GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

February 24, 1992

MFN No. 044-92
Project No. 681
EEN-9228

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: R. C. Pierson, Director
Standardization and Non-Power Reactor Project Directorate

Subject: **SBWR Licensing Plan**

Reference: Letter from D.M. Crutchfield to P.W. Marriott, "Submittal of Standard Certification Application for the Simplified Boiling Water Reactor (SBWR) (TAC No. M80718)," dated January 31, 1992

Dear Mr. Pierson:

The General Electric Company (GE), in cooperation with the Department of Energy (DOE), plans to submit its application for a Final Design Approval (FDA) and Design Certification (DC) for the Simplified Boiling Water Reactor (SBWR) in accordance with 10CFR52. As a first step prior to tendering the standard safety analysis report (SSAR) in August, 1992, the subject plan has been prepared to communicate the plans which GE will follow in the FDA and DC application. It is intended to serve as a communication tool between NRC management and GE to help facilitate staff review of the SBWR. GE does not request review of the plan and submits it for information purposes only.

The plan includes the scope of the DC application, the proposed FDA and DC program schedule, the specific manner of incorporating technical issue resolutions achieved by the EPRI Utility Requirements Program, and delineation of approaches for dealing with the requirements of 10CFR52.

The referenced letter requested that GE provide the plan for submittal of the SBWR DC application by February 21, 1992. The SBWR Licensing Plan responds to this request by providing the submittal schedule in Section 3.

GE will make a complete SBWR submittal on August 31, 1992 of all the information required by the Standard Review Plan NUREG-0800. A supplementary submittal in six months, on February 28, 1993, will include the type of information that the staff historically evaluates after assimilating the integral plant design.

D040 /1

This approach is not a modular approach, as was the case for the ABWR DC submittals where all SSAR information was submitted in five groups over a period of fifteen months (i.e., modular) in order to support and coordinate with the EPRI Utility Requirements Program.

With its recent experience (GESSAR and ABWR), GE fully understands the information and schedular requirements of the staff. It is clear that the information identified in Table 3-1, Remaining SSAR Sections to be submitted February 28, 1993, is supportive and meets the staff's review needs.

Sincerely,

P. W. Marriott, Manager
Regulatory and Analysis Services
M/C 382, (408) 925-6948

cc: G. Bockhold (EPRI)
D.M. Crutchfield (NRC)
F.A. Ross (DOE)
J.F. Quirk (GE)

GE Nuclear Energy

SBWR Licensing Plan

Table of Contents

Table of Contents

SBWR Licensing Plan

1. Introduction

The General Electric Company (GE), in cooperation with the Department of Energy (DOE), plans to submit its application for a Final Design Approval (FDA) and Design Certification (DC) for the Simplified Boiling Water Reactor (SBWR) in accordance with 10CFR52. As a first step prior to tendering the Standard Safety Analysis Report (SSAR) in August 1992, a licensing plan has been prepared to communicate (1) the administrative and procedural aspects of the review and (2) issue resolution approaches which GE plans to follow in the SBWR design and license application in order to facilitate its review. This plan is intended to serve as a communication tool between the NRC and GE throughout the SBWR review.

The plan includes the scope of the application for certification, the proposed design certification program schedules, the specific manner of incorporating technical issue resolutions achieved by the ALWR Utility Requirements Program, and delineation of approaches for dealing with the requirements of 10CFR52 consistent with Commission guidance contained in the SECY-90-377 Staff Review Memorandum (SRM) of February 15, 1991.

GE will be seeking a Part 52 review and approval sequence that will result in NRC staff issuance of a FDA followed by design certification. The product of the staff review for FDA is a staff safety evaluation report (SER) on all technical issues associated with an essentially complete design and, if the review is satisfactory, issuance of an FDA. Following issuance of the FDA, the ensuing DC review will be focused on formalating the content of the proposed DC rule and formalizing the results in a notice of proposed rulemaking.

2. Scope Of Application

The scope of the SBWR SSAR comprises an essentially complete standardized nuclear power plant. In the Statements of Consideration accompanying Part 52, an "essentially complete nuclear power plant" is defined as a design which includes all structures, systems and components which can affect safe operation of the plant except for site-specific features such as the service water intake structure and the ultimate heat sink. As such, the scope of the SBWR standard plant design includes the entire nuclear island, the turbine island and the radwaste facility.

The application will conform to all the requirements of 10CFR52.47. The SSAR Table of Contents, constructed from NUREG-0800 Standard Review Plan has been previously reviewed by the NRC.

3. Schedule Information

The following list are the key milestones during the design approval and certification process.

8/20/92	Conduct SSAR review kickoff meeting with the NRC.
8/31/92	Submit SSAR to NRC.
2/28/93	Submit remaining SSAR sections (shown in Table 3 - 1) to NRC.
2/28/93	Submit ITAAC to NRC
9/93	ALWR Passive Plant Utility Requirements Document SER issued by the NRC (SECY-91-161 Date)
4/30/94	Resolve SSAR questions/comments, and open items.
6/94*	NRC issue final SER and FDA.
6/95*	Obtain NRC Design Certification of the SBWR

Table 3 - 1 Remaining SSAR Sections to be submitted February 28, 1993

Section	Title
1.8	Interfaces for Standard Design
1.9	Conformance with Standard Review Plan and applicability of codes and standards
App 1A	Response to TMI related matters
App 1B	Failure modes and effects analysis
App 1C	SBWR Compliance with EPRI Utility Requirements Document
3.4	Water level (flood) design
3.9.6	Inservice testing of pumps and valves
App 3A	Seismic Soil-Structure Interaction Analysis
App 3F	Structural Evaluation
5.2.4[1]	Reactor coolant pressure boundary inservice inspection and testing
6.6	Inservice inspection of class 2 and 3 components
9.5.1[2]	Fire protection system
21.7	Logic diagrams from Chapter 7

* SECY-91-161 dates are 1/95 for FDA and 7/96 for DC, dates will be reconciled with the NRC Staff

1 Includes "Detection and Sizing Capability Test for Regulatory Guide 1.150," Appendix 5A

2 Includes "Fire Hazard Analysis," Appendix 9A

4. Level Of Detail Required

In accordance with the Commission's Staff Review Memorandum (SRM) dated February 15, 1991, the SSAR information submitted for the SBWR will:

1. Reflect a design which, for all structures, systems or components that can affect safe operation of the plant, is complete, except to the extent that some further adjustment to the design within established design envelopes may be necessary to accommodate actual, as-procured hardware characteristics.

2. Encompass a depth of detail no less than that in an FSAR at the operating stage for a recently licensed plant, except for site-specific, as-procured, and as-built information.

3. Be sufficient to allow staff to evaluate the resolution of severe accident issues in the design, as well as to incorporate the experience from operating events in current designs which are to be prevented.

4. Provide a sufficient level of detail to ascertain how the risk insights from the design-specific Probabilistic Risk Assessment (PRA) are addressed in the design.

The SBWR SSAR will have a level of design detail that satisfies the above guidance and will be comparable to the information contained in the ABWR SSAR.

5. ALWR Utility Requirements Document (URD)

An assessment of the SBWR design for compliance with the URD Volume III will be provided in Chapter 1 of the SSAR. This assessment will include the areas of the URD that are discussed in the SSAR.

6. Testing to Support the SBWR Design

The SBWR is firmly rooted in prior BWR experience. New technology selected for SBWR application is proven, in the sense that performance of innovative safety features will have been confirmed through analysis, appropriate test programs, experience or a combination thereof prior to submittal of the SSAR for NRC review. The safety features of the SBWR are logical extensions of existing BWR technology, developed with over 30 years of design, licensing and operating experience from over 100 BWRs in service worldwide.

These features are listed in Table 6-1. For most of the SBWR design, compliance with NRC requirements will be demonstrated in the same manner as for current large commercial BWR power plants, including ABWR.

The following discussions summarize the presentation given to the NRC staff on January 24, 1992. The additional information provided during this presentation was originally requested by the NRC in a letter dated November 6, 1991 from V.M. McCree to P.W. Marriott, *Preliminary Evaluation and Request for Information on the Simplified Boiling Water Reactor Testing Program.*

6.1 Natural Circulation

The Dodewaard reactor in The Netherlands has been successfully operating for 24 years. The SBWR is physically similar to this plant and will operate in a similar manner. There are no current unresolved operational problems with this plant. Additionally, forced circulation reactors have been tested with natural circulation core flow. The only concern in this area is the stability of the reactor during these conditions. The SBWR power operating conditions are different (lower bundle power and higher bundle flow) from those of a forced circulation operating plant operating with natural circulation core flow. This prevents any instabilities form occurring at steady-state power conditions. Scram setpoints for low level and high neutron flux prevent transient instabilities from occurring. Also, in the unlikely event of oscillations, instrumentation displays and alarms will be available to the operator so that manual action can be taken. The second issue concerns stability during startup and low power operation. The conditions for instability can be prevented from occurring by enforcing operating procedures that ensure conditions to avoid instability. The operating procedures will be validated through the use of analytical methods qualified against plant data. (See item 7). Therefore, no testing is required to address these issue.

6.2 Use of Isolation Condensers

Several operating plants (Dresden 2 & 3, Millstone 1, Nine Mile Point 1, Oyster Creek)have Isolation Condenser Systems. The SBWR design is similar to these systems. Therefore no further testing is required other than confirmatory test listed in Section 6.4.

6.3 Low Pressure inventory Control with Depressurization

Automatic Depressurization System (ADS) - In addition to the use of conventional S/RVs for vessel depressurization, a comprehensive evaluation was performed to select a diverse valve type. It was concluded that a squib valve actuated by a propellant is the optimum approach for the SBWR. Sample cartridges containing the proposed propellant material were subjected to the expected SBWR environmental conditions: irradiation, accelerated thermal aging and a LOCA steam test. After aging, over 70 boosters were test fired successfully proving that radiation, temperature, and steam will have a negligible effect on booster performance. A full scale valve was constructed and actuated four times under actual pressures and flow rates. The valve was also tested to qualify it for the environmental conditions that it is expected to operate under. These tests provide the basis for classifying the valve as ASME safety class 1, seismic category I, IEEE Class 1E.. The results of the DPV tests are detailed in a report prepared for the Department of Energy (Reference 1).

Blowdown Simulation - The blowdown behavior of the SBWR is not unlike that of operating plants or the ABWR. Previous blowdown testing programs for operating plants and the ABWR are directly applicable. Therefore, no further testing is required.

Gravity Driven Cooling System (GDCS) - GE designed and built the GDCS Integrated Systems Test (GIST) facility. The GIST facility is a full vertical height section-scale of an SBWR plant, simulating the reactor pressure vessel with an electrically heated core, drywell, wetwell, elevated pool and all significant flow paths. At this facility, GE has run a series of low pressure blowdown tests simulating real time water and steam loss-of-coolant accidents (LOCAs) and non-LOCA events. These tests are the first and only tests for an Advanced Light Water Reactor design which have been conducted to show the total plant response to a LOCA and to the subsequent behavior of a gravity-driven water make-up system. The results of the GDCS tests are detailed in a report prepared for the Department of Energy (Reference 2). The results show that a simple passive GDCS system can be used to prevent core uncovery for any design basis accident at an SBWR and to provide sufficient cooling water to supply long-term cooling.

6.4 Long Term Containment Performance

GE studied three basic containment cooling systems. Thermal-hydraulic analyses have been performed for the containment configurations and the associated Passive Containment Cooling System (PCCS) to demonstrate their long-term (> 3 days) decay heat removal capability. The three concepts were then evaluated on the basis of the established criteria of event capability, plant safety, economics, licensing and plant layout considerations. The study concluded that a modular condenser is a viable PCCS component.

The testing of the PCC condenser is divided into four areas;

1) Basic heat transfer data (MIT/Berkeley)

2) 1D integral system testing (JAPC, GE, Hitachi, Toshiba)

3) Full scale component system testing (SIET)

4) 3D integral system testing (PSI)

Basic heat transfer data (MIT) - Steam-air condensation experiments were performed in a plenum geometry and a heat transfer correlation was developed that extends previously published data. Additionally steam-air experiments in a cooled-tube geometry are being performed to develop a heat transfer correlation.

Basic heat transfer data (Berkeley) - Steam-air condensation experiments were performed in a tube geometry under natural circulation conditions. On-going experiments under natural and forced circulation will provide additional input to the heat transfer correlations.

1D integral system testing (JAPC, Toshiba, GE, Hitachi) - This program confirmed the 1) PCC heat transfer characteristics, 2) the nitrogen purge mechanism, 3) the integrated PCCS performance for post-LOCA containment cooling, and 4) provided data for the qualification of analytical models. The integrated test facility is full-scale in the vertical direction and 1:20 in the horizontal direction.

Full scale component system testing (SIET) - This test will confirm the primary and secondary side thermal-hydraulic performance of the PCC condenser. Additionally, it will confirm the adequacy of the mechanical design of the heat exchanger hardware.

3D integral system testing (PSI) - This confirmatory test will emphasize the parallel channel effects of the condenser. Additional information to be obtained includes the effect of non-condensible gas composition and the suppression pool mixing and stratification characteristics. This test is confirmatory because previous tests (MIT/Berkeley/Toshiba) have already determined the condensation heat transfer characteristics of single and multiple vertical tubes in the SBWR containment configuration.

6.5 Standby Liquid Control System (SLCS)

An extensive testing program has been carried out to support the ABWR. The boron mixing test facilities used two models. The first is a 1/6 scale 3D model of a BWR/5 vessel. The second is a 1/6 scale 3D model of the ABWR vessel. Various injection points were studied such as a standpipe, jet pump Δp line, high pressure core spray (HPCS) spargers, and the reactor internal pump (ABWR) suction line. Natural and forced circulation were both tested. These tests showed that a primary flow path exists between the upper plenum and active channels (bypass flow) and that boron injected through the HPCS ring spargers into the bypass region was distributed uniformly. These results justify the injection location for the SBWR (into the same region as the HPCS ring sparger).

6.6 Fine Motion Control Rod Drive (FMCRD)

The SBWR uses the Fine Motion Control Rod Drive (FMCRD) design used in the ABWR. This drive has been used in European BWRs and has undergone testing at the LaSalle plant.

Table 6 - 1 Key Safety Features of the SBWR Design

Feature	Tests and Experience to support the feature
Natural Circulation	Natural (Dodewaard) and Forced Circulation operating BWRs
Use of isolation condensers	Operating plants (20 years), Full scale component testing
Low Pressure Inventory Control with Depressurization	
Automatic depressurization system (ADS)	■ Depressurization Valve Development Test Program ■ S/RVs identical to operating plants
Blowdown simulation	Operating plants (20 years) and BWR - LOCA programs
Gravity Driven Cooling System (GDCS)	Full Height, volume scaled GDCS Integrated System Test (GIST)
Long Term Containment Performance	■ Basic heat transfer data (MIT and Berkeley) ■ 1D integral system testing (JAPC, Toshiba, GE, Hitachi) ■ Full scale component system testing (SIET) ■ 3D integral system testing (PSI)
Standby Liquid Control System (SLCS)	Boron mixing test part of the ABWR program
Fine Motion Control Rod Drive (FMCRD)	■ Operating European BWRs ■ ABWR Test Program ■ In-plant demonstration test

7. Analytical Methods

The best-estimate computer code TRACG has been modified to realistically predict the SBWR response to accidents. Specific SBWR components and phenomena will be tested and the data used to supplement the TRACG qualification base. TRACG will be used to analyze pressurization transients and Loss of Coolant Accidents (LOCAs). FABLE and TRACG will be used for stability evaluations. Tests used for TRACG qualification are listed in Tables 7 - 1 through 7 – 4.

In addition, previously approved codes such as TGBLA, PANACEA, and ISCOR will be used in the design analysis. TGBLA is an infinite lattice physics code used for determining detailed nuclear characteristics of fuel designs. PANACEA is a three dimensional model of the core used for bundle arrangement. ISCOR is a thermal-hydraulic model of the core used to determine input parameters for PANACEA.

The Probabilistic Risk Assessment (PRA) will follow the methods described in the EPRI ALWR URD Chapter 1, Appendix A, *Key Assumptions and Groundrules.*

Table 7 - 1 TRACG Qualification: Separate Effects Test

Effect	Test Facility
Void fraction	OF36 OF64, CISE, PSTF, Christensen, Horizontal flow, Large H_D pipes and tanks
Heat transfer	THTF, CSHT, GOTA
CCFL	CSHT
Critical flow	PSTF, Edward, Marviken
Pressure drop	ATLAS, FRIGG, ISCOR
Level swell	PSTF
Critical Power	ATLAS, ISCOR
Stability	FRIGG, Two bundle loop
Kinetics	PANAC, Process computer, GEBSCRAM
Boron	Vallecitos 1/6 scale

Table 7 - 2 TRACG Qualification: Component Tests

Component	Test Facility
Pump	Semiscale
Jet pump	INEL 1/6 scale, 5-nozzle, 1-nozzle
Separator	Moss Landing
Upper plenum	16° sector, Horizontal test Facility, SSTF
Isolation condenser	Toshiba, SIET
Geysering	Japanese tests

Table 7 - 3 TRACG Qualification: System Effects Tests

Facility
TLTA
FIST
FIX
GIST
TBL
ROSA-III
ESTA
SSTF
Toshiba isolation condenser test facility
PANDA (PSI)

Table 7 - 4 TRACG Qualification: Plant Data

Plant Test
Peach Bottom turbine trip tests
KKM turbine trip test
Hatch two pump trip test, isolation test
Cofrentes start up tests
LaSalle core wide oscillations
Leibstadt regional oscillations
Caorso regional oscillations
Vermont Yankee stability data
Cofrentes regional oscillations
Forsmark stability data
Dodewaard

8. Inspections, Tests, Analyses, And Acceptance Criteria (ITAAC)

The ITAAC for SBWR will closely follow the approach that is being taken for the ABWR with regards to both format and content. In keeping with the staff's desire that ITAAC be reviewed by the same reviewers who conduct the safety review leading to FDA issuance, the ITAAC for SBWR will be submitted in a time frame that will permit its review in parallel with the SSAR. This will allow the necessary coordination between the ITAAC and the safety review.

The ITAAC will be submitted on February 28, 1993.

9. Severe Accident Mitigation Design Alternatives (SAMDA)

According to 10CFR Parts 51 and 52, a National Environmental Policy Act (NEPA) environmental impact statement (EIS) is not required for a DC. However, the Limerick court of appeals decision required a NEPA analysis of Severe Accident Mitigation Design Alternatives at the operating license stage. This means the COL applicant is now required to address all SAMDA matters including those that pertain to the certified design. This could result in reopening the review of the certified design at the COL stage in order to obtain closure of SAMDA-related matters. In keeping with the purpose of DC, it is desirable to address design specific SAMDA issues at the DC stage, eliminating from consideration such matters at the COL stage.

Following this approach, the design certification rulemaking should contain a Commission determination that the SAMDA findings for the certified design shall be used for SAMDA assessment for any combined license which references the subject design. The SAMDA findings for the certified design shall also be usable to confirm that SAMDA considerations in the early site permit are adequately addressed.

NEPA treatment in issuance of a DC for the SBWR is expected to closely follow NEPA treatment in issuance of a DC for the ABWR, which is currently being developed.

10. Technical Issues Central To The GE SBWR Design

Key technical issues that greatly affect the SBWR design are listed below. Timely approval of the SBWR Design Basis approach to these issues is necessary in order to limit the number of design modifications which may be required during subsequent reviews. In most cases, the SSAR will submitted with the listed design basis.

10.1 Regulatory Treatment of Passive Safety Systems

10.1.1 Passive Safety Systems

EPRI ALWR URD Reference: Chapter 5, Section 1.2.1 Passive Safety Systems
Chapter 5, Sections 2.2.1, 2.2.5, 2.2.12, 3.1.2

Issue: The use of passive safety-related systems is a new concept in LWR designs. Reliance on these systems without operator action (for 72 hours), with simple operator action (> 72 hours), and the subsequent use of non-safety-related active systems is a new combination of systems.

SBWR Design Basis: Passive safety-related systems are used to meet all relevant regulations without the need for operator action for 72 hours. A passive system is one that does not require AC power such as the Gravity Driven Cooling System (GDCS), the Passive Containment Cooling System (PCCS), and the Isolation Condenser System (ICS). These systems are expected to be more reliable than their active counterparts that exist in current plants because they contain fewer components.

In addition to the passive systems, the SBWR design includes active non-safety-related AC powered systems such as the Control Rod Drive System, the Fuel and Auxiliary Pool Cooling System (F & APCS) and the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System that are designed for high reliability.

The combination of the reliable, passive safety-related and the active non-safety-related systems is expected to produce higher overall reliability than the current operating plant safety-related active systems.

Effect on Design: If active systems are required to be safety-related, Class 1E AC power is needed necessitating an enlargement of the electrical building to accommodate the divisional diesel generators. Additionally, active systems such as the Reactor Component Cooling Water System and Reactor Service Water System will have to be upgraded in function. This will result in higher fabrication and construction costs and create a serious economic disadvantage for the SBWR.

Importance: Acceptance of this concept is vital to the SBWR design. If this approach is not accepted, the SBWR will be at an economic disadvantage because of its excessive fabrication and construction costs.

10.1.2 Regulatory Treatment of Non-Safety Systems

Issue: The concern is the regulatory treatment (if any) of non-safety systems that may have been safety-related previously. Traditionally, non-safety systems have been credited in certain special evaluations such as Probabilistic Risk Assessment, Severe Accident Mitigation, and Station Blackout Emergency Response. For passive designs it is proposed that non-safety systems be handled in a manner identical to current plants, whereby all relevant regulations are met with safety systems and appropriate credit for non-safety systems is taken in special evaluations.

SBWR Design Basis: Reducing requirements for active systems is a fundamental goal of the SBWR. Non safety-related active systems can be powered from two non-safety-related diesel generators in the event of loss of offsite power. In the event of an accident, these systems can be used to control the accident or to supplement the passive safety-related systems. They will be subjected to an augmented reliability program. The use of these non safety-related systems is essential to overall plant reliability. Credit for the non-safety systems in the PRA will significantly reduce overall core damage risk by reducing challenges to safety systems.

Effect on Design: Recent (August 16, 1991 Request for Additional Information to EPRI) NRC requirement requests for non-safety systems have included :

- Redundancy to meet the single failure criteria
- Divisional separation
- Seismic qualification

Requirements such as these are greatly different from the current treatment of non-safety-related systems. The design impact is the same as 10.1.1.

Importance: Acceptance of this concept is vital to the SBWR design. If this approach is not accepted, the SBWR will be at an economic disadvantage because of its excessive fabrication and construction costs.

10.2 Severe Accident

10.2.1 Containment Performance

EPRI ALWR URD Reference:
Chapter 5, Section 6.6.2 Containment Performance
Chapter 5, Sections 6.6.2.1 - 6.6.2.5
Chapter 5, Section 7 BWR Mitigation/Containment Requirements
Section 7.2 - 7.27 Performance Criteria

Issue: Deterministic criteria performance criteria will be submitted as an alternative to Conditional Containment Failure Probabilities (CCFP).

SBWR Design Basis: Systems such as the Isolation Condenser System, Gravity Driven Cooling System, and Passive Containment Cooling System provide a reliable and rugged containment system that limits the magnitude and likelihood of specific severe accident challenges. See item 10.2.2 for additional features.

This approach not only assures that the containment will perform its function of limiting offsite doses, but also provides high assurances that uncertainties and known challenges are addressed by specific plant features. The application of CCFP is not technically justifiable for a plant that has been specifically designed to accommodate severe accidents. This is because the core damage frequency is so low that the CCFP is no longer an effective measure of containment performance.

Effect on Design: Changes to the design to produce a low CCFP would increase the complexity of the plant while doing little to improve the containment performance for accidents with a probability of occurrence greater than 10^{-6}. The SBWR has been designed to specifically address severe accidents and application of the CCFP will require costly system additions with little improvement in accident coping capability.

Importance: Acceptance of this concept is vital to the SBWR design. If this approach is not accepted, the SBWR will be at an economic disadvantage because of its excessive fabrication and construction costs.

10.2.2 Core Debris Coolability

EPRI ALWR URD Reference: Cavity Sizing to Promote Long-Term Debris Coolability
Chapter 5, Section 6.6.3.2.1
Chapter 5, Section 6.6.3.2.2

Issue: During a severe accident, core debris must be adequately cooled and contained to insure the integrity of the containment and limit radioactive releases.

SBWR Design Basis: Specific features have been provided in the SBWR to address severe accidents these include:

- Lower drywell flooder with fusible link
- Use of basaltic concrete to reduce non-condensible gas generation
- Concrete barrier (1 meter) between the lower drywell and the containment boundary
- Use of splash shields to reduce impingement of core debris on the lower drywell walls

The first two features have been accepted by NRC for application to the ABWR. The last two features provide additional assurance that core debris will be contained and cooled. Therefore, no additional features are required for the SBWR.

Effect on Design: Additional requirements could delay the design finalization and increase program costs. The second two features provide additional assurance that core debris will be contained and cooled. The magnitude depends on the additional plant modifications required.

Comment: The SSAR will be submitted as described above.

10.2.3 Seismic Hazard

Issue: NRC accepts two different methodologies for the evaluation of seismic hazard of current plants: the seismic PRA approach and the Seismic Margins Approach (SMA). If a seismic PRA approach is used, there are two sets of seismic hazard curves available, the EPRI Seismic Owners Group (SOG) hazard curves and the LLNL seismic hazard curves.

SBWR Application: If a seismic PRA is performed, GE will use the SOG hazard curves until the resolution of the differences between SOG hazard curves and the LLNL hazard curves is accomplished. Alternatively, the SBWR may use the SMA methodology to evaluate seismic plant capability beyond the SSE level. The analysis choice will be made in late 1992.

Effect on Design: Some eastern sites may not meet EPRI core damage frequency goals (10^{-6}) if the LLNL hazard curves are used.

Comment: The SSAR will be submitted as described above.

10.2.4 Containment Vent Penetration

EPRI ALWR URD Reference: Chapter 1, Appendix B, Section 2.5.4.1 Dedicated Containment Vent Penetration - ALWR Position
Chapter 1, Appendix B, Section 2.5.4.3 Assessment

Issue: A dedicated containment vent is not needed for containment overpressure control. A vent if provided, only addresses low probability events (cumulative probability < 10^{-6}/year).

SBWR Design Basis: The SBWR containment is specifically designed to accommodate severe accidents (> 10^{-6} probability) without overpressure protection. This eliminates the necessity for a vent.

Effect on Design: Provisions for the containment vent require a minor rearrangement of the reactor building or redesign of the Atmospheric Control System depending on the type of overpressure protection device required.

Comment: The SSAR will be submitted as described above.

10.2.5 Hydrogen Control

EPRI ALWR URD Reference: Chapter 1, Appendix B, Section 2.5.3.1 Hydrogen Control - ALWR Position
Chapter 5, Section 2.4.2.6

Issue: Long term hydrogen control is needed to reduce the hydrogen produced by radiolysis.

SBWR Design Basis: Containment inerting is a passive means of insuring hydrogen combustion control in the short term. In addition, DC powered igniters are included for long term hydrogen control.

Effect on Design: If igniters are not an acceptable method of hydrogen control, then recombiners or another system possibly requiring Class 1E AC power would be necessary. The safety-related AC power system will require enlargement of the electrical building as well an upgrading of the electrical systems to Class 1E standards. See Items 10.1.1 and 10.1.2.

Comment: The SSAR will be submitted as described above.

10.3 Source Term Treatment for the Passive Plant

10.3.1 Use of a Physically-Based Source Term

EPRI ALWR URD Reference: Chapter 1, Appendix B, Section 2.5.2.1, Source Term Treatment for the Passive ALWR
Chapter 5, 2.4.1 Source Term Definition
Chapter 5, 1.2.3 ALWR Treatment of Source Term Issues
Chapter 1, Appendix B, Section 2.5.2, Source Term Treatment for the Passive ALWR

Issue: Historically, TID 14844 has been used as a source term for design basis accidents. As an alternative, a physically-based source term can be used to provide a more rational design basis. Application of the physically-based source term also necessitates using appropriate removal coefficients to accurately model the radionuclide transport.

SBWR Design Basis: The SBWR will use a physically based source term that takes advantage of the features of the containment. Additionally, appropriate removal coefficients are used for radionuclide transport.

Effect on Design: A TID 14844 approach will require a safety-related air conditioning system for the control room and standby gas treatment system along with safety-related AC power. The safety-related AC power will require enlargement of the electrical building as well an upgrading of the electrical systems to Class 1E standards. See Items 10.1.1 and 10.1.2.

Comment: The SSAR will be submitted as described above.

10.3.2 Control Room Habitability

EPRI ALWR URD Reference: Chapter 5, Section 1.2.1.3 Control Room Habitability
Chapter 5, Section 6.4.5.1 Control Room

Issue: Heat loads and radiation levels during design basis accidents in the main control room have been greatly reduced. This, together with more realistic radiation levels make a safety-related HVAC system unnecessary.

SBWR Design Basis: The SBWR does not need a safety-related, filtered control room HVAC system to maintain acceptable temperature or radiation levels in the control room. A safety-related pressurized air system (bottled air) will maintain a positive pressure in the main control room to minimize in-leakage and maintain adequate oxygen levels for breathing with sufficient capacity for a 72 hour duration. This is possible due to the modernizing of the control room equipment and the use of a physically-based source term with appropriate removal coefficients.

Effect on Design: If a safety-related HVAC system is required, a safety-related Class 1E AC power source would be required. The safety-related AC power will require enlargement of the electrical building as well an upgrading of the electrical systems to Class 1E standards. See Items 10.1.1 and 10.1.2.

Comment: The SSAR will be submitted as described above.

10.3.3 Radionuclide Attenuation

EPRI ALWR URD Reference:
Chapter 5, Appendix B, Section 3.4 In-containment Fission Product Behavior
Chapter 5, Section 1.2.3.1 Charcoal Filters
Secondary Building Fission Product Holdup and Removal
Chapter 5, Section 1.2.3.7
Chapter 5, Section 6.4.3.1
Chapter 5, Section 6.3.4.5
Chapter 5, Sections 6.4.3 and 6.4.4

Issue: Containing and reducing leakage is an approach that meets off-site dose limits without requiring Standby Gas Treatment System (SGTS) or safety-related containment sprays. This also allows simplified emergency planning. This issue is closely tied to the Source Term issue 10.3.1.

SBWR Design Basis: The SBWR reactor building design incorporates multiple barriers which combine to produce a passive system that results in extremely low leakage of fission products to the atmosphere. The design uses retention, plateout, holdup and decay of fission products inside the reactor building.

Effect on Design: Addition of a SGTS and the associated systems requires a major enlargement of the reactor building.

Comment: The SSAR will be submitted as described above.

10.4 Emergency Planning

10.4.1 Off-site Emergency Planning (simplification)

EPRI ALWR URD Reference:
Chapter 1, Appendix B, 2.1.4.1
Chapter 5, 1.2.5 Off-site Emergency Planning

Issue: High assurance of containment integrity together with reduced offsite doses from accidents allow simplified emergency planning. Simplified means eliminating early notification of the public, planning for evacuation of the public, and provisions for exercising the offsite plan. This issue is closely tied to the Source Term issue 10.3.1.

SBWR Design Basis: The SBWR is designed to assure that containment functions are performed and to minimize containment penetrations and leakage pathways to insure low off-site doses in the event of a severe accident. The offsite doses due to core melting are very low (1 rem at the site boundary) and the probability of containment integrity is very high.

Effect on Design: If simplified emergency planning is not allowed, the design will not need to be changed. However, conventional emergency planning complicates plant licensing and requires periodic exercises throughout the plant's lifetime thus increasing its operating costs. Also, experience has shown that emergency planning is difficult because of the necessity of public acceptance.

Comment: The SSAR will be submitted as described above.

10.4.2 Design Basis Duration for Safety Systems

EPRI ALWR URD Reference:
Chapter 1, Section 2.3.3.2
Chapter 1, Section 2.2.3.3
Chapter 5, 2.2.10 Overall Requirements
Chapter 5, 2.4.2.8 Mitigation

Issue: For design basis accidents, current plant designs require operator action in less than an hour and off-site support in 7 to 30 days to supplement the active safety-related systems. The SBWR does not require any operator action or use of active systems (using AC power) for 72 hours. After 72 hours, only simple operator actions are required to maintain the passive safety-related systems effectiveness.

SBWR Design Basis: After an accident, recovery actions will begin immediately. The recovery actions will utilize non-safety-related systems (F&APCS, RWCU/SDC) to supplement the safety-related systems (GDCS, PCCS, ICS). This combination of systems provides a higher reliability for recovery than the current operating plant safety-related active systems only. Additionally, provisions will be provided to install or use temporary systems to provide cooling in the event of active or passive system unavailability.

Without active non-safety-related assistance, the simple operator actions required after 72 hours include:

- Replenishing the Spent Fuel Pool water to prevent uncovering the fuel assemblies
- Refilling the Isolation Condenser System and Passive Containment Cooling System Pools

Effect on Design: If passive safety-related systems are required to function without simple operator action for more than 72 hours, pool size will have to be increased. This will require major enlargement of the reactor building.

Comment: The SSAR will be submitted as described above.

10.5 Emergency Shutdown

EPRI ALWR URD Reference: Chapter 1, Appendix B Section 2.5.6.3 Safe Shutdown Assessment
Chapter 5, Section 3.3.2 PDHRS Requirements

Issue: For accidents and transients, all reactor pressure vessel and containment design limits are met using safety-related passive systems. Regulatory Guide 1.139 requires a temperature of 212°F to be reached during an emergency shutdown within 36 hours, using safety-related equipment only. Passive safety-related heat removal necessitates a large ΔT to remove heat. Therefore, a slightly higher temperature is achieved when the plant is shut down during an emergency with safety-related systems only.

SBWR Design Basis: The SBWR has highly reliable decay heat removal which includes the use of safety-related passive and non-safety-related active systems The combination of the passive safety-related and the active non-safety related systems is expected to produce higher reliability than the current operating plant safety-related active systems. The passive systems which rely on a high ΔT are the ICS and PCCS.

Effect on Design: Meeting the letter of Regulatory Guide 1.139 will require that the non-safety-related active RWCU/SDC system be switched to safety-related. This costly requirement will require safety-related Class 1E AC power and an enlargement of the electrical building to accommodate the divisional diesel generators. See items 10.1.1 and 10.1.2.

Comment: The SSAR will be submitted as described above.

10.6 Seismic Engineering

10.6.1 Elimination of OBE

EPRI ALWR URD Reference: Chapter 1, Section 4.5.2.4.1
Chapter 1, Sections 4.5.2.4, 4.6.1, 4.7.2, 4.7.3, 4.8.1.1
Chapter 1, Appendix B, Section 2.1.1
Chapter 1, Tables 1.2-3, 1.2-6, 1.4-4, 1.4-5, 1.4-6, 1.4-7, 1.4-8

Issue: A separate analysis for the OBE will not be performed.

SBWR Design Basis: The SSE is the design basis for the SBWR. A separate OBE analysis will not be performed for certification purposes. Non-safety-related equipment will analyzed to meet Uniform Building Code acceptance criteria.

Effect on Design: If OBE results are still required, a reanalysis will have to be produced from the SSE results.

Comment: The SSAR will be submitted as described above.

10.6.2 Dynamic Analysis Methods and Design Standards

EPRI ALWR URD Reference: Chapter 1, Section 4.7.3.1
Chapter 1, Section 4.7.3.2, 4.7.3.3, 4.7.3.9, 4.7.3.13

Issue: There are a number of code cases and standards that have not been accepted by NRC. The use of these code cases and standards will reduce analysis efforts and reduce costs without compromising the design when the SBWR detailed piping design is performed.

SBWR Design Basis: The code cases and standards that will have the most impact are:

- Code case N-411 (damping) for use with time history analysis and independent support motion response spectra method
- Code Cases ASME N-451 and N-462
- Use of NCIG-14
- Use of AISC N-690 in lieu of ASME NF supports for linear component supports

Effect on Design: There is no immediate impact on the design because the piping analysis has not been performed yet. However, if the code cases and standards in question are approved, the savings in manpower and plant costs may exceed 10 million dollars.

Needed Resolution Date: Prior to FDA.

10.7 Advanced Digital Control Systems

EPRI ALWR URD Reference: Chapter 10, Section 2.2.13 ALWR M-MIS Policy Statement, Regulatory Stabilization
Chapter 10, Section 6.2.6.2 Isolation Devices
Chapter 10, Section 5.2.1.1 Data System Structure
Chapter 10, 2.2.1

Issue: There is a concern about common cause failures due to lack of diversity (in software, microprocessors, etc). Shielding from electromagnetic interference must also be considered. Additionally, SECY-91-272 expresses concern that the advanced control room in the passive plant raises new questions about the operator's role and interface with the passive and active systems.

SBWR Design Basis: The hardware will resemble the ABWR control room currently being reviewed by the NRC. Included in the design will be cathode-ray tube (CRT) displays, touch-screen displays, flat panel displays and a wide panel display. Four separate essential multiplexing networks using fiber optic cables will be used for safety-related systems. A standardized set of microprocessor-based instrument modules is used to implement SBWR monitoring and control function. These standardized modules include self-diagnostics, automatic calibration, user interactive front panels, and a standardized man-machine interface.

Effect on Design: This issue addresses a fundamental design basis for the SBWR. A major control system and possibly reactor building redesign is required to reduce the degree of multiplexing or reliance on microprocessors.

Comment: The SSAR will be submitted as described above.

10.8 Leak-Before-Break (LBB) for Sub-compartment Design

EPRI ALWR URD Reference: Chapter 1, Section 4.4.3.3.1 Regulatory Positions
Chapter 1, Section 4.5.5.1.3 Rupture of Piping
Chapter 5, Section 7.2.4 BWR Containment Performance Criteria

Issue: The NRC is hesitant to approve methodology and desires to approve applications on a case-by-case basis. GE supports methodology approval to lessen the risk of redesign late in the design process.

SBWR Design Basis: Leak-before-break methodology will be used for subcompartment design. Criteria for application of LBB to pipe rupture analyses are presented in ANSI/ANS 58.2 and NUREG-1061. These criteria include:

- Demonstrating that the materials, configuration, and service loads insure that crack growth is very unlikely.

- Demonstrating that any flaws permitted by ASME Section XI and related in-service inspections (ISI) would not grow significantly during service.
- Demonstrating that a flaw of the size resulting in leakage, which would be assured of detection using installed instrumentation, would be stable by a significant margin when subjected to an algebraic combination of service and SSE loads.

Effect on Design: If the SBWR application of leak-before-break is not approved, significant construction delays will occur because the piping will need to be redesigned to meet additional requirements.

Comment: The SSAR will be submitted as described above.

10.9 Tornado Design

EPRI ALWR URD Reference: Chapter 1, Appendix B, Sec''on 2.1.3.1 ALWR Position
Chapter 1, Table 1.2-6 En.elope of ALWR Plant Site Design Parameters (Wind and Tornado only)

Issue: GE had previously used ANSI/ANS 2.3 1983 as a reference for tornado design. This standard uses more realistic design goals than Reg. Guide 1.76 or its interim document. Subsequent discussions between GE and NRC concerning the ABWR have produced a consensus between these two standards.

SBWR Design Basis: The design will be based upon a tornado wind speed of 300 mph and its associated components of rotational and translational speeds, pressure drop, etc.

Effect on Design: There will be no impact on design as long as the site specific tornado parameters (corresponding to median values with a probability of exceedance of 10^{-6} /year) are bounded by the design basis tornado parameters.

Comment: The SSAR will be submitted as described above.

10.10 Industry Codes and Standards

EPRI ALWR URD Reference: Chapter 1, Section 4.4
Chapter 1, Sections 4.4.1 - 4.4.3,
Table 1.4-1, Table 1.4-2, Table 1.4-3

Issue: The SRPs reference codes and standards that have been revised or superceded. Some standards have been changed to be less restrictive than the previous standard. Additionally, the NRC is determining how to cope with revisions to codes and standards after the design is certified.

SBWR Design Basis: The SBWR is designed to the codes and standards listed in Tables 1.4-1, 2, and 3 of the ALWR URD.

Effect on Design: Any standards not accepted by the NRC prior to certification could require a redesign of the SBWR if the approved standard is more restrictive.

Comment: The SSAR will be submitted as described above.

10.11 In-Service Testing of Pumps and Valves

EPRI ALWR URD Reference: Chapter 1, Section 1.2.4.3.1
Chapter 1, Section 8.6.1 In-service Inspection (ISI) Features
Chapter 1, Section 12.2.7 Valve In-service Testing
Chapter 1, Section 12.4.3

SBWR Design Basis: In-service testing of ASME Code class 1, 2, and 3 pumps and valves and safety-related valves will be performed in accordance with Section XI of the ASME Code and applicable Addenda as required by 10CFR50, Section 50.55a(g).

Details of the inservice testing program, including test schedules and frequencies will be reported in the inservice inspection and testing plan. This plan will include baseline pre-service testing to support the periodic in-service testing of the components as required by the technical specifications. The plan will also include the proposed frequency of commitment to disassemble and inspect the pumps, check valves, and motor operated valves (MOVs) within the Code and safety-related classification stated above.

Comment: The SSAR will be submitted as described above.

11. References

[1] P.F. Billig, et al, *Simplified Boiling Water Reactor (SBWR) Program Depressurization Valve Development Test Program - Final Report*, GE Nuclear Energy, GEFR-009879, October 1990.

[2] P.F. Billig, *Simplified Boiling Water Reactor (SBWR) Program Gravity-Driven Cooling System (GDCS) Integrated Systems Test - Final Report*, GE Nuclear Energy, GEFR-00850, October 1989.