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JNF 038-83

June 7, 1983

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

Attention: Mr. D.G. Eisenhut
Division of Licensing

Gentlemen:

SUBJECT: IN THE MATTER OF 238 NUCLEAR ISLAND
GENERAL ELECTRIC STANDARD SAFETY ANALYSIS REPORT
(GESSAR II) DOCKET NO. STN 50-447

DRAFT AMENDMENT PERTAINING TO CP/ML AND SRP RULES

Attached please find a draft of GESSAR II Amendment Number 17 pertaining to our response to the CP/ML Rule (10CFR50.34(f)) and SRP Rule (10CFR50.34(g)). Our responses to these rules are included separately in this transmittal as Attachment Number 1 (CP/ML Rule) and Attachment Number 2 (SRP Rule).

We plan to formally file Amendment Number 17 in mid June 1983.

If there are any questions on the information provided herein, please contact J.N. Fox of my staff at (408) 925-5039.

Sincerely,

Willa

Glenn G. Sherwood, Manager
Nuclear Safety & Licensing Operation

Attachments

cc: F.J. Miraglia (w/o attachments)
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ATTACHMENT NO. 1

DRAFT OF GESSAR II
AMENDMENT NO. 17
RESPONSE TO CP/ML RULE

APPENDIX 1G

RESPONSE TO CP/ML RULE
10CFR50.34(F)

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APPENDIX 1G

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APPENDIX 1G

RESPONSE TO CP/ML RULE
10CFR 50.34 (f)

1G.0 INTRODUCTION

On January 15, 1982 (47FR2286) the NRC amended 10CFR34 to include (f) Additional TMI-Related Requirements. These additional requirements were directed to each applicant for a light-water-reactor Construction Permit or Manufacturing License (CP/ML) whose application was pending as of February 16, 1982.

In its Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation (7590-01), the NRC proposed to extend its policy such that future CP applications or reactivations of CP applications previously docketed also comply with the CP/ML rule .

This appendix 1G reports GE's responses for the 238 Nuclear Island to the NRC positions taken regarding the "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License" as referenced in NUREG-0718, Rev 2. These responses have developed as the NRC positions have evolved and been clarified by the issuance of subsequent documentation by the NRC.

The responses demonstrate that the NRC requirements are satisfactorily fulfilled for the 238 Nuclear Island. For each item a summary of the NRC position is given and followed by a response. The response clarifies the issue as it pertains to the 238 Nuclear Island and/or provides a listing of applicable GESSAR II sections, relevant correspondence, or other necessary documentation that may be referenced for complete clarification of our postion. Where a particular requirement is not applicable to the 238 Nuclear Island, a statement to that effect is provided in the response.

1G.0 INTRODUCTION (Cont'd)

For NRC positions that affect equipment outside the scope of the Nuclear Island design or utility operations and procedures, the response indicates that the subject will be addressed by the applicant. Otherwise, this Appendix 1G is complete in that all of the "Additional TMI-Related Requirements" approved for implementation by the NRC as listed in 10CFR 50.34 (f) have been favorably addressed where they apply to the 238 Nuclear Island.

The bracketed item numbers at the end of each title correspond with the subsections in 10CFR 50.34 (f). Alphanumeric designations at the end of each "NRC Position" statement correspond to the related action plan items in NUREG-0718 and NUREG-0660. They are provided in 10CFR 50.34 (f) for information only.

Table 1G.0-1 is provided as a convenient cross-reference which consolidates pertinent information associated with each of the 47 requirements. This includes the 10CFR34 (f) subsection, the TMI Action Plan numbers, the GESSAR II Section number where each NRC position and response is given, the item title as given in NUREG-0718; and the GESSAR II reference detailing resolution.

TABLE ~~1~~ 1G.0-1

GESSAR II ADHERENCE TO CP RULE 10 CFR 50.34(f)

SECTION	ACTION PLAN	GESSAR II SECTION	ITEM	GESSAR II REFERENCE ~~RESOLUTION/STATUS~~
(1) (i)	II.B.8	1G.1	Probabilistic Risk Assessment	Appendix 15D
(ii)	II.E.1.1	1G.2	Auxilary Feedwater System Evaluation	(PWR Only)
(iii)	II.K.2.16 & II.K.3.25	1G.3	Effect of loss of alternating-Current on pump seals.	Section 1A.46 & 1A.66
(iv)	II.K.3.2	1G.4	Report on overall Safety Effect of PORV Isolation System	(PWR Only)
(v)	II.K.3.13	1G.5	Separation of HPCS and RCIC System Initiation Levels	Section 1A.58
(vi)	II.K.3.16	1G.6	Reduction on of Challenges and failures of safety relief valves-feasibility study & system modification	Section 1A.60
(vii)	II.K.3.18	1G.7	Modification of ADS logic-feasibility study & modification for increased diversity of some event sequences	Section 1A.62
(viii)	II.K.3.21	1G.8	Restart of core spray and LPCI systems on low level-design and modification	Section 1A.63
(ix)	II.K.3.24	1G.9	Confirm adequacy of space cooling study for HPCS and RCIC	Section 1A.65
(x)	II.K.3.28	1G.10	Verify qualification of accumulators on ADS valves	Section 1A.68
(xi)	II.K.3.45	1G.11	Evaluate depressurization with other than full ADS	Section 1A.72
(xii)	—	1G.12	Evaluation of alternative hydrogen control systems	~~In Process~~ 1G.12

~~TABLE 3 (continued)~~

SECTION	ACTION PLAN	GESSAR II SECTION	ITEM	GESSAR II REFERENCE ~~RESOLUTION/STATUS~~
(2) (i)	I.A.4.2	1G.13	Long-Term Training Upgrade	Section 18.2
(ii)	I.C.9	1G.14	Long-Term Program of Upgrading of procedures	Applicant Responsibility
(iii)	I.D.1	1G.15	Control Room Design Reviews	Section 18.4
(iv)	I.D.2	1G.16	Plant Safety Parameter Display Console	Appendix 18B
(v)	I.D.3	1G.17	Safety System Status Monitoring	Not Applicable (GESSAR II complies with R.G. 1.47)
(vi)	II.B.1	1G.18	Reactor Coolant System Vents	Section 1A.19
(vii)	II.B.2	1G.19	Plant shielding to provide access to vital areas & protect safety equipment for Post-Accident operation	Section 1A.20
(viii)	II.B.3	1G.20	Post-Accident sampling	Section 1A.21
(ix)	II.B.8	1G.21	Hydrogen Control System Preliminary Design	~~In Process~~ 1G.21
(x)	II.D.1	1G.22	Testing Requirements	Section 1A.23
(xi)	II.D.3	1G.23	Relief and Safety Valve Position Indication	Section 1A.24
(Xii)	II.E.1.2	1G.24	Auxiliary feedwater system automatic initiative and flow indicator	(PWR Only)
(xiii)	I.E.3.1	1G.25	Relability of pane supplies for natural circulation	(PWR Only)

~~TABLE 3 (continued)~~

SECTION	ACTION PLAN	GESSAR II SECTION	ITEM	GESSAR II REFERENCE ~~RESOLUTION/STATUS~~
(2) (xiv)	II.E.4.2	1G.26	Isolation Dependability	Section 1A.29
(xv)	II.E.4.4	1G.27	Purging	~~9" Purge Penetrations~~ 1G.27
(xvi)	II.E.5.1	1G.28	Design Evaluator	B & W Only
(xvii)	II.F.1	1G.29	Additional Accident Monitoring Instrumentation	Appendix 1D
(xviii)	II.F.2	1G.30	Identification of and Recovery from Conditions leading to Inadequate Core Cooling	Section 1A.31
(xix]	II.F.3	1G.31	Instrumentation for Monitoring Accident Conditions(Reg. Guide 1.97)	Appendix 1D
(xx)	II.G.1	1G.32	Power supplies for pressurizer Relief Valves,Block Valves & Level Indication	(PWR Only)
(xxi)	II.K.1.22	1G.33	Describe Automatic & Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems when FW System not operable	Section 1A.38
(xxii)	II.K.2.9	1G.34	Analysis of Upgrading of Integrated Control System	(B & W Only)
(xxiii)	II.K.2.10	1G.35	Hand-wired Safety-grade Anticipatory Reactor trips	(B & W Only)
(xxiv)	II.K.3.23	1G.36	Central Water Level Recording	Section 1A.39
(xxv)	III.A.1.2	1G.37	Upgrade License Emergency Support Facility	Applicant Responsibility
(xxvi)	III.D.1.1	1G.38	Primary Coolant Sources Outside the Containment Structure	Section 1A.77
(xxvii)	III.D.3.3	1G.39	In-Plant Radiation Monitoring	Applicant Responsibility

~~TABLE 3 (continued)~~

SECTION	ACTION PLAN	GESSAR II SECTION	ITEM	GESSAR II REFERENCE ~~RESOLUTION/STATUS~~
(2) (xxviii)	III.D.3.4	1G.40	Control Room Habitility	Section 1A.79
(3) (i)	I.C.5	1G.41	Procedures for Feedback of Operating, Design and Construction Experience	Applicant Responsibility
(ii)	I.F.1	1G.42	Expand QA List	~~Subsection 19.3.17.4~~ 1G.42
(iii)	I.F.2	1G.43	Develop More Detailed QA Criteria	~~Applicant Responsibility~~ 1G.43
(iv)	II.B.8	1G.44	~~Provide one or more~~ Dedicated Containment Penetrations, Equivalent to a Single 3-foot diameter opening	~~Two 42" penetrations~~ 1G.44
(v)	II.B.8	1G.45	Containment Integrity	section ~~Head redesign to accomodate 45 psig at Service Level C.~~ 1G.45
(vi)	II.E.4.1	1G.46	Dedicated Penetration	Section 1A.28
(vii)	II.J.3.1	1G.47	Organization and Staffing to Oversee Design and Construction	~~Applicant Responsibility~~ 1G.47

1G.1 PROBABILISTIC RISK ASSESSMENT [Item (1)(i)]

NRC Position

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)

Response

A BWR/6 Mark III Probabilistic Risk Assessment (PRA) was submitted as Section 15D.3 to GESSAR II on March 19, 1982. The PRA is currently undergoing NRC staff review.

1G.2 AUXILIARY FEEDWATER SYSTEM (AFWS) EVALUATION [Item (1)(ii)]

NRC Position

Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (applicable to PWR's 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.

Response

This requirement is not applicable to the 238 Nuclear Island. It applies only to PWR-type reactors.

1G.3 IMPACT OF RCP SEAL DAMAGES FOLLOWING SMALL-BREAK ~~POTENTIAL REACTOR COOLANT PUMP SEAL DAMAGE [Item (1)(iii)]~~ LOCA with Loss of Offsite Power [Item (1)(iii)] all caps

NRC Position

Perform an evaluation of the potential for ~~an~~ d 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)

Response

Item II.K.2.16 is not applicable to the BWR. See Subsection 1A.46.

(II.K.3.25)
The consequences of a loss of (cooling water to the reactor recirculation pump seal coolers) has been analyzed with favorable results as indicated in Subsection 1A.66. The NRC approval of the test is found in NUREG 0979, page 15-7 (Section 15.2).

1G.4 REPORT ON OVERALL SAFETY EFFECT OF PORV ISOLATION SYSTEM ~~SMALL-BREAK LOCA CAUSED BY STUCK OPEN POWER RELIEF VALVE~~ [Item (1)(iv)]

NRC Position

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 LOCA's 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 PWR's only). (II.K.3.2)

Response

This requirement is not applicable to the 238 Nuclear Island. It applies only to PWR-type reactors.

1G.5 SEPARATION OF HPCS AND RCIC SYSTEM INITATION LEVELS ~~SAFETY EFFECTIVENESS IN SEPARATING HPCS/RCIC INITIATION LEVELS~~ [Item (1)(v)]

NRC Position

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. (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") (Applicable to BWR's only). (II.K.3.13)

Response

The evaluation was performed and is described in Subsection 1A.58. Separation of initiation levels was found to be unnecessary. However, the RCIC will be modified to allow automatic restart (following high level trip). NRC staff approval of these conclusions is documented in NUREG 0979, Page 5-22.

1G.6 REDUCTION OF CHALLENGES AND FAILURES ~~CHALLENGE REDUCTION OF RELIEF VALVES [Item (1)(vi)]~~ OF SAFETY RELIEF VALVES – FEASIBILITY STUDY AND SYSTEM MODIFICATION [Item (1)(vi)]

NRC Position

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 bWR's only). (II.K.3.16)

Response

A study was performed and is described in Subsection 1A.60. As a result of the study, a modification was proposed and will be implemented following NRC staff approval. The staff acknowledged the modification proposal in NUREG 0979, page 5-7.

1G.7 MODIFICATION OF ADS LOGIC-FEASIBILITY STUDY AND ~~OPTIMUM AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) DESIGN (Item (1)(viii))~~ MODIFICATION FOR INCREASED DIVERSITY OF SOME EVENT SEQUENCES [Item (1)(vii)]

NRC Position

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

Response

The study was performed by the BWR Owners' Group and is described in Subsection 1A.62. The favorable options and NRC concurrence is presented in NUREG 0979, page 6-41.

RESTART OF CORE SPRAY AND LPCI SYSTEMS
ON LOW LEVEL-DESIGN AND MODIFICATION

1G.8 ~~AUTOMATIC RESTART OF HPCS AND LPCI~~ [Item (1)(viii)]

NRC Position

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 BWR's only). (II.K.3.21)

Response

The study was performed by General Electric Company in conjunction with the BWR Owners' Group. As explained in Subsection 1A-63, the study concluded that no changes are necessary or appropriate. The NRC staff has approved GE's position in NUREG 0979, Section 7.3.2.4 (Page 7-30).

CONFIRM ADEQUACY OF SPACE COOLING STUDY FOR HPCS AND RCIC

1G.9 ~~SPACE COOLING FOR RCIC AND HPCS~~ [Item (1)(ix)]

NRC Position

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 two (2) hours. (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") (Applicable to BWR's only). (II.K.3.24)

Response

The completed study is described in Subsection 1A.65. Results indicated that the cooling systems for the HPCS and RCIC systems are adequate and no plant modifications are required.

VERIFY QUALIFICATION OF ACCUMULATORS (N ADS VALVES)

1G.10 ~~ADS STUDY WITH NO CREDIT FOR NON-SAFETY RELATED EQUIPMENT~~ [Item (1)(x)]

NRC Position

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 non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWR's only). (II.K.3.28)

Response

The study and its favorable results are described in Subsection 1A.68.

EVALUATE DEPRESSURIZATION WITH OTHER THAN FULL ADS

1G.11 ~~ALTERNATE DEPRESSURIZATION METHODS~~ [Item (1)(xi)]

NRC Position

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 BWR's only) (II.K.3.45)

Response

The completed evaluation is described in Subsection 1A.72. It was concluded that no change in depressurization rate is required or appropriate. NRC concurrence is indicated in NUREG 0979, Page 6-42.

1G.12 EVALUATION OF ALTERNATIVE HYDROGEN CONTROL SYSTEMS [Item (1)(xii)]

NRC Position

Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of ~~this section~~ 10 CFR 50.34(f). As a minimum include consideration of a hydrogen ignition and post-accident 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 (f)(2)(ix) of ~~this section.~~ 10 CFR 50.34(f)

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

Response

(A) Comparison of Costs and Benefits of the alternative systems considered will be provided by the applicant.

(B) Compliance with Requirement

The Applicant will provide the analyses and test data to verify compliance with the requirements of 10CFR50.34(f)(2)(ix

(C) Design Descriptions

The Applicant will provide the design descriptions of equipment, function, and layout.

LONG-TERM TRAINING UPGRADE

1G.13 ~~SIMULATOR CAPABILITY~~ [Item (2)(i)]

NRC Position

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

Response

The GE training simulator, near Tulsa, Oklahoma, was used as part of the control room human factors system/task analysis discussed in Section 18.2. This simulator was selected for use because it closely resembles the control room design for the 238 Nuclear Island except as noted in Chapter 18. Small-break LOCA's were among the transients selected for review. No limitations in the simulator's capability to simulate small breaks were noted.

It shall be the applicant's responsibility to establish a training program which addresses the concerns related to item I.A.4.2 of NUREG 0718.

1G.14 LONG-TERM PROGRAM ~~FOR IMPROVING~~ OF UPGRADING OF ~~PLANT~~ PROCEDURES [(Item (2)(ii)]

NRC Position

Establish a program, to begin during construction and follow 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 and other industry efforts. (Applicable to construction permit applicants only) (I.C.9)

Response

Plant procedures are the applicant's responsibility to provide. The Emergency Procedure Guidelines (EPGs) are established as part of the 238 Nuclear Island design as recommendations for operator action. The human factors system/task analysis (Sections 18.2 and 18.3) is based on actions consistent with the EPGs. A representative sample of emergency procedures for the 238 Nuclear Island is provided in Appendix 18A.

1G.15 CONTROL ROOM DESIGN REVIEWS [Item (2)(iii)] ~~STATE OF THE ART HUMAN FACTORS IN CONTROL ROOM [Item (2)(iii)]~~

NRC Position

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

Response

Chapter 18 provides a control room review, consistent with the Standard Review Plan (SRP), NUREG 0800, Section 18. It provides an EPG-based system/task analysis (Sections 18.2, 18.3) as well as a control room panel and arrangement review (Section 18.4) of selected panels. These were based on state-of-the-art human factors principles as established by General Electric and the BWR Owner's Group.

The Human Factors Engineering Branch (HFEB) of the NRC reviewed the text of the document and performed an audit of the finished control room modeled by the Black Fox Simulator near Tulsa, Oklahoma. The audit was conducted utilizing the human factors principles contained in NUREG 0700. The NRC review and results are contained in the GESSAR II Safety Evaluation Report (NUREG-0979). Also indicated in that report is GE's 5-point plan for resolution of human engineering discrepancies (HEDs) identified as a result of the control room audit. The plan was judged acceptable by the staff as indicated in NUREG 0979, Page 18-4 (Section 18.6).

1G.16 PLANT SAFETY PARAMETER~~S~~ DISPLAY SYSTEM (SPDS) [Item (2)(iv)]

NRC Position

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)

Response

The 238 Nuclear Island design includes an Emergency Response Information System (ERIS) which satisfies the requirements of NUREG 0718, Item I.D.2. The ERIS is described in Appendix 18B.

1G.17 SAFETY SYSTEM STATUS MONITORING [Item (2)(v)]

NRC Position

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

Response

The 238 Nuclear Island design fully complies with Regulatory Guide 1.47 (See Subsection 19.3.7.16). The automatic indication of bypassed and inoperable status of safety systems is therefore inherent in the design.

*Assumed "operable" incorrect; changed to "inoperable."

1G.18 REACTOR COOLANT SYSTEM VENTS ~~HIGH POINT VENTING~~ [Item (2)(vi)]

NRC Position

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)

Response

The vent provisions are part of the plant's original design and are covered by the original design bases. A complete description and comparison with requirements is given in Subsection 1A.19. NRC Staff approval is given NUREG 0979, Page 5-7.

PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS & PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION

1G.19 ~~RADIATION AND SHIELDING DESIGN REVIEW~~ [Item (2)(vii)]

NRC Position

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

Response

The details of the completed design review are described in Attachment A to Appendix 1A. As indicated in the attachment and in the response in Subsection 1A.20, no corrective action is required.

1G.20 POST-ACCIDENT SAMPLING ~~PROMPT ANALYSIS OF SAMPLES FROM REACTOR COOLANT SYSTEM AND CONTAINMENT~~ [Item (2)(viii)]

NRC Position

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 non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)

Response

A post-accident sample system has been added as indicated in the response of Subsection 1A.21. Technical descriptions are found in Attachment B to Appendix 1A and Subsection 19.3.9.17. NUREG 0979, Section 9.3.2.2 (Page 9-21) has approved those portions of the system within GE scope of supply. The remainder shall be supplied by the applicant.

1G.21 HYDROGEN CONTROL SYSTEM PRELIMINARY DESIGN [Item (2)(ix)]

NRC Position

Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 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) ~~this section~~ 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 post-accident atmosphere will not support hydrogen combustion.

(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or 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.

1G.21 HYDROGEN CONTROL SYSTEM PRELIMINARY DESIGN [Item (2)(ix)] (Continued)

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

Response

The ~~G/UNP~~ Applicant will provide a hydrogen control system capable of handling hydrogen generated by the equivalent of a 100% active fuel-clad metal water reaction. ~~The hydrogen control system will consist of igniters distributed throughout the drywell, the containment, and any local area which has the potential of pocketing hydrogen. These igniters will burn hydrogen as it is generated and will reasonably assure that uniformly-distributed hydrogen concentrations will not exceed 10% during and following an accident~~.

The following criteria will be used to design the hydrogen igniter system:

a. Burning of the hydrogen generated by the equivalent of a 100% active fuel-clad metal water reaction such that:

   (1) Uniformly-distributed hydrogen concentrations will not exceed 10% during and following the accident.

   (2) Local pocketing of hydrogen in the drywell, the containment, or local areas will be prevented.

b. The system will be single active failure proof.

c. Operation of the hydrogen ignition system will not adversely affect the safe shutdown of the plant.

d. The system will be protected from tornado and external missile hazards.

e. The system will not compromise the containment design.

The hydrogen control system shall provide with reasonable assurance that:

(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 post-accident atmosphere will not support hydrogen combustion.

(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or 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.

The following criteria will be used to design the hydrogen control system:

1. The system will be single active failure proof.

2. Operation of the hydrogen control system will not adversely affect the safe shutdown of the plant.

3. The system will be protected from tornado and external missile hazards.

4. The system will not compromise the containment design.

If the method chosen for hydrogen control is a post-accident inerting system, inadvertant actuation of the system must be safely accommodated during plant operation.

1G.22 TESTING REQUIREMENTS ~~TEST PROGRAM FOR QUALIFICATION OF SAFETY RELIEF VALVES (SRVs)~~ [Item (2)(x)]

NRC Position

Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, 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)

Response

A generic test program has been conducted through the BWR Owner's Group and is described in the response of Subsection 1A.23. A plant modification was recommended as a result of the program, and has been approved by the NRC staff (See NUREG 0979, Section 7.3.2.1). The deletion of the high drywell pressure interlock will therefore be implemented in the GESSAR II design prior to its reference by the first applicant.

Consideration of ATWS conditions shall be included in the test program following ATWS rulemaking as indicated in Section 15.8.

1G.23 ~~SRV~~ RELIEF AND SAFETY VALVE POSITION INDICATION [Item (2)(xi)]

NRC Position

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

Response

A system providing positive position indication is described in the response of Subsection 1A.24. Subsequent approval by the NRC staff was indicated in NUREG 0979, Section 7.3.2.3. Therefore, the design modification will be implimented prior to GESSAR II's reference by its first applicant.

1G.24 AUXILIARY FEEDWATER ~~(AFW)~~ SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION [Item (2)(xii)]

NRC-Position

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

Response

This requirement is not applicable to the 238 Nuclear Island. It applies only to PWR-type reactors.

1G.25 RELIABILITY OF POWER SUPPLIES FOR NATURAL CIRCULATION
~~NATURAL CIRCULATION IN HOT STANDBY CONDITIONS~~
[Item (2)(xiii)]

NRC Position

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 PWR's only) (II.E.3.1)

Response

This requirement is not applicable to the 238 Nuclear Island. It applies only to PWR-type reactors.

1G.26 ~~CONTAINMENT~~ ISOLATION DEPENDABILITY [Item (2)(xiv)]

NRC Position

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

(A) Ensure all non-essential systems are isolated automatically by the containment isolation system,

(B) For each non-essential 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 set point 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.

Response

The containment isolation system has been reviewed in accordance with NUREG-0737, Item II.E.4.2. See Section 1A.29 and Attachment C to Appendix 1A.

Criterion (C) above required design modification as described in Subsection 1AC.3. NRC staff approval is given in NUREG 0979, Section 6.2.7 (Page 6-33).

1G.27 [redacted] PURGING [redacted] [Item (2)(xv)]

NRC Position

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)

Response

The general safety concern over containment purging stems from the presumption that the purge line provides a path for accident releases prior to isolation, and further, that the dynamic effects of the accident may interfere with effective isolation of the purge line.

These presumptions are not directly applicable to the Mark III containment of the GESSAR II design. The reactor coolant system piping is enclosed in the drywell which communicates with the containment only through the suppression pool. Releases from the primary system are subjected to the quenching and scrubbing action of the suppression pool before entering the containment, so the purge system does not provide a path for primary system releases in the same sense as other containment designs. Even so, special care is being taken in the purge system design, specifically for valve operability assurance. ~~(Items 2 and 3 below)~~ Finally, the isolation valves are provided with positive leakage control.

¶ The specific points of ~~NUREG 0718~~ of this item are addressed below:

~~Purging Consistent with ALARA~~ 6.28 (480.23)

The basis for the purge system design is justified in the response to ~~NRC~~ Question ~~042.63 and was found acceptable in Section 6.2.15 of the Skagit SER (NUREG-0309).~~

N. ¶ The present 9" design provides for continuous purging of the containment during power operation at 5000 cfm through an ~~10"~~ line to reduce airborne radionuclide concentrations to a level which permits continuous access. This is in keeping with occupational ALARA considerations, because extensive containment access for routine maintenance is required.

~~Performance of Purge and Vent Valves Against Accident Pressure~~

¶ The performance of prototype 6" ~~the~~ purge isolation valves has been evaluated and meets the requirements of BTP CSB 6-4, ~~[illegible]~~ for isolation and dependability under accident pressures. The Applicant will provide performance information for the specific 9" purge isolation valves. These purge valves are conservatively designed to close against the containment design pressure of 15 psig ~~even though~~ (the drywell would only be a few psi at start of closure). The performance of the 42" purge valves will be demonstrated during refueling when the pressure differential is negligible.

19.3.6.28 QUESTION/RESPONSE 6.28 (480.23)

QUESTION 6.28

REVISED RESPONSE

You state on pages 1.8-171 and 9.4-62 of your FSAR that the containment purge valves are open during normal operation and that containment pressure is controlled through these valves. However, we state our position on this matter in branch technical position CSB6-4 that the use of large purge and vent lines should be restricted to cold shutdown conditions and refueling outages. Provide your basic for purging the containment continuously in light of our position on this matter. (6.2.4)

RESPONSE 6.28

Continuous purging of the primary containment outside of the drywell during reactor operation is required for access, inspection, and maintenance associated with the control rod drive - hydraulic control units (one per CRD); safety related instrument calibrations, water sampling of reactor water, suppression pool, and upper containment pools, RWCU system and feedwater.

These activities involve several operating personnel occupying the primary containment during a significant portion of each shift.

The ventilation rate of 5,000 cfm provides an air change in the containment only every 3 hours and 45 minutes. This is minimal for controlling humidity, odors, and dilution of potential airborne radioactivity released from a small number of safety/relief valve vents, RWCU System filter-demineralizer maintenance, and upper containment pool walls at the wet-dry line.

19.3.6.28 QUESTION/RESPONSE 6.28 (480.23) (Continued)

During the NRC staff review for Preliminary Design Approval, ~~When Branch Technical Position CSB 6-4 was initially issued,~~ the containment purge ~~ventilation~~ penetrations were modified to reduce their size for normal operating continuous purge from a 42-in. diameter to an 18-in. diameter. ~~The~~ larger 42-in. penetrations ~~was~~ are open only during reactor shutdown and refueling to allow for higher ventilation flow rates when more operating personnel would be present and potential airborne radiation levels could be higher than during normal operation.

Fast closing isolation valves [redacted] ~~were~~ are provided to close on LOCA signal or whenever the exhaust radiation sensors detected radiation levels high enough to exceed plant operating limits. During normal reactor operations with the 5,000 cfm ventilation rate, the airborne radiation levels must be less than 10CFR20 limits inside the primary containment but outside the drywell. The drywell purge vents are closed during reactor operation. ~~and this is normal for all BWRs.~~

The radiation monitors located in the exhaust duct are installed far enough away from the primary containment isolation valves so these valves can close before airborne radiation is released from the primary containment. As a further precaution, radiation monitors are also located near the upper containment pool surface so early detection of potential radioactivity can be detected during refueling when the reactor is open.

A further reduction in size from 18-in. diameter to 9-in. diameter for normal operating continuous purge was made as a result of the NRC staff review for Final Design Approval of the GESSAR II design. ~~As with the preliminary design~~ In addition, these 9-in diameter penetrations ~~are~~ for continuous purge during operation are now separate. Thus, the

19.3.6.28 QUESTION/RESPONSE 6.28 (480.23) (Continued)

~~A~~ The suppression pool cleanup system ~~has been provided for the BWR-6, Mark III to~~ ensures that radioactivity in the pool water can be kept low and to reduce the amount of airborne radioactivity during abnormal plant operating events.

This same topic ~~was~~ is also an issue contained in NUREG-0737, Item II.E.4.2, which ~~was~~ is addressed in Section 1A.29 "Containment Isolation Dependability." Studies of suppression pool scrubbing action have been found to retain considerable amounts of iodine in the event of a major pipe break and failure of fuel cladding. This same capability also applies to normal and abnormal reactor operations when potential safety/relief valve simmering occurs, and RCIC testing releases condensed reactor steam to the suppression pool. All of these plant operating conditions have been studied and discussed with the staff during the various reviews and licensing ~~procedures.~~ of GESSAR II

Periodic isolation valve testing is required during reactor operation to ensure that the Ventilation Isolation System is ever ready to function. This includes Appendix "J" leakage testing and radiation sensor test and calibration.

DESIGN EVALUATOR

1G.28 ~~ECCS AND RPS DESIGN CRITERION~~ [Item (2)(xvi)]

NRC-Position

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)

Response

This requirement is not applicable to the 238 Nuclear Island. It applies only to PWR-type (B&W designed) reactors.

ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

1G.29 ~~CONTAINMENT PARAMETERS INSTRUMENTATION IN CONTROL ROOM?~~
[Item (2)(xvii)]

NRC Position

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)

Response

Discussion of each of these variables is contained in appendix 1D.

1G.30 IDENTIFICATION OF AND RECOVERY FROM CONDITIONS LEADING TO INADEQUATE ~~UNAMBIGUOUS INDICATION OF INADEQUATE~~ CORE COOLING [Item (2)(xviii)]

NRC Position

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

Response

As indicated in Subsection 1A.31, GE believes the existing highly redundant direct water level instrumentation already provides an unambiguous indication of inadequate core cooling and does not plan to include core-exit thermocouples in the 238 Nuclear Island design.

The NRC staff agreed to broaden the issue from the specific requirements for incore thermocouples to that of monitoring inadequate core cooling (ICC). (See NUREG 0979, Section 4.4.7, Page 4-35.) This issue is contained within Regulatory Guide 1.97 and will be resolved in conjunction with GESSAR II's conformance with the guide as indicated in the response to Section 1G.29.

INSTRUMENTATION FOR MONITORING ACCIDENT CONDITIONS (REG. GUIDE 1.97)

1G.31 ~~POST ACCIDENT MONITORING~~ [Item (2)(xix)]

NRC Position

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

Response

The GESSAR II design is assessed against Regulatory Guide 1.97 Revision 2 in Appendix 1D with supplementary justification for deviations provided on the docket (cover letter G. G. Sherwood to D. G. Eisenhut dated April 28, 1983). The assessment including the deviations and justifications are still under review by the NRC.

GE agrees that the GESSAR II design will meet designated portions of the guide; and any deviations to the remainder of the guide will be justified to the satisfaction of the staff prior to referencing GESSAR II by the first applicant.

POWER SUPPLIES FOR PRESSURIZER RELIEF VALVES, BLOCK VALVES & LEVEL INDICATION

1G.32 ~~PRESSURIZER INSTRUMENTATION~~ [Item (2)(xx)]

NRC Position

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 PWR's only). (II.G.1)

Response

This requirement is not applicable to the 238 Nuclear Island. It applies only to PWR-type reactors.

DESCRIBE AUTOMATIC & MANUAL ACTIONS FOR PROPER FUNCTIONING OF AUXILIARY HEAT REMOVAL SYSTEMS WHEN FW SYSTEM NOT OPERABLE

1G.33 ~~AUXILIARY HEAT REMOVAL SYSTEM DESIGN AND PROCEDURES~~
[Item (2)(xxi)]

NRC Position

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 BWR's only). (II.K.1.22)

Response

The actions of the auxiliary heat removal system for these conditions is described in Subsection 1A.38. the NRC Staff approval is documented in NUREG 0979, Section 5.4.2, pages 5-26 & 5-27.

1G.34 ~~FMEA~~ ANALYSIS OF UPGRADING OF INTEGRATED CONTROL SYSTEM [Item (2)(xxii)]

NRC Position

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)

Response

This requirement is not applicable to the 238 Nuclear Island. It applies only to PWR-type (B&W designed) reactors.

HAND-WIRED SAFETY-GRADE
1G.35 ANTICIPATORY REACTOR TRIPS [Item (2)(xxiii)]

NRC-Position

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)

Response

This requirement is not applicable to the 238 Nuclear Island. It applies only to PWR-type (B&W designed) reactors.

1G.36 ~~POST-ACCIDENT~~ CENTRAL WATER LEVEL ~~INSTRUMENTATION~~ RECORDING [Item (2)(xxiv)]

NRC Position

Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (Applicable to BWR's only). (II.K.3.23)

Response

The reactor vessel water level instrumentation used for post-accident recording is described in Subsection 1D.2-3-4. A technical description for enhanced level recording is described in attachment C to appendix 1D.

UPGRADE LICENSE EMERGENCY SUPPORT FACILITY
1G.37 ~~ONSITE SUPPORT CENTERS~~ [Item (2)(xxv)]

NRC Position

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)

Response

The response to this requirement will be provided by the applicant.

1G.38 ~~LEAKAGE CONTROL PROGRAM FOR SYSTEMS~~ PRIMARY COOLENT SOURCES OUTSIDE THE CONTAINMENT STRUCTURE
[Item (2)(xxvi)]

NRC-Position

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)

Response

The leakage control program is detailed in Section 1A.77. NRC acceptance is found in NUREG 0979, Section 9.3.4 (Page 9-24) of the TMI action plan.

1G.39 INPLANT RADIATION MONITORING ~~AND AIRBORNE RADIOACTIVITY~~ [Item (2)(xxvii)]

NRC-Position

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

Response

The response to this requirement will be supplied by the applicant.

1G.40 CONTROL ROOM HABITABILITY [Item (2)(xxviii)]

NRC Position

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)

Response

As indicated in Section 1A.79, analysis demonstrates no changes to the 238 Nuclear Island are required to satisfy this item.

PROCEDURES FOR FEEDBACK OF OPERATING, DESIGN AND CONSTRUCTION EXPERIENCE

1G.41 ~~ADMINISTRATIVE PROCEDURES FOR EVALUATING EXPERIENCE~~
[Item (3)(i)]

NRC Position

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)

Response

The response to this requirement will be supplied by the applicant.

1G.42 ~~EXPANDED QUALITY ASSURANCE (QA) LIST~~ EXPAND QA LIST [Item (3)(ii)]

NRC Position

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

Response

As discussed in Subsection 17.1.2, ~~T~~the identification of safety-related structures, systems, and components (Q-list) to be controlled by the quality assurance program is the responsibility of the Applicant. The Applicant will supplement and clarify its Q-list in accordance with Question 17.3 (NRC Question 260.3). The appropriate items will be added to Table 3.2-1. The remaining items will be subject to the pertinent requirements of GE's and/or the Applicant's QA programs unless otherwise justified.

DEVELOP MORE

1G.43 DETAILED QA CRITERIA [Item (3)(iii)]

NRC Position

Establish a quality assurance (QA) program based on consideration of: (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; and (H) providing a QA role in design and analysis activities. (I.F.2)

Response

(LATER)

1G.44 DEDICATED CONTAINMENT PENETRATIONS, ~~[Item (3)(iv)]~~ EQUIVALENT TO A SINGLE 3-FOOT DIAMETER OPENING [Item (3)(iv)]

NRC Position

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)

Response

As a result of the NRC staff review for Final Design Approval, GE has agreed to provide separate penetrations for purging during operation and during refueling. ~~However,~~ The isolation valves for the 42-in diameter refueling purge penetrations are now locked closed during normal operation and have been dedicated as containment penetrations as required by this item. The details of the penetration arrangement will be provided prior to the first Applicant references GESSAR II.

1G.45 CONTAINMENT INTEGRITY [Item (3)(v)]

NRC Position

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 Pressure Vessel Code, Section III, Division 2 Subsubarticle CC-3720, Factored 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 post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above 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)

1G.45 CONTAINMENT INTEGRITY [Item (3)(v)] (Continued)

of ~~this section~~ 10CFR50.34, 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 post-accident 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.

1G.45 CONTAINMENT INTEGRITY [Item (3)(v)] (Continued)

Response

(A) The containment design basis is 15 psig. Appendix G of Subsection 15D.3 (PRA) provides the corresponding containment capability analyses. These analyses demonstrate that all areas of the containment exceed 45 psig Service Level C Limits except for the knuckle region of the 2:1 torispherical ~~dome~~ head. Preliminary analysis indicates that the knuckle region ~~can not~~ can also exceed 45 psig Service Level C Limits by modifying the curvature of the head using a three-center design. (no other modifications are necessary).

The containment vessel ~~is the~~ design capability ~~Applicant's responsibility; however, the~~ of 45 psig Service Level C ~~limits~~ ~~the containment capability~~ is now included as an interface requirement (Table 1.9-1).

(B) Containment structure loadings produced by an inadvertent full actuation of a post accident ~~not~~ inerting hydrogen control system shall not exceed ASME Section III, NE 3220 Service Level A Limits as calculated per 10CFR50.34(f)(3)(v)(B).

1G.46 DEDICATED PENETRATION ~~EXTERNAL HYDROGEN RECOMBINERS~~ [Item 3 ~~(A)~~ ~~(vi)~~]

NRC Position

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)

Response

As indicated in Section 1A.28, this requirement is not applicable to the 238 Nuclear Island (GESSAR II) design. NRC Concurrence is given in NUREG 0979, Section 6.2.7 (page 6-33).

1G.47 ORGANIZATION AND STAFFING TO OVERSEE ~~MANAGEMENT PLANNING AND PROCEDURES FOR PLANT~~ DESIGN AND CONSTRUCTION [Item (3)(vii)]

NRC Position

Provide a description of the management plant for design and construction activities, to include: (A) the organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director 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. (II.J.3.1)

Response

(A) The organizational and management structure is dipicted in Figure 1G.47-1.

(continue next page here)

~~Supplier~~ designer

No ¶ The Utility evaluates the type of LWR desired. Studies are conducted to select an appropriate site. A ~~[illegible]~~ is selected and licensing of the proposed site is secured by the Applicant.

No ¶ General Electric supplies the pre-approved GESSAR II ~~Nuclear Island~~ design and furnishes technical direction during procurement and construction. Changes, non-conformances and as-builts are reviewed and approved and ~~[illegible]~~ by General Electric as the engineer of record for the Nuclear Island.

~~The Turbine Island and etc.~~

The constructor is selected by the Applicant and under his direct control.

B (Tentative answer, depending on more precise definition of info. wanted)

(B) The utility that applies ~~a~~ GESSAR II ~~[illegible]~~ ~~Nuclear Island~~ obtains a design that is basically completed. Therefore, an ~~A/E~~ architect-engineer (A/E) is generally retained to furnish the closure engineering effort that may be required. The A/E is a subcontractor to the utility and the utility can mobilize the full resources of the A/E organization. Likewise, General Electric has the capacity ~~[illegible]~~ to mobilize a very large amount of qualified technical personnel if so requested or deemed necessary by the utility.

(C) Utilization of ~~the~~ the GESSAR II ~~the~~ design greatly simplifies interfaces. The coordination between the NSSS & the Balance of the Nuclear Island is performed internally by General Electric using existing procedures. Since the GESSAR II ~~Nuclear Island~~ design is essentially complete, the amount of interface required is further simplified from ~~previous non-standardized, non preapproved concepts~~ custom designs.

The Utility has to verify that the preselected and pre approved site meets the envelope requirements of the GESSAR II ~~Nuclear Island~~. Therefore, the services of an A/E organization are generally retained by the utility. The utility forms a project group under the leadership of an experienced executive that oversees all the principal organizations: General Electric; Constructor and "house" A/E.

This Project group within the Utility organization is responsible to coordinate among contractors, verify contractual performance, monitor schedules, licensing and support construction activities.

~~Jack: We can add some "fluff" here regarding what we did in STRIDE to control design.~~

(D) Transition of operations from construction to operation of the plant is Utility unique & should be supplied by each applicant.

(E) Generally, each Project is managed by a senior executive within the utility who reports to the Vice President of Engineering. Timely reports are prepared on status of all major contracts. Discreet reviews are systematically performed and problems are tracked to successful resolution.

The V.P. of Engr. is intimately familiar with the design & construction of each Nuclear plant being added to the company's grid since it represents a very large percentage of the utilities' investments. Delays and changes are extremely costly and are constantly monitored. Procedures and guidelines implementation

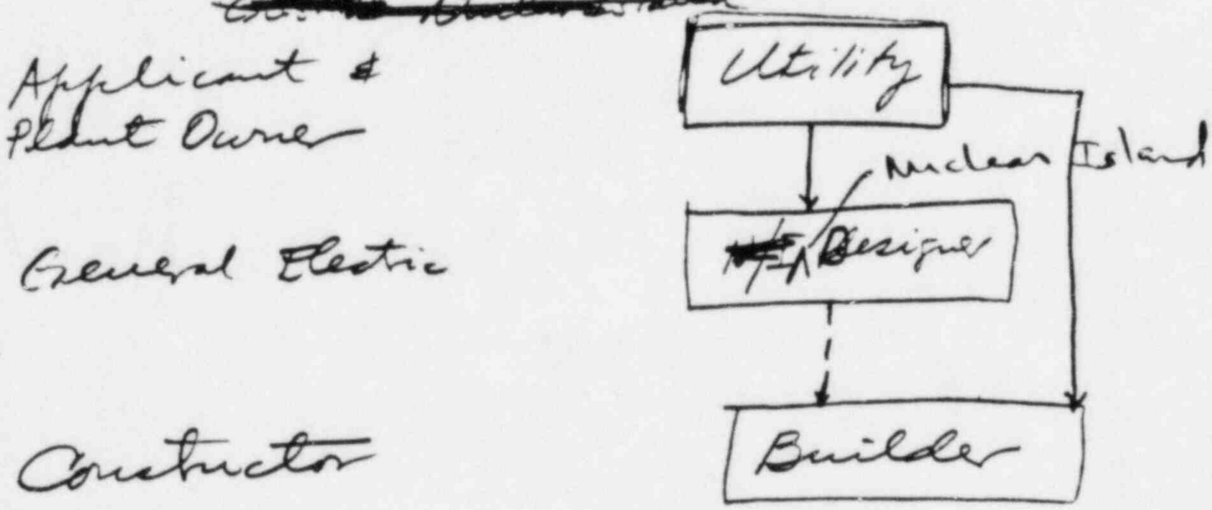
are normally reported by the QA/QC organization and they impact directly the job's progress which is thoroughly reviewed by top management.

Reviews are periodically made by means of memos; letters; standard forms; meetings and telephone calls.

General Electric's top management is responsible for the implementation of all procedures and they must sign off on pertinent sheet to indicate agreement/disagreement with the contents. Periodic, systematic reviews are made by top management in order to responsibly control the work.

Figure 16.47-1 ~~cc: J. Fox~~
GESSAR II ~~L. McC~~
ORGANIZATIONAL AND MANAGEMENT STRUCTURE

~~GESSAR Nuclear Island~~



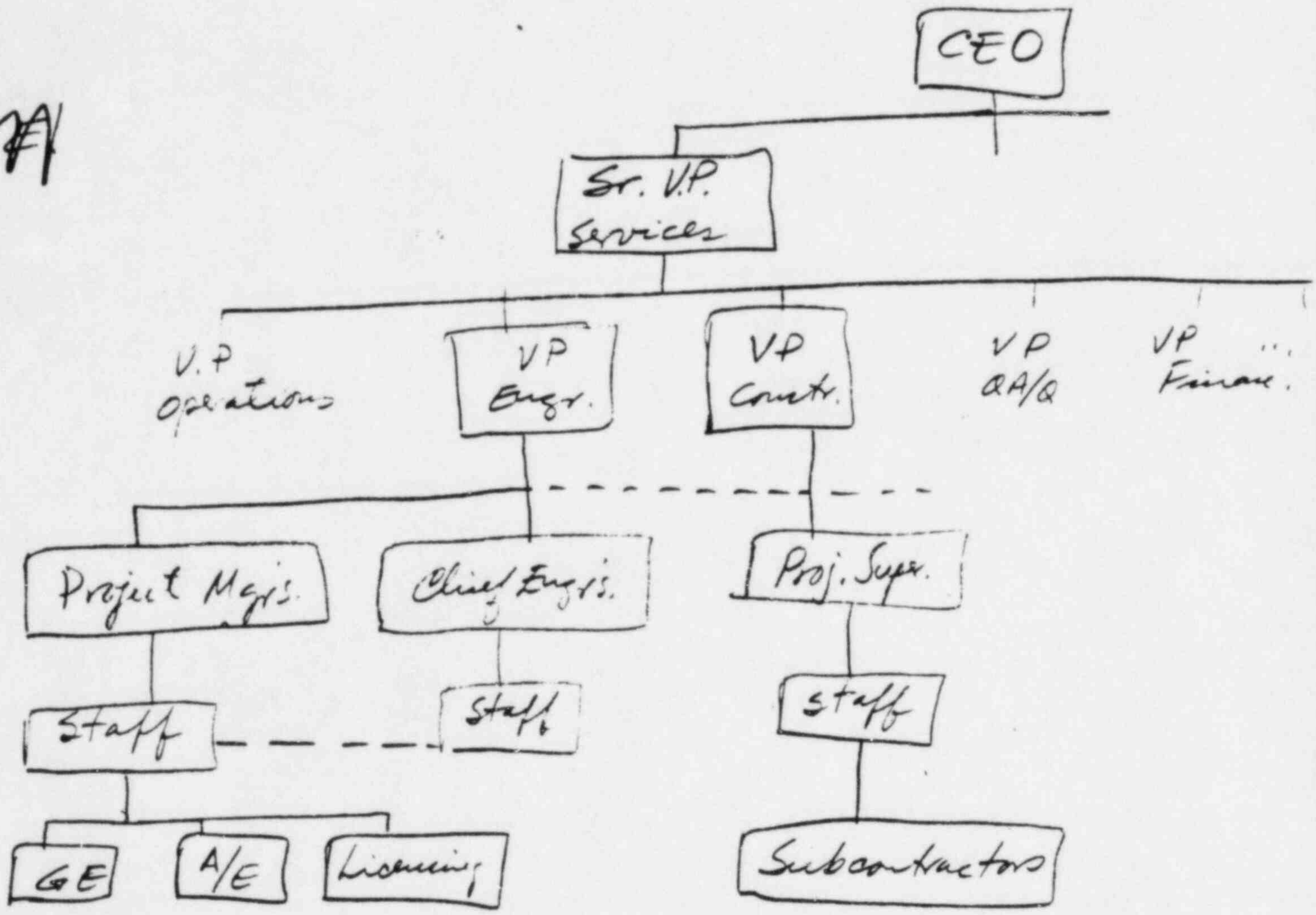


Figure 1G.47-2
Top Level Management Oversight and Technical Control

ATTACHMENT NO. 2

DRAFT OF GESSAR II
AMENDMENT NO. 17
RESPONSE TO SRP RULE

SECTION 1.8

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1.8 CONFORMANCE WITH THE STANDARD REVIEW PLAN

1.8.0 Purpose

The purpose of this section is to provide an evaluation of the GESSAR II design against the Standard Review Plan (NUREG-0800) as required by 10 CFR 50.34(g). Since the NRC regulatory guides are an integral part of NUREG-0800, this section also shows the consistency of the design with the regulatory guides.

1.8.0.1 Differences from Standard Review Plan

Since the GESSAR II design scope is limited to the Nuclear Island, there are Balance of Plant (BOP) portions that are the responsibility of the Applicant. In addition, the Applicant is responsible for information within the scope of the Nuclear Island that will not be available until GESSAR II is utilized by an Applicant. Finally, GE has chosen for commercial reasons to delay the submittal of certain information until the first Applicant references GESSAR II. All of this information is presented in Tables 1.9-1 through 1.9-19. Hence, it is not possible at this time to demonstrate that the GESSAR II design satisifies all of the NUREG-0800 requirements. However, GE has reviewed the GESSAR II design against the revelent portions of NUREG-0800, and concludes that it meets the applicable acceptance criteria, except as noted in Table 1.8.0-0. The cited references include evaluations that describe the basis by which GE concludes that the underlying requirements are satisified.

The Applicant will provide a summary of deviations from NUREG-0800 for those plant design features covered by the GESSAR II/FSAR interface Tables 1.9-1 through 1.9-19 with corresponding evaluations that describe the basis by which the Applicant concludes that the underlying requirements are satisified.

Table 1.8.0-0

SUMMARY OF DEVIATIONS FROM NUREG-0800

NUREG-0800 SECTION	NUREG-0800 CRITERIA	DIFFERENCE	GESSAR II SUBSECTION
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

Table 1.8.0-0 (continued)
SUMMARY OF DEVIATIONS FROM NUREG-0800

NUREG-0800 SECTION	NUREG-0800 CRITERIA	DIFFERENCE	GESSAR II SUBSECTION
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 ass'y.	Design basis allows up to 0.52" "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 ≤ than 90% of the irradiated yield stress.	Topical Report is under review.	4.2.1
4.2 (Rev. 2)	II.A.3.(e) - Analytical procedure 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 under review	4.2.2

Table 1.8.0-0 (continued)
SUMMARY OF DEVIATIONS FROM NUREG-0800

NUREG-0800 SECTION	NUREG-0800 CRITERIA	DIFFERENCE	GESSAR II SUBSECTION
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) - Regulatory Guide 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 require operator action within 10 minutes for some events.	19.3.6.56
6.7 (Rev. 2)	II.1 - MSIV leakage control meeting Regulatory Guide 1.96.	Exception taken to Position C.9 of Regulatory Guide 1.96.	1.8.96
7.1 (Rev. 2)	II - Regulatory Guide 1.75 (Table 7-1).	Alternates to portions of R.G. 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)

Table 1.8.0-0 (continued)
SUMMARY OF DEVIATIONS FROM NUREG-0800

NUREG-0800 SECTION	NUREG-0800 CRITERIA	DIFFERENCE	GESSAR II SUBSECTION
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 on low water level if the initiation signal is syil.	1A.63
7.3 (Rev. 2)	II - Paragraph 4.17 of IEEE 279	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 - Regulatory Guide 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 ± s% on grid voltage.	19.3.8.5
9.5.1 (Rev. 3)	II.2.a- Implementation of fire protection program in accordance with BTP CMEM 9.5-1.	Lack of 3-hour-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
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

Table 1.8.0-0 (continued)
SUMMARY OF DEVIATIONS FROM NUREG-0800

NUREG-0800 SECTION	NUREG-0800 CRITERIA	DIFFERENCE	GESSAR II SUBSECTION
15.3.3 - 15.3.4 (Rev. 2)	II.8 - Use of non-safety grade equipment.	Credit is taken for non-safety grade equipment and failure of non-safety 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

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

The purpose of this section is to show that the design of the 238 Nuclear Island is consistent with the requirements of the regulatory guides issued by the NRC.

~~1.8.0 Summary~~
1.8.0.2 Consistency with NRC Regulatory Guides

The NRC (AEC) began in 1970 to issue regulatory guides (safety guides) which state, in detail, methods acceptable to the NRC staff of meeting applicable Federal Regulations. Since that time, new and revised regulatory guides have been issued on an on-going basis.

During the construction permit stage, GE agreed in GESSAR PDA to comply with the appropriate regulatory guides issued through March 1, 1974 (Regulatory Guides 1.1-1.75), plus Regulatory Guides 1.76, 1.89, and 1.96. For the FDA, however, GE elected to base GESSAR II on compliance with all regulatory guides in effect as of the date of docketing. Therefore, Regulatory Guides 1.1 through 1.150, 8.8 and 8.19 with the revisions in effect as of February 22, 1982 are applicable to GESSAR II.

Table 1.8.0-1 lists these regulatory guides (and revisions) used as design bases and defines the GESSAR II subsection that describes the manner in which the design complies with the applicable regulatory guide.

1.8.122 Regulatory Guide 1.122, Revision 1, Dated February 1978

Title: Development of Floor Design Spectra for Seismic Design of Floor Supported Equipment in Components

This guide describes methods for developing design response spectra at various floors or other equipment support locations of interest from the time-history motions resulting from the dynamic analysis of the supporting structures.

Evaluation

GE complies with all guidelines of the regulatory guide except Position C.2 where, instead of using ±15 percent in frequency for spectrum broadening, GE uses ±10 percent. Justification of this exception is provided in GESSAR II Subsection ~~3.7.3.7~~ 3.7.2.9 which illustrates the conservative assumptions that have been included in the calculation of the floor response spectra.

Table 1.9-1

CHAPTER 1
GESSAR II/FSAR INTERFACES (Continued)

ITEM NO.	SUBJECT	DESCRIPTION	PAGE	SUBSECTION	RELATED QUESTION	INTERFACE CATEGORY
1.20.1	Differences from Standard Review Plan	Identify differences in plant design features covered by Tables 1.9-1 through 1.9-13 with evaluations that describe the basis to conclude the underlying requirements are met.	1.8.0-1	1.8.0.1		3

Table 1.9-4

CHAPTER 4
GESSAR II/FSAR INTERFACES

ITEM NO.	SUBJECT	DESCRIPTION	PAGE	SUBSECTION	RELATED QUESTION	INTERFACE CATEGORY
4.1	Core Loading Pattern	Provide fuel designations and number loaded for the reference core loading pattern.	4.3-6	Table 4.3-1		3
4.2	Core Loading Pattern	Provide reference core loading pattern figure.	4.3-9a	Figure 4.3-1		3
4.3	Loose Parts	Describe monitoring equipment and procedures to be used to detect excessive vibration and the occurrence of loose parts per R.G. 1.70 Subsection 4.4.6~~.~~ and R.G. 1.133.	4.4-10	4.4.6.1		1
4.4	Safety Injection Lines	Provide information for safety injection lines.	4.4-19	Table 4.4-6		3
4.5	Post irradiation Surveillance Program	Provide a post irradiation surveillance program	4.2-4	4.2.4		3
4.6	Thermal Hydraulic Stability	Provide stability analysis using decay ratio agreed by the NRC staff.	4.4-10	4.4.4		3
4.7	Reactor Coolant pressure drop.	Assure that Process Monitoring System is capable of 3% pressure drop in the coolant flow with flow test every 24 hours.	4.4-10	4.4.5		3.

New Page

Table 1.9-11

CHAPTER 11

GESSAR II /FSAR INTERFACES (continued)

ITEM No.	SUBJECT	DESCRIPTION	PAGE	SUBSECTION	RELATED QUESTION	INTERFACE CATEGORY
11.12	Isolation Valves, dampers actuation Set Points	Provide in the Plant technical Specification Set points for actuation of valves, dampers or diversion valves.	11.5-23	11.5.3.4		3

Table 1.9-15

CHAPTER 15
GESSAR II/FSAR INTERFACES (Continued)

ITEM NO.	SUBJECT	DESCRIPTION	PAGE	SUBSECTION	RELATED QUESTION	INTERFACE CATEGORY
15.8	Rod Withdrawal Error	Provide results of either the generic or plant specific rod withdrawal error event per R.G. 1.70, Chapter 15.	15.4-7	15.4.2.3.2		3
15.9	Misplaced Bundle Accident	Analyze the misplaced bundle accident using the plant specific core configuration per R.G. 1.70, Chapter 15.	15.4-17	15.4.7.3		3
15.10	Dispersion Data Control Rod Drop	Provide site boundary and low population zone distances, using both design and realistic assumptions in the control rod drop accident.	15.4-37	Table 15.4-12		4
15.11	Dispersion Data Control Rod Drop	Provide site boundary and low population zone distances, using both design and realistic assumptions in the control rod drop accident.	15.4-37	Table 15.4-12		4
15.12	Dispersion Data Steamline Break	Provide site boundary and low population zone distances, using both design and realistic assumptions in the steamline break accident.	15.6-31	Table 15.6-2		4
15.13	Dispersion Data LOCA	Provide site boundary and low population zone distances, using both design and realistic assumptions in the loss-of-coolant accident.	15.6-37	Table 15.6-7		4
15.9.1	Fuel loading errors.	Include in the plant operating procedures/ Technical Specifications provisions for potential fuel loading errors.	15.4-16	15.4.7.1.1		3

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Piping

Fifty (50) peak OBE cycles are postulated for fatigue evaluation 3.150

3.7.3.2.2 Other Equipment and Components

INSERT 3.7.3.2.2

To evaluate the number of cycles engendered by a given earthquake, a typical Boiling Water Reactor Building reactor dynamic model was excited by three different recorded time histories: May 18, 1940, El Centro NS component, 29.4 sec; 1952, Taft N69° W component, 30 sec; and March 1957, Golden Gates 89°E component, 13.2 sec. The modal response was truncated so that the response of three different frequency bandwidths could be studied, 0^{+}-to-10 Hz, 10-to-20 Hz, and 20-to-50 Hz. This was done to give a good approximation to the cyclic behavior expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior given in Table 3.7-51 was formed.

Independent of earthquake or component frequency, 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level.

3.150

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake is found in the following manner:

(1) the fundamental frequency and peak seismic loads are found by a standard seismic analysis (i.e., from eigen extraction and forced response analysis);

INSERT 3.7.3.2.2

The SRP 3.7.3 criteria II.2.b recommends that at least one safe shutdown earthquake (SSE) and five operating basis earthquake (OBE) should be assumed during the plant life. It also recommends that a minumum of 10 maximum stress cycles per earthquake should be assumed (i.e. 10 cycles for SSE and 50 cycles for OBE.) For equipment and components other than piping, 10 peak OBE stress cycles are postulated for fatigue evaluation based on the following justification.

4.2 FUEL SYSTEM DESIGN

See Appendix A, Section A.4.2 of Reference 1.

4.2.1 Design Bases

See Appendix A, Subsection A.4.2.1 of Reference 1.

→ INSERT 4.2.1

4.2.2 Description and Design Drawings

See Appendix A, Subsection A.4.2.2 of Reference 1.

→ INSERT 4.2.2

4.2.2.1 Control Rods

The control rods perform the dual function of power shaping and reactivity control. A design drawing of the control blade is seen in Figure 4.2-1 and 2. Power distribution in the core is controlled during operation of the reactor by manipulating selected patterns of control rods. Control rod displacement tends to counterbalance steam void effects at the top of the core and results in significant power flattening.

The control rod consists of a sheathed cruciform array of stainless steel tubes filled with boron-carbide powder. The control rods are 9.868 in. in total span and are separated uniformly throughout the core on a 12-in. pitch. Each control rod is surrounded by four fuel assemblies.

The main structural member of a control rod is made of Type-304 stainless steel and consists of a top handle, a bottom casting with a velocity limiter and control rod drive coupling, a vertical cruciform center post, and four U-shaped absorber tube sheaths. The top handle, bottom casting, and center post are welded into a single skeletal structure.

INSERT 4.2.1

Acceptance Criterion II.A.1.(b) of SRP Section 4.2 requires that the cumulative number of strain fatigue cycles on the structural members of the fuel system should be significantly less than the design fatigue lifetime, which is based on appropriate data and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles. The design limit for fatigue cycling in Ref. 1 has the following limiting condition:

$$\frac{\text{Actual time at stress}}{\text{Allowable time at stress}} + \frac{\text{Actual number of cycles at stress}}{\text{Allowable number of cycles at stress}} \leq 1.0$$

Since the Ref. 1 limit is more conservative than that of the SRP, the deviation is acceptable.

Acceptance Criterion II.A.1.(c) requires that the allowable fretting wear on major structural members of the fuel assembly be stated. The GESSAR II fretting wear design basis design for fuel system components (letter, Charnley to Staff, Jan. 25, 1983) is: the fuel assembly components. This statement plus the discussion on fretting wear in Section 2.6.3 of Ref. 1 show that fretting wear is considered in the design analysis and the intent of the SRP is met.

Acceptance Criteria II.A.1.(g) of SRP Section 4.2 requires that the fuel assembly hold down capability (gravity and sprimp) exceed the worst-case hydraulic loads for normal operation, which includes anticipated operation occurrences. The GESSAR II design limit for fuel assembly lift off is 0.52" as documented in NUREG-0979. This limit was calculated to be the largest which would not permit sufficient lateral displacement of the fuel assembly to result in control blade interference. Since control blade interference is prevented, this design limit is acceptable.

Acceptance Criterion II.A.2.(e) states that for normal operation and anticipated operational occurrences centerline melting of the fuel is not permitted. The GESSAR II design basis for fuel pellet overheating (Letter, J.S. Charnley to NRC Staff, January 25, 1983) is: the fuel rod is evaluated to ensure that fuel rod failure due to excessive fuel melting will not occur during steady - state operation. This design limitation clearly shown that the GE design objective is to avoid fuel failures due to fuel melting and thus meets the intent of the SRP criterion.

INSERT 4.2.1 (Continued)

Acceptance Criterion II.A.2.(g) states that the uniform fuel cladding strain (plastic & elastic) should not exceed 1.0% (steady-state creepdown and irradiation growth are excluded). The Reference 1 model for evaluation of the 1% strain limit does not include elastic strain. The basis for the model is contained in Appendix A, Subsection A.4.2.1 of Reference 1.

Acceptance Criterion II.A.2.(i) limits the applied stress on the cladding to 90% of the irradiated yield stress at the appropriate temperature. The mechanical fracture analysis for the GESSAR II fuel design is given in a topical report on the LOCA and SSE loads evaluation, NEDE-21175-3, which is currently under evaluation by the NRC staff. In NEDE-21175-3, the maximum externally applied load on the fuel cladding is determined to be less than 60% of the irradiated ultimate tensile strength at the appropriate temperature. The cladding design is thus concluded to be adequate in terms of resistance to mechanical fracturing.

Acceptance Criterion II.A.3.(e) describes analytical procedures for the determination of fuel assembly structural deformation. The GESSAR II fuel assembly structural analysis is described in Topical Report NEDE-21175-3-P. In this report, each major fuel assembly component part is shown to be functionally adequate to withstand the separate and combined peak loadings from the dynamic and LOCA blowdown events without experiencing structural failure.

INSERT 4.2.2

Acceptance Criterion II.B. lists design parameters and drainage to be included in the fuel system description. The GESSAR II fuel system description, given in Ref. 1, does not include all of the design parameters listed in Acceptance Criterion II.B. However, sufficient information is given to provide a reasonably accurate representation of the GESSAR II fuel system, satisfying the intent of the SRP.

4.2.2.2 Velocity Limiter (Continued)

The velocity limiter is in the form of two nearly mated, conical elements that act as a large clearance piston inside the control rod guide tube. The lower conical element is separated from the upper conical element by four radial spacers 90 degrees apart and is at a 15-degree angle relative to the upper conical element, with the peripheral separation less than the central separation.

The hydraulic drag forces on a control rod are proportional to approximately the square of the rod velocity and are negligible at normal rod withdrawal or rod insertion speeds. However, during the scram stroke, the rod reaches high velocity, and the drag forces must be overcome by the drive mechanism.

To limit control rod velocity during dropout, but not during scram, the velocity limiter is provided with a streamlined profile in the scram (upward) direction.

Thus, when the control rod is scrammed, water flows over the smooth surface of the upper conical element into the annulus between the guide tube and the limiter. In the dropout direction, however, water is trapped by the lower conical element and discharged through the annulus between the two conical sections. Because this water is jetted in a partially reversed direction into water flowing upward in the annulus, a severe turbulence is created, thereby slowing the descent of the control rod assembly to less than 3.11 ft/sec.

4.2.3 Design Evaluation

See Appendix A, Subsection A.4.2.3 of Reference 1.

Acceptance Criterion II.C.3(a) lists phenomonological models to be included in fuel system thermal calculations. The GESSAR II fuel thermal model does not include the use of approved gadolinia fuel properties. However, as discussed with the NRC staff, the General Electric Company does not license material properties for design analyses but, rather, maintains these analyses up-to-date. To fulfill our quality control obligations under 10CFR50, Appendix B, the latest property values are incorporated into design applications only after they are qualified in the design code. An improved fuel rod thermal-mechanical design code has recently been developed and qualified which includes the revised gadolinia fuel thermal conductivity relations. The results of the fuel centermelting analysis using this improved fuel rod design code verifies that gadolinia fuel melting is not expected to occur during normal steady-state operation or during the largest whole core anticipated operational transient.

The GESTAR II (NEDE-24011-P-A) amendment incorporating the application of the above thermal-mechanical design code is currently under review by the Core Performance Branch. Since GESSAR II references the "latest approved revision" of GESTAR II, this issue will be resolved when the GESTAR II amendment is approved.

Acceptance Criterion II.C.3.(d) describes acceptance criteria for evaluation of fuel assembly structural response to externally applied forces.

An analysis has been performed (NEDE-21175-3) to show that the GESSAR II fuel meets structural requirements (including liftoff) similar to those of Appendix A of Section 4.2 of the SRP (NUREG-0800). That analysis is currently under review by the NRC staff. Because previous generic analytical methods presented in earlier versions of NEDE-21175 have been approved by the NRC staff (letter from O, P. Parr (NRC), May 17, 1979) and because favorable sample results were also presented in Amendment 2 of NEDE-21175, the new GE analysis is expected to be approved.

4.2.4 Testing, Inspection and Surveillance Plans

See Appendix A, Subsection A.4.2.4 of Reference 1.

4.2.5 References

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, latest approved revision.

¶ The Applicant will provide a routine fuel inspection program to provide information on irradiated and discharged fuel as indicated in SRP Section 4.2.II.D.3.
A typical program involves visual examination of selected assemblies (commonly 5 to 10% of the discharged fuel), concentrating on the lead bundles. Visual examinations normally include, but are not necessarily limited to, crud buildup, rod bowing, and missing components. Additional inspections should be performed depending on the results of operational monitoring including coolant activity and the visual inspections.

4.4.3.6 Thermal and Hydraulic Characteristics Summary Table

The thermal-hydraulic characteristics are provided in Table 4.4-1 for the core and tables of Section 5.4 for other portions of the reactor coolant system.

4.4.4 Evaluation

See Appendix A, Subsection A.4.4.4 of Reference 1.

4.4.5 Testing and Verification

See Appendix A, Subsection A.4.4.5 of Reference 1.

4.4.6 Instrumentation Requirements

See Appendix A, Subsection A.4.4.6 of Reference 1.

4.4.6.1 Loose Parts

To be supplied by Applicant.

4.4.7 References

1. "General Electric Standard Application for Reactor Fuel," (NEDE-24011, latest approved revision).

¶ ~~The Applicant will provide~~ Results of ~~the core~~ a stability analysis. will be provided before the first Applicant references GESSAR II.

¶ The Applicant will include in the plant technical specifications the requirement that core flow will be checked at least once every 24 hours to detect flow reduction.

5.2.3.3.2 Control of Welding

5.2.3.3.2.1 Regulatory Guide 1.50: Control of Preheat Temperature Employed for Welding of Low-Alloy Steel

Regulatory Guide 1.50 delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Sections III and IX.

The use of low-alloy steel is restricted to the reactor pressure vessel. Other ferritic components in the reactor coolant-pressure boundary are fabricated from carbon steel materials.

Preheat temperatures employed for welding of low alloy steel meet or exceed the recommendations of ASME Code Section III, Subsection NA. Components were either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat was maintained until post-weld heat treatment. The minimum preheat and maximum interpass temperatures were specified and monitored.

INSERT 5.2.3.3.2.1

All welds were nondestructively examined by radiographic methods. In addition, a supplemental ultrasonic examination was performed.

~~For commitment and revision number, see Section 1.8.~~

5.2.3.3.2.2 Regulatory Guide 1.34: Control of Electroslag Weld Properties

No electroslag welding was performed on BWR components.

5.2.3.3.2.3 Regulatory Guide 1.71: Welder Qualification for Areas of Limited Accessibility

Qualification for areas of limited accessibility is discussed in Subsection 5.2.3.4.2.3.

INSERT 5.2.3.3.2.1

¶ Acceptance criterion II.3.b.(1)(a) of SRP 5.2.3 for control of preheat temperature requires that minimum and maximum interpass temperatures be specified. While the GESSAR II control of low-hydrogen electrodes to prevent hydrogen cracking (provided in Subsection 5.2.3.3.4) does not explicitly meet this requirement, the GESSAR II control will assure that cracking of components made from low-alloy steels does not occur during fabrication. Further, the GESSAR II control minimizes the possibility of subsequent cracking resulting from hydrogen being retained in the weldment.

5.2.3.4.2.3 Regulatory Guide 1.71: Welder Qualification for Areas of Limited Accessibility (Continued)

All ASME Section III welds were fabricated in accordance with the requirements of Sections III and IX of the ASME Boiler and Pressure Vessel Code. There are few restrictive welds involved in the fabrication of BWR components. Welder qualification for welds with the most restrictive access was accomplished by mockup welding. Mock-ups were examined with radiography or sectioning.

~~For commitment and revision number, see Section 1.8.~~

5.2.3.4.3 Regulatory Guide 1.66: Nondestructive Examination of Tubular Products

For discussion of compliance with Regulatory Guide 1.66, see Subsection 5.2.3.3.3.

5.2.4 Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

This section discusses the inservice inspection and testing program for the NRC Quality Group A components; i.e., ASME Boiler and Pressure Vessel Code Section III, Class 1, components. It will show how the program meets requirements of Section XI of the ASME Code.

5.2.4.1 System Boundary Subject to Inspection

The reactor pressure vessel, system piping, pumps, valves, and components within the reactor coolant pressure boundary defined as quality Group A (ASME Code Section III, Class I) were designed and fabricated to permit full compliance with ASME Code Section XI. (Applicant will provide applicable code and addenda dates.) Access is provided for volumetric examination of pressure

The acceptance criterion II.3.b.(3) of SRP 5.2.3 is based on Regulatory Guide 1.71. GESSAR II meets the intent of this regulatory guide by utilizing the alternate approach given in Section 1.8.71.

6.4 HABITABILITY SYSTEMS (Continued)

Radiation Protection	Section 12.3 and Chapter 15.0
Heating, Ventilating and Air Conditioning (HVAC)	Subsection 9.4.1
Fire Protection	Subsection 9.5.1
Lighting Systems	Subsection 9.5.3
Power Systems	Chapter 8
Radiation Instrumentation and Monitoring	Subsections 7.6.1.2 and 12.3.4, and Section 11.5
Control Room Isolation Instrumentation and Controls	Subsection 7.3.1.1.17

Equipment and systems are discussed in this section only as necessary to describe their connection with control room habitability. References to other sections are made where appropriate.

The term "control building" typically includes the main control room, areas adjacent to the main control room containing plant information and equipment necessary to normal and emergency operations, and kitchen and sanitary facilities. It is also the entire zone service~~s~~d by the control room ventilation system. "Emergency conditions" include such postulated releases as radioactive materials, toxic gases, smoke and steam.

These areas include the rooms which comply with the requirements of SRP Section 6.4 for "Control Room Emergency Zone".

6.7.1.1 Safety Criteria (Continued)

(8) The MSPLCS, including instrumentation and circuits necessary for the functioning of the system, is designed to standards applicable to an engineered safety feature.

(9) The MSPLCS controls include interlocks to prevent inadvertent operation of the system. In particular, interlocks are provided to prevent damage to the MSPLCS, or to the main steam system, due to accidental opening of any system isolation valves when the pressure in the connecting main steam piping exceed MSPLCS operating pressure. All such controls and interlocks are activated from appropriately designed safety systems or circuits.

(10) The MSPLCS is designed to permit testing of the operability of controls and actuating devices during power operation to the extent practical, and complete testing of system function during plant shutdowns.

(11) The MSPLCS is designed so that: (a) thermal stresses and pressures associated with flashing and thermal deformations, under the loading conditions associated with the activated system shall not affect the structural integrity or operability of the main steam system or main steam isolation valves; and (b) any deformation of isolation valve internals shall not induce leakage of the main steamline isolation valve beyond the capacity or capability of the MSPLCS.

(12) Equipment is provided (as part of the MSPLCS) to prevent the release of valve stem packing leakage to the environment from main steam system isolation valves outside the containment.

(13) The design of the MSPLCS complies with the requirements of SRP Section 6.7 and Regulatory Guide 1.96 with the exceptions discussed in Section 1.8.96.

9.5 OTHER AUXILIARY SYSTEMS

9.5.1 Fire Protection System

9.5.1.1 Design Bases

The bases for the design of the fire protection program are presented in detail in Appendix 9A (Fire Hazard Analysis). The program's intent is to provide a "defense-in-depth" design resulting in an adequate balance in:

(1) preventing fires from starting;

(2) quickly detecting and extinguishing fires that occur, thus limiting fire damage; and

(3) designing safety-related systems so that a fire that starts in spite of the fire prevention program and burns out of control for a considerable length of time will not prevent safe shutdown.

In addition, fire protection systems are designed so that their inadvertent operation or occurrence of single failure in any of these systems will not prevent plant safe shutdown.

Possible fires that could affect safety-related systems and significant combustible loadings are presented in Appendix 9A on a room-by-room basis. Fire barriers and fire protection systems are discussed for each safety and nonsafety-related area. Each room is also analyzed for its potential radioactive release due to a postulated fire. Noncombustible or fire-resistive materials having a flame-spread, smoke-evolved and fuel-contributed index of 25 or less are used wherever practicable.

→ INSERT 9.5.1.1

Containment isolation valves and included piping of the Fire Protection System are classified as ASME Section III, Class 2 and

INSERT 9.5.1.1

SRP Acceptance Criterion II.2.a requires adherence to BTP CMEB 9.5-1.

Three-hr. fire rated dampers (required by paragraph C.5.f of BTP CMEB 9.5-1) have not been provided in HVAC ducts in the smoke removal systems which have 3 hr fire rated barriers in the Control bldg., the auxiliary bldg., and in the HVAC ductwork that penetrates the reactor bldg. wall from the auxiliary bldg. and fuel bldg.

Some of these ventilation ducts are shared systems in that they also provide normal ventialation. Other ducts are for smoke venting only. Based on the discussion below the present GESSAR II design should be adequate and should be acceptable to the NRC.

The auxiliary building smoke removal system is shown on Figure 9.4-4 and described in Section 9.4.3.2.1.11. Each set of duct work serves and traverses only fire areas of one safety division. There is a smoke vent intake in each fire area with a remote manually operated fire damper which is normally closed. There is a fusible link from the air operator to the vanes so that the damper will close on high temperature. The fire rating of the dampers is 1½ hours. The duct is heavy gage, welded construction which exceeds the requirements for 3 hour fire rated construction. Hence, the design is considered completely adequate for the service.

One of the design objectives of GESSAR II is to avoid fire dampers in smoke vents, as their automatic closure would render the smoke vent inoperative at the very time it was needed. With two exceptions, smoke vents pass through safety areas only of the same division as the vented area. The two exceptions are the Division 2 cable tunnel vent and the primary containment vent.

The Division 2 cable tunnel located in the corridor of (-)6' 10 elevation of the auxiliary building has a dedicated smoke removal system, which passes through the division 1 area. The duct opening is 2.5 sq. ft. and is designed to with stand a 3-hr. fire.

There is a containment vent and a containment supply. The supply takes air from the auxiliary building roof top intake. The fans are located in a room on the top floor of the auxiliary building. The boundaries of the room have a 3 hour fire rating. The supply duct goes directly into the reactor building from the room. A fire in the room cannot prevent safe or alternate shutdown. There is an inboard and a outboard isolation valve for the duct.

INSERT 9.5.1.1 (continued)

The containment exhause has two inboard (1 manual) isolation valves and one outboard isolation valve. If a fire occurs, either the inboard valves or the outboard valve would be located out of the fire area and could be closed. The valve within the fuel building is located in a room with 2 hour rated walls. The room is directly accessible from the fuel building or the stair tower between the fuel and auxiliary building. All return registers except for the pool sweep are located high in the containmen containment so that bulk mixing, aided by the dome mixing system, would occur before any combustion gases enter the ventilation duct. The containment is more sensitive to bulk air temperature than the ventilation duct. If a fire raised the bulk temperature excessively, containment spray would be initiated to protect the containment at a temperature well below the threshold of damage to the ventilation duct. For these reasons, the current GESSAR II design for the containment ventilation is considered proper and adequate. The exhaust ducting which is schedule 20 welded pipe will be designed with a 3 hour fire rating.

The remaining smoke vents which do not have fire dampers are the two in the control building. Each one of these smoke vents serves and traverses one division. Since it is impossible for these smoke vents to allow the fire in the area of one division to spread to another division, the current GESSAR II design is considered to be adequate and proper.

11.5.3.1 Basis for Monitor Location Selection

Monitor locations are selected to assure that all effluent materials comply with regulatory requirements as covered in Regulatory Guide 1.21, Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Release of Radioactive Effluent from Light Water-Cooled Nuclear Power Plants.

11.5.3.2 Expected Radiation Levels

Expected radiation levels are in the ranges listed in Tables 11.5-2 and 11.5-3.

11.5.3.3 Instrumentation

Radiation monitors used are listed in Table 11.5-1.

Grab samples are analyzed to identify and quantify the specific radionuclides in effluents and wastes. The results from the sample analysis are used to establish relationships between the gross gamma monitor readings and concentrations or release rates of radionuclides in continuous effluent releases.

11.5.3.4 Setpoints

Setpoints are listed in Table 11.5-1.

11.5.4 Process Monitoring and Sampling

11.5.4.1 Implementation of General Design Criterion 60

All potentially significant radioactive discharge paths are equipped with a control system to automatically isolate the

The Applicant will provide isolation valves, dampers, or diversion valves with automatic control features should fail in the closed or safe position. Setpoints for actuation of automatic control features initiating actuation of isolation valves, dampers, or diversion valves should be specified in the plant technical specifications

12.1.2.2 Equipment Design Considerations For ALARA Exposures

12.1.2.2.1 General Design Criteria

engineering design procedures require that the component design engineer consider the applicable regulatory guides as a part of the design criteria. This includes Regulatory Guide 8.8. In this way, the radiation problems of a component or system are considered. A summary survey of the components designs was made to determine the factors considered. The following paragraphs cite some examples of design considerations made to implement ALARA.

12.1.2.2.2 Equipment Design Considerations to Limit Time Spent in Radiation Areas

(1) Equipment is designed to be operated and have its instrumentation and controls in accessible areas both during normal and abnormal operating conditions. Equipment such as the Reactor Water Cleanup (RWCS) System and the Fuel Pool Cleanup (FPCCU) System are remotely operated, including the backwashing and precoat operations. Other equipment has been redesigned in order to lengthen service life. For example, seal water is applied to the recirculation pump seals to keep them clean. This increased the maintenance interval from

No specific instructions have been given to component designers and engineers regarding ALARA design as provided by specific acceptance criterion II.2 of SRP 12.1. However, the

12.2 RADIATION SOURCES

12.2.1 Contained Sources

12.2.1.1 Source Terms

With the exception of the vessel and drywell shields, shielding designs are based on fission product and activation product sources consistent with Section 11.1. For shielding, it is conservative to design for fission product sources at peak values rather than an annual average, even though experience supports a lower annual average than the design average (Reference 1). It should be noted that activation products, principally Nitrogen-16, control shielding calculations in most of the primary system. In areas where fission products are significant, conservative allowance is made for transient decay while at the same time providing for transient increase of the noble gas source, daughter product formation and energy level of emission. Areas where fission products are significant relative to Nitrogen-16 include: (1) the condenser off-gas system downstream of the jet air ejector; (2) liquid and solid radwaste equipment; (3) portions of the RWCS; and (4) portions of the feedwater system downstream of the hotwell including condensate treatment equipment.

For application, the design sources are grouped first by location and then by equipment type (e.g., reactor building, core sources). The following paragraphs represent the source data in various pieces of equipment throughout the plant. General locations of equipment are shown in the general plant arrangement drawings of Section 1.2.

Specific acceptance criterion II.6 of SRP 12.2 provides that in addition to the location of contained sources, their approximate size and shape be shown. Though this has not always been included, the source strength or concentration has been provided in Chapter 12 tables and detailed geometry is provided in Table 12.2-1 for the reactor, and in Chapter 5 for the main steam and recirculation piping.

12.2.1.2.7.1 Radioactive Sources in Main Steam System (Continued)

source is dominated by Nitrogen-16. In components where N-16 has decayed, the other activities carried by the steam become significant. During plant shutdown, there is a residual activity resulting from prior plant operations. These data will be provided by the Applicant.

12.2.1.2.7.2 Radioactive Crud in Piping and Steam Systems

The inside surfaces of the piping and all reactor and power systems components become coated with activated corrosion products, commonly called crud. The quantity of crud on the components is dependent on a number of factors, including power history, water quality and fuel experience. The piping and components carrying reactor water are coated with higher levels of crud than piping and components carrying steam. Figure 12.2-2 shows the data used in the design of this plant to characterize crud accumulation in Recirculation System Piping. Crud levels in steam piping are estimated to be about 1% of those in the recirculation piping. 12.2

INSERT 12.2.1.2.7.2

12.2.1.2.8 Radioactive Sources in the Spent Fuel

The radiation source for spent fuel is given in Subsection 12.2.1.2.1.1.4 (Table 12.2-3) in terms of MeV/sec/W. The design calculation is carried out for a mean element for an appropriate decay time.

12.2.1.2.9 Other Radioactive Sources

12.2.1.2.9.1 Reactor Startup Source

The reactor startup source is shipped to the site in a special cask designed for shielding. The source is transferred under water while in the cask and loaded into beryllium containers. This is then loaded into the reactor while remaining under water. The

INSERT [redacted] 12.2.1.2.7.2

Criterion II.6 of SRP 12.2 provides that the buildup of activated corrosion products in various components and systems should be addressed and allowances made in design source terms should be explained. Based on current data and analysis, activated corrosion products are most significant in the recirculation piping.

15.3.3.2 Sequence of Events and Systems Operations

15.3.3.2.1 Sequence of Events

Table 15.3-5 lists the sequence of events for Figure 15.3-5.

15.3.3.2.1.1 Identification of Operator Actions

The operator should ascertain that the reactor scrams from reactor water level swell. The operator should regain control of reactor water level through RCIC operation or by restart of a feedwater pump, and he should monitor reactor water level and pressure control after shutdown.

15.3.3.2.2 Systems Operation

In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection, and reactor protection systems.

INSERT 15.3.3.2.2

Operation of safe shutdown features, though not included in this simulation, is expected to be utilized in order to maintain adequate water level.

15.3.3.2.3 The Effect of Single Failures and Operator Errors

Single failures in the scram logic originating via the high vessel level (L8) trip are similar to the considerations in Subsection 15.3.1.2.3.2 (see Appendix 15A for further details).

INSERT [redacted] 15.3.3.2.2

Acceptance criterion II.8 of SRP 15.3.3 provides that only safety grade equipment should be used to mitigate the consequences of this event. It also provides that safety functions be accomplished assuming the worst single failure of a safety system active component. The actual simulation used for this event provides a more conservative basis for evaluating system performance for this transient than would result from direct application of this SRP criterion. Justification for this difference is given in Section IE.11. Acceptance criterion II.10 of SRP 15.3-3 also provides that the analysis assume turbine trip and coincident loss of offsite power. Should a coincident loss of offsite power occur, the consequences would be similar to the consequences of the loss of offsite Power Transient described in Subsection 15.2.6; however, this event would be less severe due to the faster reactor flow coastdown and the earlier feedwater pump trips.

15.4.4.3.2 Results (Continued)

before decreasing after the cold water washed out of the loop at about 18 sec. No damage occurs to the fuel barrier and MCPR remains significantly above the safety limit as the reactor settles out at its new steady-state condition. Therefore, this event does not have to reanalyzed for specific core configurations.

INSERT 15.4.4.3.2

15.4.4.4 Barrier Performance

No evaluation of barrier performance is required for this event since no significant pressure increases are incurred during this transient (Figure 15.4-1).

15.4.4.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, since no radioactive material is released from the fuel.

15.4.5 Recirculation Flow Control Failure with Increasing Flow

15.4.5.1 Identification of Causes and Frequency Classification

15.4.5.1.1 Identification of Causes

Failure of the master controller of neutron flux controller can cause an increase in the core coolant flow rate. Failure within a loop's flow controller can also cause an increase in core coolant flow rate.

15.4.5.1.2 Frequency Classification

This transient disturbance is classified as an incident of moderate frequency.

INSERT [redacted] 15.4.4.3.2

Acceptance criterion II.2.(b) of SRP 15.4.4 provides that fuel clad integrity shall be maintained by ensuring that the CPR remains above the MCPR safety limit. Since this event does not result in a significant increase in pressure and it is initiated from a low power condition, no MCPR calculation was performed.

15.4.5.3.1 Input Parameters and Initial Conditions (Continued)

Maximum stroking rate of a single recirculation loop value for a loop controller failure is limited by hydraulics to 30%/sec.

15.4.5.3.2 Results

15.4.5.3.2.1 Fast Opening of One Recirculation Valve

Figure 15.4-2 shows the analysis of a failure where one recirculation loop main valve is opened at its maximum stroking rate of 30%/sec. Table 15.4-4 provides the sequence of events of this failure.

The rapid increase in core flow causes a sharp rise in neutron flux, initiating a reactor scram at approximately 1.3 sec. The peak neutron flux reached was 235% of NBR value, while the accompanying average fuel surface heat flux reaches 73% of NBR at approximately 2.2 sec. MCPR remains considerably above the safety limit and average fuel temperature increases only 108°F.

INSERT 15.4.5.3.2.1

Reactor pressure is discussed in Subsection 15.4.5.4.

15.4.5.3.2.2 Fast Opening of Two Recirculation Valves

Figure 15.4-2 illustrates the failure where both recirculation loop main valves are opened at a maximum stroking rate of 11%/sec. Table 15.4-5 shows the sequence of events for this failure. It is very similar to the above transient. Flux scram occurs at approximately 1.6 sec, peaking at 162% of NB rated, while the average surface heat flux reaches 67% of NB rated at approximately 2.3 sec. MCPR remains considerably above the safety limit and average fuel temperature increases 80°F. Therefore, this event does not have to be reanalyzed for specific core configurations.

INSERT ~~[illegible]~~ 15.4.5.3.2.1

Acceptance criterion II.2.(b) of SRP 15.4.4 provides that fuel clad integrity shall be maintained by ensuring that the CPR remains above the MCPR safety limit. Since this event does not result in a significant increase in pressure and it is initiated from a low power condition, no MCPR calculation was performed.

15.4.7.1.1 Identification of Causes (Continued)

incorrect location or discharged. Third, the misplaced bundles would have to be overlooked during the core verification process performed following core loading.

15.4.7.1.2 Frequency Classification

This unlikely event occurs when a fuel bundle is loaded into the wrong location in the core. It is assumed the bundle is misplaced to the worst possible location, and the plant is operated with the mislocated bundle. This event is categorized as an infrequency incident based on the following data:

Expected Frequency: 0.002 events/operating cycle

The above number is based upon past experience.

15.4.7.2 Sequence of Events and Systems Operation

15.4.7.2.1 Sequence of Events

The postulated sequence of events for the misplaced bundle accident (MBA) is presented in Table 15.4-6.

15.4.7.2.2 Systems Operation

A fuel loading error, undetected by in-core instrumentation following fueling operations, may result in an undetected reduction in thermal margin during power operations. For the analysis reported herein, no credit for detection is taken and, therefore, no corrective operator action or automatic protection system functioning is assumed to occur.

¶ The Applicant will include in the plant operating procedures/Technical Specification provisions for potential fuel loading errors.

15.6.5.5 Radiological Consequences (Continued)

10CFR100 guidelines. This analysis is referred to as the "design basis analysis".

(2) The second is based on assumptions considered to provide a realistic estimate of radiological consequences. This analysis is referred to as the "realistic analysis".

A schematic of the transport pathway is shown in Figure 15.6-2.

Additional parameters and information for specific design basis accidents are provided in Subsection 19.3.15.1. 15.1

15.6.5.5.1 Design Basis Analysis

The methods, assumptions and conditions used to evaluate this accident are in accordance with those guidelines set forth in Regulatory Guides 1.3 and 1.7. The specific models, assumptions and computer code used to evaluate this event based on the above criteria are presented in Reference 2. Specific values of parameters used in this evaluation are presented in Table 15.6-7.

15.6.5.5.1.1 Fission Product Release from Fuel

It is assumed that 100% of the noble gases and 50% of the iodine are released from an equilibrium core operating at a power level of 3651 MWt for 1000 days prior to the accident. While not specifically stated in Regulatory Guide 1.3, the assumed release of 100% of the core noble gas activity and 50% of the iodine activity implies fuel damage approaching melt conditions. Even though this condition is inconsistent with operation of the ECCS system (Section 6.3), it is assumed applicable for the evaluation of this accident. Of this release, 100% of the noble gases and 50% of the iodine become airborne. The remaining 50% of the iodine is removed by plate-out and condensation; therefore, it is not available for airborne release to the environment. The activity airborne in the containment is presented in Table 15.6-8.

For determining equipment leakage contribution to the LOCA dose, it is assumed that the 50% "plateout and condensation" fraction of the released iodine finds its way into the suppression pool water. This is consistent with R.G. 1.3 though not with acceptance criterion II.(2) of SRP 15.6.5 the latter document provides that 50% of the core iodine activity should be assumed to be missed in the sump water being circulated through the containment external piping. The assumptions used in this calculations are the more conservative with respect to BWR post-LOCA total dose calculations. See item 16 of Subsection 19.3.15.1 for a detailed description of equipment leakage contribution to off-site dose.

19.3.3.74 QUESTION/RESPONSE 3.74 (220.33)

QUESTION 3.74

In Section 3.8.3.3.6.3.2 of your FSAR, you indicate that you satisfy three out of the four load combinations presented in Item II.3.c (ii)(a) of Section 3.8.3 of the SRP for the factored load conditions for steel structures using the elastic working stress design method. State why you omitted Equation (4) of Item II.3.c(ii)(a) and verify that you satisfy our position on the load combination represented by Equation (4). (3.8.3)

RESPONSE 3.74

4

Subsection 3.8.3.3.6.3.2 was revised to include the missing Equation 3 of SRP 3.8.3 II.3.c(ii)(a). Equation 4 is more severe than and bounds Equation 3. Equation 4, not Equation 3, is applied to the current design of GESSAR II.

19.3.15 Chapter 15 - Responses

19.3.15.1 QUESTION/RESPONSE 15.1 (440.3)

QUESTION 15.1

Address each item identified in Item 1 of Table 15-4 of Regulatory Guide 1.70, Revision 3, or indicate an interface to provide the information. (15.6.5)

RESPONSE 15.1

1a. Hydrogen Purge Analysis

As noted in Subsection 6.2.5, redundant Class 1E hydrogen recombiners are provided. Even assuming the arbitrary failure of one recombiner, the remaining recombiner is capable of maintaining hydrogen concentrations below the ignitable or detonable level; therefore, there will be no need to purge the containment and there will be no additional dose contribution from this source.

1b. Equipment Leakage Contribution to LOCA Dose

The potential dose contribution from this source is determined in a manner consistent with RG 1.3 and SRP 15.6.5 unless otherwise noted.

(1) Fission Product Source Term. Appendix B to SRP 15.6.5 suggests 50% of the iodine contained in the core at shutdown is released to and contained within the suppression pool. RG 1.3 suggests that 50% of the iodine in the core is released to the containment where 50% remains airborne and 50% is lost due to washout/plateout phenomena. RG 1.7 suggests that 50% of the iodine remains in the core. Since there is no question that the core cannot initially