-basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Service Load Category.

(2) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.

Discussion

In response to Item (3)(v), GESSAR II states that all areas of the containment exceed 45-psig Service Level C Limits, except for the knuckle region of the 2:1 torispherical head. Preliminary analysis indicates that the knuckle region can also meet 45-psig Service Level C Limits by modifying the curvature of the head using a three-center design (no other modifications are necessary). GE will provide the supporting information as part of the containment structural analysis. The staff will report on this item in a future supplement to the SER.

Item (3)(vi)

For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere (II.E.4.1).

Discussion

This item is not applicable to GESSAR II since the recombiners are located inside the containment.

Item (3)(vii)

Provide a description of the management plan for design and construction activities, to include (II.J.3.1):

(a) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant.

(b) Technical resources directed by the applicant.

(c) Details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect-engineer and the nuclear steam supply vendor.

(d) Proposed procedures for handling the transition to operation.

(e) The degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort.

Discussion

Item (3)(vii) is outside the scope of GESSAR II and will be provided by utility applicants that reference GESSAR II.

Reference

U.S. Atomic Energy Commission, Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962.

APPENDIX H

CONFORMANCE WITH THE STANDARD REVIEW PLAN (SRP) RULE (10 CFR 50.34(g))

Draft NUREG-1070, "NRC Policy on Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulations," requires, in part, that applications with previously granted final design approvals (FDAs) must perform evaluations of their design in accordance with 10 CFR 50.34(g) before the design can be referenced in new construction permit (CP) applications.

The staff's review of GESSAR II documented in the SER and its supplements has been carried out in accordance with the applicable acceptance criteria identified in the Standard Review Plan (NUREG-0800). The staff has, in some cases, identified differences from the SRP; these differences are noted and discussed in the appropriate sections of the SER. In GESSAR II, Section 1.8, GE identified the areas where differences to the SRP acceptance criteria were found. The differences are summarized in Table H.1. The staff has reviewed these differences from the SRP acceptance criteria and concludes that they provide an acceptable method of complying with the Commission's regulations. See Section 7.5 of SER Supplement 1 for a discussion related to the exceptions taken to RG 1.97.

Utility applicants who reference GESSAR II will provide the appropriate documentation for deviations of those design features outside the scope of GESSAR II, in accordance with 10 CFR 50.34(g). These design features are identified in GESSAR II Tables 1.9-1 through 1.9-19 and Table 1.10 of NUREG-0979 and its supplements.

Table H.1 Summary of differences from SRP (NUREG-0800)
(Source: GESSAR II Table 1.8.0-0)

SRP Section	Specific SRP acceptance criteria	Summary description of difference	GESSAR II subsection where discussed
3.7.1 (Rev. 1)	II.1.b - Design time history and damping values criteria	For higher damping values, the response spectra from synthetic time history are not in agreement with the enveloping values of the criteria.	19.3.3.48
3.7.3 (Rev. 1)	II.2.b - Determination of number of OBE cycles	For equipment and components other than piping, 10 rather than 50 peak OBE stress cycles are used	3.7.3.2.2
4.2 (Rev. 2)	II.A.1.(b) - Sets limit on the number of strain fatigue cycles	NEDE-24011 sets a more conservative limit than that in the SRP	4.2.1
4.2 (Rev. 2)	II.A.1.(c) - Fretting wear of structural members should be stated	Wear limits are not stated	4.2.1
4.2 (Rev. 2)	II.A.1.(g) - States that "worst case hydraulic loads" may not exceed the hold-down capability of the fuel assembly	Design basis allows up to 0.52-inch "lift-off"	4.2.1
4.2 (Rev. 2)	II.A.2.(e) - Prohibits any fuel melting	Design basis allows fuel melting that is not "excessive"	4.2.1
4.2 (Rev. 2)	II.A.2.(g) - Specifies uniform strain (elastic & plastic) limit of 1%	Elastic strain not included in the 1% limit	4.2.1
4.2 (Rev. 2)	II.A.2.(i) - Limits applied stress to ≤ 90% of the irradiated yield stress	Topical report is under review	4.2.1

Table H.1 (Continued)

SRP Section	Specific SRP acceptance criteria	Summary description of difference	GESSAR II subsection where discussed
4.2 (Rev. 2)	II.A.3.(e) - Analytical procedures are prescribed	Topical report is under review	4.2.1
4.2 (Rev. 2)	II.B - Lists parameters to be included in fuel description	Fuel description does not include all parameters listed in SRP	4.2.1
4.2 (Rev. 2)	II.C.3.(a) - Lists models to be included in thermal calculations	Gadolinia fuel properties not appropriate in model	4.2.2
4.2 (Rev. 2)	II.C.3.(d) - Describes acceptance criteria for design evaluation	Topical report is under review	4.2.2
5.2.3 (Rev. 2)	II.3 b.(1)(a) - Welding procedure qualification	Minimum preheat and maximum interpass temperature not specified	5.2.3.3.2.1
5.2.3 (Rev. 2)	II.3 b.(3) - RG 1.71, "Welder Qualification for Areas of Limited Accessibility"	Alternate position employed	5.2.3.4.2.3
6.2.1.1.C (Rev. 5)	II.9 - Compliance with NUREG-0783	GESSAR II analysis takes credit for weir wall annulus water	19.3.6.10 (Comparison to Section 5.7.1 of NUREG-0783)
6.2.1.2 (Rev. 2)	II.B.1 - Humidity for shield wall annulus analysis	1% relative humidity used in analysis	19.3.6.14
6.3 (Rev. 1)	III.19 - Operator action following LOCA	GESSAR II requires operator action within 10 minutes for some events	19.3.5.56
6.7 (Rev. 2)	II.1 - MSIV leakage control meeting RG 1.96	Exception taken to Position C.9 of RG 1.96	1.8.96

Table H.1 (Continued)

SRP Section	Specific SRP acceptance criteria	Summary description of difference	GESSAR II subsection where discussed
7.1 (Rev. 2)	II - RG 1.75 (Table 7-1)	Alternates to portions of RG 1.75 are utilized	7.1.2.10.18
7.2 (Rev. 2)	II.1 and II.2 - IEEE-279 (1971) and GDC 2	Some RPS inputs come from devices mounted on non-seismically qualified equipment and/or are located in non-seismically qualified enclosures	Table 19.3.7.14-1(j)
7.3 (Rev. 2)	II - TMI Item II.K.3.21: Restart of Core Spray and Low-Pressure Coolant Injection Systems (Table 7-2)	Core spray and LPCI systems do not automatically restart after being on low water level if the initiation signal is still present	1A.63
7.3 (Rev. 2)	II - Paragraph 4.17 of IEEE-279 (1971)	HPCS, LPCS, LPCI, ADS, and the containment spray mode of RHR share common interlocks between the automatic and manual initiation modes	19.3.7.42
7.5 (Rev. 2)	II - RG 1.97 (Table 7-1)	Exception taken to some of the requirements*	Appendix 1D
8.3.2 (Rev. 2)	BTP PSB-1 Section 1(c)(3) - Second level of undervoltage protection for Class 1E equipment	GESSAR II design based on maximum fluctuation of ±5% on grid voltage	19.3.8.5
9.5.1 (Rev. 3)	II.2.a - Implementation of fire protection program in accordance with BTP CMEB 9.5-1	Lack of 3-hr-fire-rated dampers in ventilation system	9.5.1.1
12.1 (Rev. 2)	II.2 - Instructions to designers and engineers regarding ALARA	No specific instructions provided	12.1.2.2.1

*See SSER 1 for more discussion.

Table H.1 (Continued)

SRP Section	Specific SRP acceptance criteria	Summary description of difference	GESSAR II subsection where discussed
12.2 (Rev. 2)	II.6 - Contained source descriptions	Size and shape of vessels with contained sources not provided	12.2.1.1
12.2 (Rev. 2)	II.6 - Buildup of activated containment sources	Buildup of activated corrosion products provided only for recirculation piping	12.2.1.2.7.2
15.3.3-15.3.4 (Rev. 2)	II.8 - Use of nonsafety-grade equipment	Credit is taken for nonsafety-grade equipment and failure of nonsafety-grade equipment is not assumed	15.3.3.2.2
15.3.3-15.3.4 (Rev. 2)	II.10 - Coincident loss of offsite power	Not analyzed with coincident loss of offsite power	15.3.3.2.2
15.4.4-15.4.5 (Rev. 2)	II.2.(b) - Fuel cladding integrity	MCPR not calculated	15.4.4.3.2, 15.4.5.3.2.1 & 15.4.5.3.2.2
15.6.5 Appendix B (Rev. 1)	II.(2) - Distribution of iodine inventory	Radiological analysis for LOCA assumes 25% of iodine is in suppression pool	Part 1b to 19.3.5.1