TECHNICAL ASPECTS OF ADVANCED LIGHT WATER REACTOR EMERGENCY PLANNING

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on behalf of the

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EXECUTIVE SUMMARY AND CONCLUSIONS

Since 1985, U.S. utilities have been working, through the Advanced Light Water Reactor (ALWR) Program, to develop a technical foundation for a new generation of nuclear power plants. The new plant designs are building on the extensive experience base of existing LWRs in the U.S. and around the world, and will improve upon these existing plant designs in many important respects. One aspect of potential improvement is in the area of emergency planning. This report is intended to establish a thorough and solid technical basis, for use by industry and NRC decision makers, in considering updated emergency planning for ALWRs.

This report provides an integrated treatment of the factors to be considered in developing an updated technical basis for emergency planning for the ALWR. These factors include the technical reasons to update emergency planning for the ALWR, the Utility Requirements Document (URD) emergency planning design criteria and methodology, and the ability of the passive plant designs to meet the design criteria.

The focus of ALWR Program emergency planning work to date is the passive plant. For this reason, the report addresses the Passive ALWR. In general, however, the technical basis for emergency planning, as outlined in the report, could apply to any ALWR standard plant design. On that basis, the conclusions in the report should not be considered as being limited to passive plants, since they could be adopted for Evolutionary ALWRs as well.

Technical Reasons to Update ALWR Emergency Planning

The primary reasons for updating the technical basis for ALWR emergency planning are the significant advances in severe accident technology and in plant design capability over the past 15 years. The emerging ALWR designs have superior core damage prevention and severe accident mitigation capability, and the current technical understanding of severe accident risk is greatly improved compared to that available when the existing emergency planning requirements were established in the late 1970s. Therefore it is appropriate and timely to update the ALWR emergency planning technical basis to ensure that it reflects technical reality for ALWRs.

Design Criteria and Methodology

Technical design criteria and associated methodology have been defined for ALWR emergency planning in the areas of containment performance and offsite dose. The complete set of criteria and methodology are specified in Volume III, Chapter 1 of the URD and may be summarized as follows:

Containment Performance Criterion and Methodology

Plant design characteristics and features shall be provided to preclude core damage sequences which could bypass containment and to withstand core damage sequence loads. Containment loads representing those associated with low pressure core damage sequences shall not exceed ASME Service Level C/Unity Factored Load limits. Accident sequences will be shown not to result in loads exceeding those limits for approximately 24 hours; beyond approximately 24 hours, there shall be no uncontrolled release.

The methodology for demonstrating the containment performance criterion includes incorporating design characteristics and features specified in the URD to address severe accident challenges, and use of best estimate evaluations of loads associated with core damage sequences.

Dose Criterion and Methodology

Dose at 0.5 mile from the reactor from a physically-based source term shall not exceed 1 rem for approximately 24 hours.

The methodology for demonstrating the dose criterion includes the use of median dose (i.e., median meteorology) and use of effective dose equivalent with a 50 year commitment.

The criteria and methodology are intended to be applied together and are primarily deterministic. For a particular ALWR design, it is intended that the criteria and methodology eventually be demonstrated as part of design certification. A supplemental PRA evaluation (10^{-5} core damage frequency and 10^{-6}, 1 rem at 0.5 mile) is also required by the URD in support of the two criteria. As part of the PRA evaluation, it is also to be demonstrated that the prompt accident quantitative health objective of the NRC Safety Goal Policy is met with no credit for offsite evacuation prior to 24 hours. This reliance on deterministic criteria with PRA as a supplement is consistent with the NRC Severe Accident Policy.

Passive ALWR Design Conformance

Using Standard Safety Analysis Report information, an evaluation was made of the two passive plant designs being developed for design certification against the above URD criteria in order to establish that there will in fact be actual standard design certification applicants which have the capability to pursue ALWR emergency planning. The assessment indicates that both designs, the AP600 and the SBWR, will be able to meet the emergency planning criteria with margin.

Conclusions

The overall conclusion from the work performed to date on the technical aspects of ALWR emergency planning is that the likelihood and consequences of a severe accident for an ALWR are fundamentally different from that assumed in the technical basis for existing emergency planning requirements 15 years ago. Specific conclusions are as follows:

> *The updated emergency planning technical basis should be utilized for the ALWR.* The primary reason for this is that the ALWR plant design capability, along with the greatly improved technical understanding of severe accident risk which has evolved over the last 15 years, result in significantly reduced ALWR radiological risk.
>
> *A strong technical basis for updated emergency planning exists in the URD.* A set of deterministic criteria in the areas of severe accident containment performance and offsite dose, supplemented by PRA goals, have been developed for ALWR emergency planning and included in Volume III of the URD. For standard plant designs which demonstrate that these criteria are met, even in the extremely unlikely event of a severe accident the containment has been designed to maintain integrity and thus any radioactivity release will be very slow and small. A period of approximately 24 hours or more exists before reaching offsite dose levels at which the U.S. EPA recommends that actions be taken to protect members of the public.
>
> *ALWR designs have excellent potential to meet the design criteria.* A preliminary assessment of AP600 and SBWR conformance with the ALWR emergency planning design criteria has been performed and indicates that the designs will meet the criteria. The Plant Designers have committed to provide demonstrations as part of design certification that their respective designs meet the criteria.

Section 1.0

INTRODUCTION AND BACKGROUND

1.1 PURPOSE AND SCOPE

Since 1985, U.S. utilities have been working, through the Advanced Light Water Reactor (ALWR) Program, to develop a technical foundation for a new generation of nuclear power plants. The new plant designs are building on the extensive experience base of existing LWRs in the U.S. and around the world, and will improve upon these existing plant designs in many important respects. One aspect of potential improvement is in the area of emergency planning. This report is intended to establish a thorough and solid technical basis, for use by industry and NRC decision makers, in considering updated emergency planning for ALWRs.

The objective of the report, "Technical Aspects of Advanced Light Water Reactor Emergency Planning," is to provide an integrated treatment of the factors to be considered in developing an updated technical basis for emergency planning for the ALWR. These factors include the reasons to update the technical basis of ALWR emergency planning, the ALWR Utility Requirements Document (URD) emergency planning design criteria, and the ability of the passive plant designs to meet the design criteria. The report supports the conclusion that the likelihood and consequences of a severe accident for an ALWR are fundamentally different than that which is the basis for existing emergency planning requirements.

1.2 APPLICABILITY

The focus of this report is the Passive ALWR. For that reason, Volume III of the ALWR URD[1] specifies emergency planning design criteria for the Passive ALWR. In general, however, the technical basis for emergency planning, as outlined in the following sections, could apply to any ALWR standard plant design. On that basis, the conclusions herein should not be considered as being limited to passive plants, since they could be adopted for Evolutionary ALWRs as well.

1.3 TECHNICAL REASONS TO UPDATE ALWR EMERGENCY PLANNING

The primary reason for updating the technical basis for ALWR emergency planning is that, as discussed in this document, the likelihood and consequences of a severe accident for an ALWR are fundamentally different from that assumed in the basis for existing emergency planning requirements. The emerging ALWR designs have superior core damage prevention and severe accident mitigation capability, and the current technical understanding of severe accident risk is greatly improved compared to that available when the existing emergency planning requirements were established nearly 15 years ago. Therefore it is appropriate and timely to update the ALWR emergency planning technical basis to ensure that it reflects technical reality for ALWRs. This is discussed further below.

1.3.1 Greatly Improved Severe Accident Technology

Existing emergency planning requirements are based on the understanding of severe accidents which was available in the mid to late 1970s. The technical basis for existing emergency planning is primarily contained in NUREG 0396/EPA-520/1-78-016[2] published in December, 1978 which in turn utilized severe accident sequence evaluations from WASH 1400[3], the 1975 Reactor Safety Study which was the first comprehensive probabilistic risk assessment (PRA). The key NRC emergency planning implementation guidance document is NUREG 0654[4] which is a joint NRC and Federal Emergency Management Agency (FEMA) report published in November, 1980, shortly after the Three Mile Island Unit 2 (TMI-2) accident.

Since the time of NRC's promulgation of the emergency planning guidance, a great deal has been learned about severe accident phenomenology and how LWRs respond to severe accidents. In parallel with promulgating the emergency planning guidance, NRC, DOE, various industry organizations, and a number of research organizations worldwide initiated extensive research programs to investigate severe accidents and plant response under severe accident conditions. A number of these research programs have been completed in recent years, with major advances in understanding of severe accident phenomena. This work has significantly increased the capability to predict LWR severe accident effects, and supports the ability of LWRs to withstand severe accidents to a much greater extent than believed in the 1960s and 1970s.

In August, 1985, the NRC Severe Accident Policy[5] was issued. The policy concluded that generic changes to address severe accidents in existing reactors were not warranted, that individual plant examinations should be conducted to look for site or design specific risks that did warrant attention, and that the design of future reactors should address severe accidents as an integral part of the design process.

In addition, the NRC developed new PRA tools, culminating in the issuance of NUREG 1150[6], the 1989 update and replacement of WASH 1400. The technical groundwork of NUREG 1150 together with more recent experimental data and analyses is providing the basis for the ongoing NRC effort to update the design basis source term for ALWRs.

A comparison of the 1975 WASH 1400 study results (on which NUREG 0396 was based) with NUREG 1150 shows that the accident frequencies and source terms for current plants were originally overstated by one to two orders of magnitude[7]. It is also recognized by the authors of the study that the WASH 1400 source term was quite conservative[8]. As a result, the risk posed by nuclear plants, even of conventional design, is now understood to be much less than these very conservative values which were thought possible when today's emergency planning requirements were formulated. While this report does not address the technical aspects of emergency planning for current plants, it is appropriate to incorporate an updated technical basis into emergency planning requirements for the next generation of plants in order to avoid perpetuating this overstatement of the technical factors of risk.

1.3.2 Superior ALWR Design Capabilities

All of the above severe accident experience is being brought to bear on the ALWR design. The NRC Severe Accident Policy statement that future reactors address severe accidents as an integral part of the design process is being implemented by the ALWR designers, resulting in a high degree of severe accident protection, including both core damage prevention and accident mitigation. Highly effective core damage prevention is a central objective of the ALWR design process and has resulted in design features such as increased margin to core safety limits, use of state-of-the-art man-machine interface systems (MMIS) which greatly simplify the job of the plant operator, greatly decreased dependence on operator action after an accident, and, for the passive plants, safety systems which do not require ac power and service water. Based on ALWR design requirements and plant specific PRAs, ALWR core damage frequencies are expected to be well below 10^{-5} per year.

Accident mitigation features have also been heavily emphasized in the ALWR design to provide high assurance of containment integrity and low offsite dose even in the highly unlikely event of a severe accident. Key accident mitigation provisions include a strong containment with significant margin for severe accident loads, features to prevent containment bypass, and extremely reliable containment heat removal. As a supporting requirement for updated emergency planning, the URD specifies a mean frequency of less than 10^{-6} per year for 1 rem dose at 0.5 miles from the reactor, and preliminary assessments indicate that the requirement will be met, with margin.

In addition to the plant designer efforts to incorporate severe accident experience, as part of the ALWR regulatory review process the NRC is developing severe accident requirements. These requirements are being implemented through a number of policy papers and the safety evaluation reports for the standard plant designs. Thus, through the plant designer effort to address severe accidents proactively as part of the design process together with subsequent regulatory review, ALWRs are achieving an unprecedented level of assured severe accident performance capability.

Summarizing, the assured severe accident performance capability of ALWR designs is fundamentally different from the limited capability which was assumed in promulgating the existing emergency planning requirements. The key differences involve greatly improved core damage prevention, design features to preclude early containment failure, the adoption of a newly validated source term methodology, and the regulatory assurance of containment performance during severe accidents. These elements combine to provide an extremely low likelihood of core damage, and effective mitigation of potential releases even if core damage should occur, greatly reducing the need for offsite protective action. Thus, it is reasonable and prudent to reflect this design capability in the emergency planning requirements for the ALWR.

Section 2.0

ALWR UTILITY REQUIREMENTS - THE TECHNICAL FOUNDATION FOR ALWR EMERGENCY PLANNING

The URD sets policy, principles, and specific design requirements to produce ALWR designs which are reliable, economical, and very safe. With respect to severe accident mitigation (and therefore emergency planning) the URD establishes specific criteria, and associated methodology for demonstrating that the criteria have been met, in the areas of containment performance and offsite dose. In addition, a supplemental PRA evaluation is required by the URD in support of the demonstration of the criteria. Together, these form the technical foundation for emergency planning for the ALWR.

2.1 ALWR DESIGN PHILOSOPHY AND REQUIREMENTS FOR CORE DAMAGE PREVENTION

The URD provides for a comprehensive and balanced approach to safety. Highest priority is assigned to the prevention of core damage accidents, both through measures to ensure high accident resistance (e.g., through reduction in safety system challenges) and excellent safety systems to prevent initiating events from progressing to the point of core damage. Excellent mitigation capability is also incorporated in ALWR designs as a defense-in-depth measure to reduce even further the likelihood and consequences of serious accidents.

While the emergency planning requirements focus on containment and accident mitigation capability, it is noted that highly effective core damage prevention is key to overall plant safety and for that reason forms an important part of the technical foundation for ALWR emergency planning. Core damage prevention of the ALWR is rooted in the URD emphasis on simplicity, engineering margin, and human factors throughout the design process. Examples of requirements in these areas include:

- Natural circulation decay heat removal from the core
- No recirculation pumps or piping in the BWR
- Canned rotor pumps, thus eliminating pump seal loss of coolant accident (LOCA), in the PWR
- No loop seals and a minimal number of welds in PWR primary system piping

- Increased thermal margin in the fuel (15% above regulatory limits)
- PWR primary system hot leg temperature of 600°F or less to reduce steam generator tube corrosion
- Improved resistance to embrittlement in the reactor vessel
- Increased reactor coolant system (RCS) coolant inventory which delays core uncovery in the event of an accident
- Decreased dependence on operator action after an accident
- Improved control room which makes the plant easier and safer to operate
- Improved accessibility for maintenance
- Decreased dependence of safety systems on support systems

In addition, there are requirements specifically directed toward avoiding core uncovery during shutdown conditions. The ALWR Program reviewed existing shutdown risk issues and the Volume III URD provisions to address these issues [9]. Additional requirements were defined as a result of this review. With proper plant specific implementation of these requirements and appropriate administrative controls and procedures provided by the Plant Owner and operator, core uncovery during shutdown conditions will not be a credible event.

Finally, accident management requirements exist to prevent as well as limit the extent of core damage. Equipment and procedures for accident management are being considered as part of the plant design process, thus increasing the likelihood of successful recovery actions.

In summary, while the remainder of this report focuses on containment and accident mitigation matters, the ALWR emphasis on core damage prevention and the resulting extremely low probability of an accident are important factors in the consideration of emergency planning requirements.

2.2 ALWR EMERGENCY PLANNING DESIGN CRITERIA AND METHODOLOGY

Technical design criteria and associated methodology have been defined for ALWR emergency planning in the areas of containment performance and offsite dose. The complete set of criteria and methodology are specified in Volume III, Chapter 1 of the URD[1] and are reproduced in Appendix A of the report.

2.2.1 Summary of Criteria and Methodology

A summary of the criteria and the associated methodology is as follows:

Containment Performance Criterion

Plant design characteristics and features shall be provided to preclude core damage sequences which could bypass containment and to withstand core damage sequence loads. Containment loads representing those associated with low pressure core damage sequences shall not exceed ASME Service Level C/Unity Factored Load limits. Accident sequences will be shown not to result in loads exceeding those limits for approximately 24 hours; beyond approximately 24 hours, there shall be no uncontrolled release.

The methodology for demonstrating the containment performance criterion includes the following:

- Incorporate the design characteristics and features specified in the URD to address severe accident challenges.

- Demonstrate using best estimate severe accident methods that the loads associated with core damage sequences are no more limiting than the peak LOCA plus hydrogen loads .

- Protection of the containment for overpressurization beyond 24 hours shall be provided. Overpressure protection may be provided by the size and strength of the containment. On the order of two to three days is judged to be adequate time for actions by the plant staff to bring the accident under control.

Dose Criterion

Dose at 0.5 mile from the reactor* from a physically-based source term shall not exceed 1 rem for approximately 24 hours.

* It is intended that the dose criterion be stated as 1 rem at 0.5 mile from the reactor (vs. 1 rem at the site boundary as stated in reference [1].) This will be corrected in the next revision to reference [1].

The methodology for demonstrating the dose criterion includes the use of a probabilistic dose method (e.g., CRAC2 or MACCS), use of median dose (i.e., median meteorology), and use of effective dose equivalent with a 50 year commitment.

The criteria and methodology are primarily deterministic and, for each specific ALWR design, are eventually intended to be reflected in design certification. A supplemental PRA evaluation is also required by the URD in support of the two criteria. This reliance on deterministic criteria with PRA as a supplement is consistent with the NRC Severe Accident Policy[5]. The supporting requirements for the containment performance criterion, the dose criterion, and the supplemental PRA evaluation are described in more detail below in Sections 2.3, 2.4, and 2.5, respectively.

2.2.2 Integral Nature of Criteria and Methodology

The ALWR emergency planning design criteria are intended to be applied together with the methodology specified in the URD. Thus, for example, it would be inappropriate to require plants to meet 1 rem at 0.5 mile with a dose evaluation methodology which is more conservative than that in Volume III, Chapter 1, Section 2.6.5. Application of the criteria with the specified methodology is considered to provide adequate margin based on the following:

- The bounding nature of the core damage progression and associated radioactive release specified in the URD methodology, given any credible severe accident.

- The very low likelihood of any severe accident in an ALWR. Given this extremely low likelihood, conservatism beyond that noted above is considered unwarranted.

- The margin in the 1 rem, 24 hour dose requirement. The 1 rem is at the lower end of the U.S. Environmental Protection Agency (EPA) range of 1 to 5 rem for evacuation[10], and 24 hours provides significant margin to perform offsite protective measures.

Additional detail is provided in Sections 2.3 and 2.4 below.

2.3 CONTAINMENT PERFORMANCE REQUIREMENTS

The licensing design basis for the ALWR containment is the traditional set of deterministic loads and load combinations compared against ASME Section III limits. Loads associated with events including loss of coolant accidents and the safe shutdown earthquake are combined in the design of the plant. Further, the licensing design basis includes loads associated with generation of hydrogen in accordance with 10CFR50.34(f)[11].

In addition to the licensing design basis, the URD includes the safety margin basis which contains requirements that provide margin beyond the licensing design basis. The safety margin basis specifies severe accident requirements in support of the emergency planning containment performance design criterion defined above. These requirements have been developed from a deterministic perspective. A probabilistic perspective has also been applied to provide added confidence in the completeness of the deterministic requirements and to make use of the significant body of PRA information. Each of these perspectives is discussed below.

2.3.1 Deterministic Perspective for Severe Accident Requirements

The severe accident requirements in support of the containment performance design criterion were developed in two steps. In the first step, a set of design characteristics and features was defined to address severe accident containment challenges. A comprehensive set of potential severe accident challenges was identified based on systematic consideration of past PRAs, operating experience, severe accident research, and unique design aspects of the ALWR. Table 2-1 contains a list of these potential challenges. There are 23 challenges in the table. The first 13 challenges represent events which could occur independent of or precede core damage, such as bypass accidents. The remaining 10 challenges could occur as a result of a severe accident, such as containment pressure loads from a core damage event.

A systematic evaluation of the URD was performed to assess the degree to which each of the 23 potential challenges was addressed in the requirements [12]. This systematic evaluation contains a challenge by challenge assessment of the requirements for both the passive PWR and the passive BWR. Appendix B provides a summary of the design characteristics and features specified in the URD to address each challenge. It is concluded from this systematic evaluation that potential challenges, regardless of the extremely low likelihood of the challenge, have been systematically and explicitly addressed in the URD.

Table 2-1

Potential Severe Accident Containment Challenges

CHALLENGES/FAILURE MODES THAT ARE INDEPENDENT OF OR COINCIDENT WITH A SEVERE ACCIDENT

1. Containment Isolation
2. Interfacing System LOCA
3. Blowdown Forces
4. Pipe Whip and Jet Impingement
5. Steam Generator Tube Rupture (PWR)
6. Anticipated Transient Without Scram (ATWS)
7. Suppression Pool Bypass (BWR)
8. Reactor Pressure Vessel (RPV) Failure
9. Internal Vacuum
10. Internal (Plant) Missiles
11. Tornado and Tornado Missiles
12. Man-Made Site Proximity Hazards
13. Seismic

CHALLENGES/FAILURE MODES POTENTIALLY RESULTING FROM A SEVERE ACCIDENT

14. High Pressure Melt Ejection (HPME)
15. Hydrogen Detonation/Deflagration
16. In-vessel Debris-Water Interaction
17. Ex-vessel Debris-Water Interaction
18. Noncondensable Gas Generation During Core-Concrete Interaction
19. Containment Basemat Erosion or Reactor Pressure Vessel Support Degradation During Core-Concrete Interaction
20. Core Debris in Containment Sump
21. Core Debris Contact with Containment Shell Liner
22. Decay Heat Generation
23. Steam Generator Tube Rupture (SGTR) from Natural Circulation of Hot Gases (PWR)

In the second step, the results of this systematic evaluation were applied to establish the types of severe accident sequences for which containment response should be evaluated against the Service Level C/Unity Factored Load limits as specified in the containment performance design criterion. This is necessary since a number of accident sequence types are potentially precluded or otherwise impacted by design.

On the basis of existing plant PRAs, generically applicable severe accident research results, and preliminary passive plant design information, assessments indicated that for the first group of 13 challenges (i.e., containment bypass type challenges), as well as for high pressure melt ejection, hydrogen detonation, steam explosion, basemat erosion or pressure vessel support degradation, core debris contact with shell liner, and steam generator tube rupture from hot gases, the severe accident requirements will provide high assurance of containment integrity [12,13]. This set of challenges includes those which could pose an early threat to containment integrity. The assessments considered the engineered capabilities of the containment systems, i.e., utilize proven technology, function in the environment which the systems will experience, perform functions reliably (e.g., incorporate redundancy or passivity), avoid the need for rapid or complex operator actions, minimize dependence on support systems, and be sufficiently independent from the systems whose failure could lead to core damage in the first place so as to avoid significant vulnerability to common cause failure.

An additional factor relative to containment challenges is that, even if it was assumed that containment systems do not perform as designed, the plant operators have the ability to perform accident management actions to assure containment integrity. An example in this regard is containment isolation. Accident management procedures have been developed and implemented to address containment isolation as follows[14,15]:

- Confirmation of containment isolation. In the event of a containment isolation signal, emergency operations and/or alarm response procedures call for the operator to confirm that containment isolation valves have closed using valve position indications in the control room. For the ALWR, on the order of hours are expected to be available for the operator to perform any necessary valve closures before significant release of radioactivity into the containment.

- Continuous survey of radiation in key plant areas, providing indication of the existence and location of non-isolated or leaking lines. Monitoring systems have been designed for areas such as building ventilation stacks, sampling lines, and sumps such that if excessive leakage begins to occur, it can be detected immediately.

- In case of leakage, complementary confirmation of containment isolation including local verifications and/or operator actions when necessary.

Generally, it is considered that a relatively small, well-trained team of plant personnel can be effective in accident management for containment isolation as well as other containment challenges. As noted in Section 2.1 above, the ALWR URD specifies that accident management equipment and procedures be developed as part of the design process.

On the basis that challenges which pose an early threat to containment integrity are being addressed by well-engineered containment systems, and considering the extremely low likelihood of core damage in the first place as well as the capability of accident management to address problems, accident sequences involving early containment failure are not considered credible in ALWRs.

The remaining challenges (i.e., hydrogen plus LOCA loads, pressurization from debris-water interactions, the potential for core concrete interaction, and decay heat loads) should be considered in establishing the accident sequences for which containment response should be evaluated. In considering these remaining challenges, the effect of plant design characteristics and features on the containment loads should be included. For example, passive containment heat removal does not depend upon any electrical or mechanical equipment which must function in a severe accident environment. Thus it is reasonable to assume that passive containment heat removal functions as designed during the accident.

Thus, on the basis of the deterministic perspective, the types of severe accident sequences for which containment response should be evaluated against the Service Level C/Unity Factored Load limits are as follows:

Core Damage

- Rapid core damage progression, i.e., beginning at approximately one hour after the initiating event, and occurring over a time frame of a few hours
- Large scale core melt and associated gas and aerosol release
- Steam release out of phase with aerosol release

- Consideration of in-vessel core damage and the potential for ex-vessel core damage

Reactor Coolant System Condition

- Limited aerosol plateout in the RCS
- A vapor pathway exists in the RCS (i.e., from the core to the containment atmosphere)
- RCS is depressurized to about 100 psig or less

Containment Condition

- Containment is isolated and otherwise intact at the time of core damage (i.e., no containment bypass has occurred)
- Water exists in the reactor cavity/lower drywell prior to or immediately upon reactor vessel lower head penetration
- Containment systems are functioning as designed (heat removal, fission product removal, hydrogen control, pH control)
- Containment leaking at design basis leak rate (or leak rate proportional to pressure)

Secondary Building Condition

- Containment leakage released into secondary building volume
- Building volume mixing and exchange with the environment is based upon plant design characteristics (e.g., safety envelope leakage)
- Building volume bypass pathways taken into account

As noted in Appendix A, the above severe accident sequence types are specified in URD Chapter 5, Section 2.6, Criteria and Methodology for ALWR Emergency Planning. The loads associated with these severe accident characteristics must not exceed specified ASME limits for approximately 24 hours.

ASME Service Level C/Unity Factored Load limits were specified in order to provide high confidence that containment leakage would, at most, be a linear extrapolation of design basis leakage. This is based on several factors including:

- Service Level C assures stress levels below yield in steel containments, and unity factored load assures limits on linear deformation in concrete containments; leaks are not expected in membranes with such small deformations.
- A review of experimental and analytical evidence [16] which indicates that there is essentially no increase in penetration leakage under severe accident conditions up to Service Level C/Unity Factored Loads.

- Nuclear plant containment leak test data indicating that, for pressure increases up to design pressure, leak rate does not exceed a value proportional to the pressure [16].

An additional point is the fact that the fission product mass is almost exclusively particulate[17] and as noted in reference [17], aerosol plugging of leak paths is expected which should significantly reduce the actual mass leaked during an accident compared to that assumed in design basis leakage.

The 24 hour limit is consistent with the 1 rem, 24-hour limit specified in the dose criterion and allows appropriate time for ad-hoc public protective actions.

No uncontrolled release beyond 24 hours has been specified to provide additional margin for emergency planning. While approximately 24 hours is considered more than adequate for ad hoc evacuation, it is desirable to avoid long-term overpressure failure.

2.3.2 Probabilistic Perspective for Severe Accident Requirements

PRA has been applied to confirm that the appropriate severe accident sequence characteristics are being considered in the evaluation of containment response against the Service level C/Unity Factored Load limits. From a probabilistic perspective, the URD requires that functional sequence types with frequency greater than approximately 10^{-7} per year be evaluated for containment response. Lower frequency functional sequence types are to be reported for discussion (i.e., identification of design characteristics and features which are credited in reaching this low frequency), but are not required to be evaluated for containment response. This 10^{-7} per year frequency threshold for sequence types to be evaluated for containment response is consistent with the NUREG 1420[18] limit for insignificant risks and with previous regulatory guidance (e.g., Standard Review Plan guidance to evaluate potential accidents from hazards in the vicinity of the plant site which exceed approximately 10^{-7} per year.) Also, consideration of functional sequence types greater than approximately 10^{-7} per year helps provide assurance that the cumulative effects of such sequence types will not exceed the 10^{-6} per year probability goal for offsite consequences.

As described in Section 3 below, review of the passive plant designs indicates that accident sequences which are of the order of 10^{-7} per year or greater involve core damage into an intact containment with the reactor coolant system at least partially depressurized and containment systems functioning as designed. That is, the characteristics of these sequences from the PRA are similar to the characteristics defined from a deterministic perspective.

2.3.3 ALWR Performance for Accidents Comprising Existing Emergency Planning Basis

Given the above ALWR design requirements, it is useful at this point to examine the accident types and failure modes which dominated the risk in the existing emergency planning basis and the manner in which these sequence types and failure modes are addressed by the ALWR design. At the time of the development of the existing emergency planning basis, defined in NUREG 0396[2], WASH 1400[3] provided the most detailed perspective on the types of accident scenarios which made up the collection of "Class 9" events. Accident scenario types and containment failure modes which dominated the risk in WASH 1400 are summarized in Table 2-2, and it is these events which formed the basis for existing emergency planning requirements. Also included in Table 2-2 are important challenges identified as a result of PRA work subsequent to WASH 1400. More recently, improved understanding of severe accident behavior as well as modifications to plants and procedures have changed the characteristics of accident scenarios which dominate risk compared to WASH 1400. This applies to a significant extent in existing plants and to an even greater extent in ALWRs. ALWR design requirements directly address those events which dominated the risk in WASH 1400 and subsequent PRAs. Appendix C describes the Passive ALWR design characteristics and features that have been provided to preclude or accommodate the accident sequence types and failure modes listed in Table 2-2 as contributors to core damage and containment failure.

It is apparent from this comparison that the Table 2-2 WASH 1400 issues which dominated the risk and formed the basis for existing emergency planning, as well as subsequently identified containment challenges (shown in Table 2-2 with a footnote), have been addressed explicitly in the ALWR requirements. Therefore, the characterization of risk for ALWRs will differ significantly from a WASH 1400 type characterization, or even from the characterization in subsequent PRAs. Table 2-3 provides clear illustration of this difference in risk characterization. It is the ALWR risk characterization, which reflects the above design characteristics and features and the improved phenomenological understanding of severe accidents, that should be used in formulating ALWR emergency planning regulatory requirements.

Table 2-2

Accident Sequence Types Which Tend to Dominate Risk for Existing Emergency Planning Basis

DOMINANT ACCIDENT SEQUENCES LEADING TO CORE DAMAGE*	
PWRs	**BWRs**
LOCAs (large or small) • Loss of injection (AD, SD) • Loss of recirculation (AH, SH)	LOCAs (large or small) • Loss of injection (AE, [illegible]E)
Vessel Rupture (R)	Vessel Rupture ([illegible])
Interfacing LOCA (V)	
Transients • Loss of secondary heat removal (TML) • Station blackout (TMLB')	Transients • Loss of containment heat rem[illegible] (T[illegible]V) • Loss of all injection (TQUV)
ATWS (TKQ)	ATWS (TC)
Shutdown Conditions**	Shutdown Conditions **

POTENTIAL CONTAINMENT FAILURE MODES*	
PWRs	**BWRs**
Overpressure (δ) In-Vessel Steam Explosion (α) Hydrogen Combustion (δ) Containment Isolation (δ,ε) Basemat Penetration (ε) Direct Containment Heating** Steam Generator Tube Rupture**	Overpressure (∂) In-vessel Steam Explosion (α) Containment Isolation (β) Liner Melt-Through** Ex-Vessel Steam Explosion** Overtemperature**

Notes:

* Characters in parentheses are sequence and failure mode designators from WASH 1400

** Issues which were identified in PRA work subsequent to WASH 1400.

Table 2-3

Comparison Between WASH-1400 and ALWR Requirements

	Mean Core Damage Frequency	Mean Frequency of Exceeding 1 Rem	Mean Frequency of Exceeding Prompt Effects Dose
WASH-1400 (doses at 10 miles from reactor)	$\sim 1.5 \times 10^{-4}$ /yr	$\sim 4 \times 10^{-5}$ /yr (1)	$\sim 4 \times 10^{-6}$ /yr (2)
ALWR Requirements (doses at 0.5 miles from reactor)	$<10^{-5}$ /yr	$<10^{-6}$ /yr	(3)
Passive Plant (doses at 0.5 miles from reactor)	$\sim 10^{-6}$ /yr (4)	$<10^{-7}$ /yr (4)	$<10^{-8}$ /yr (4)

Notes:

(1) Based on mean core damage frequency of $\sim 1.5 \times 10^{-4}$/yr (i.e., 3 x the WASH-1400[3] median value of 5×10^{-5}) and, from Figure I-11 of NUREG-0396[2], ~0.3 conditional probability of exceeding 1 rem at 10 miles.

(2) Based on mean core damage frequency of $\sim 1.5 \times 10^{-4}$/yr and, from Figure I-11 of NUREG-0396, ~0.03 conditional probability of exceeding prompt effects dose at 10 miles.

(3) Functional sequence types which could threaten containment must be less than $\sim 10^{-7}$/yr.

(4) Preliminary estimates based on initial AP600 and SBWR PRA work.

2.4 OFFSITE DOSE REQUIREMENTS

As part of the technical foundation for emergency planning in ALWRs, an offsite dose limit is required. A maximum dose of 1 rem at 0.5 mile from the reactor for a period of approximately 24 hours after the beginning of fission product release to containment has been specified on the basis of EPA guidance[10] for actions to protect the public in the early phase of a nuclear incident. The approximately 24-hour period is considered to provide significant margin for accident detection, notification of the public in the community around the site, and offsite protection measures such as ad hoc evacuation.

The methodology for demonstrating the 1 rem dose criterion is based on deterministic analyses. The source term to be utilized by the design certification applicant as part of the demonstration is a physically-based source term. A physically-based source term is proposed for design basis applications for the ALWR as well as for emergency planning use. It specifies fission product release timing and magnitude to containment, chemical form of the fission products, fission product removal from containment, and fission product holdup in the secondary building. The physically-based source term is based on fission product release and removal phenomena from actual ALWR core damage sequences which, although extremely low in probability, are considered credible for purposes of defining the source term. The physically-based source term has been defined so as to envelope potential source terms from such sequences i.e., sequences having the characteristics defined above in Section 2.3. Thus, the physically-based source term provides significant margin beyond the actual fission product release which would be expected if a core melt accident were assumed to occur at an ALWR. The physically-based source terms which were developed by the ALWR Program in early 1992 for the passive PWR and BWR are given in Tables 2-4 and 2-5[17]. Additional ALWR Program work, mainly on fission product removal, was submitted to NRC in 1993 (for example see reference [19]). NRC is presently working on an updated design basis source term[20] which is similar to the ALWR physically-based source term. The source term to be used by design certification applicants will reflect the resolution of differences between the NRC and ALWR source term, which is being addressed as of this writing. Major differences are not expected.

The methodology specified for the dose evaluation is similar in concept to what is typically done in Level 3 PRA evaluations, e.g., a CRAC2 or MACCS calculation. Median meteorological conditions are specified on the basis that the ALWR physically-based source term has significant

Table 2-4

PWR Release Fractions to Primary Containment Atmosphere*

Nuclide	0-1hr. Coolant Activity	1-5 hr. Early In-Vessel	5 hr.*** Ex-Vessel	5-24 hr. Late In-Vessel	Total
Nobles	**	0.80	--	0.20	1.0
I		0.38	--	0.17	0.55
Cs		0.30	--	0.18	0.48
Te		0.08	--	0.03	0.11
Sr, Ba		0.004	--	--	0.004
Ru		0.004	--	--	0.004
Remainder		0.00004	--	--	0.00004

Table 2-5

BWR Release Fractions to Primary Containment Atmosphere*

Nuclide	0-1hr. Coolant Activity	1-3 hr. Early In-Vessel	3 hr.*** Ex-Vessel	3-24 hr. Late In-Vessel	Total
Nobles	**	0.80	--	0.20	1.0
I		0.30	--	0.20	0.50
Cs		0.23	--	0.18	0.41
Te		0.06	--	0.03	0.09
Sr, Ba		0.003	--	--	0.003
Ru		0.003	--	--	0.003
Remainder		0.00003	--	--	0.00003

Notes:

* All numbers are fraction of original core fission product inventory.

** Coolant activity makes a negligible contribution to the source term from a core damage event and so is not included here.

*** All nobles released either early or late in-vessel. Remaining fission products retained in quenched debris or scrubbed through overlying water pool in reactor cavity (PWR) or drywell (BWR).

margin to that expected from any credible ALWR core damage sequence source term as noted above. Thus the combination of median meteorology and the physically-based source term bounds most core melt sequences. The site meteorology which has been specified for design certification applicant dose calculations is that which is in the URD Key Assumptions and Groundrules for PRA. This site was selected to have a Chi/Q greater than 80 to 90 percent of U.S. operating nuclear plant sites to provide siting flexibility for the ALWR. Committed effective dose equivalent (CEDE) is to be used (as opposed to the older whole body concept) on the basis of the recent EPA report[10] and revised 10CFR20[21].

2.5 SUPPLEMENTARY PRA EVALUATION

As described in Sections 2.3 and 2.4, the two ALWR emergency planning criteria, containment performance and offsite dose, stress a deterministic approach. To complement the deterministic approach associated with the criteria, a supporting PRA evaluation has also been specified. The PRA is required to demonstrate that core damage frequency is less than 10^{-5} per year and that the cumulative frequency for sequences that result in greater than 1 rem for 24 hours at 0.5 mile from the reactor is less than 10^{-6} per year. As part of the PRA evaluation, it is also to be demonstrated that the prompt accident quantitative health objective of the NRC Safety Goal Policy[22] is met with no credit for offsite evacuation prior to 24 hours.

The PRA goals are not emergency planning criteria, nor is it intended that the goals be made part of design certification or any other rulemaking. Rather the PRA is intended to demonstrate the integrated effectiveness of the two emergency planning criteria and to serve as a tool for the Plant Designer for refining and optimizing the design. Also, the PRA will provide additional confidence to the NRC in the overall safety of the design and in the margin to NRC guidelines on core damage frequency and large release. Finally, the NRC Safety Goal Policy quantitative health objective provision demonstrates that an acceptable level of radiological risk to the public, as defined by the NRC Safety Goal Policy, can be achieved with ad hoc evacuation which can be accomplished with significant margin within 24 hours.

As noted in Section 2.2 above, this approach of deterministic criteria, with PRA used as a supporting evaluation, is consistent with the industry interpretation of the NRC Severe Accident Policy[5] which states that safety acceptability should be based on an approach which stresses deterministic engineering analysis and judgment, complemented by PRA.

Section 3.0

PRELIMINARY ASSESSMENT OF PASSIVE ALWR DESIGN CONFORMANCE WITH REQUIREMENTS

Two passive plant designs, the Westinghouse AP600 and the General Electric SBWR, have been submitted to NRC for design certification under 10CFR52. A preliminary assessment of these standard passive plant designs has been conducted to determine the degree to which they meet the ALWR emergency planning design criteria. The assessment is based on a review of the AP600 Standard Safety Analysis Report (SSAR)[23] which was completed in June, 1992, and the SBWR SSAR[24] which was completed in February, 1993.

While this preliminary assessment has been conducted for the passive plants, both ABB-CE and General Electric have committed to perform similar assessments for their evolutionary designs, System 80+ and ABWR, which are presently in the design certification process.

3.1 CONTAINMENT PERFORMANCE CRITERION

The containment performance criterion for emergency planning appears in Chapter 1 of the URD and is repeated in Appendix A and discussed in Section 2 above.

The steps used for the preliminary assessment of compliance with the criterion were as follows:

1. Confirm that the design meets the requirements of the URD, Chapter 5, Section 6.6.2.1 by performing a comparison between the passive plant design characteristics and features and the requirements identified in Reference 14 and summarized in Appendix B.

2. Confirm that containment loads representing those from core damage sequences do not exceed ASME limits specified in the URD Chapter 5, Section 6.6.2.2 for approximately 24 hours under realistic severe accident assumptions.

3. Confirm that no uncontrolled release will occur beyond approximately 24 hours.

3.1.1 Plant Design Characteristics and Features to Address Containment Challenges

The preliminary assessment was performed by reviewing the respective SSARs to confirm, for each containment challenge, the existence of specific design features or characteristics to fulfill the key URD requirements associated with the challenge. The list of challenges and associated requirements as summarized in Appendix B was used for this review. A requirement was considered met when an explicit reference to the system, feature, or characteristic was made in the SSAR.

Table D-1 in Appendix D summarizes the results of the preliminary assessment for AP600. This table lists the challenges and associated requirements from Appendix B, and identifies in brackets the sections of the AP600 SSAR which address each requirement. With the exception of the items identified in Table D-2, specific SSAR design features or capabilities have been identified in response to the requirements.

Table E-1 in Appendix E summarizes the results of the preliminary assessment for SBWR. This table also lists the challenges and associated requirements from Appendix B, and identifies in brackets the sections of the SBWR SSAR which address each requirement. With the exception of the items identified in Table E-2, specific SSAR design features or capabilities have been identified in response to the requirements.

On the basis of the preliminary assessment, it is expected that the AP600 and SBWR will be able to demonstrate that the requirements of Chapter 5, Section 6.6.2.1 of the URD are met. While there are several exceptions which require additional action to resolve, these exceptions are not major and are expected to have little, if any, impact on the design.

3.1.2 Containment Evaluation Against ASME Limits

As discussed in Section 2, for plant designs which meet all of the URD provisions related to containment challenges, the severe accident sequences for which containment performance should be evaluated are low pressure core melts into an intact containment with the RCS at low pressure and containment systems functioning as designed.

A preliminary assessment of AP600 containment performance against ASME limits has been performed by evaluating a low pressure core melt sequence from the AP600 PRA. In this Base

Case sequence presented in the PRA, the accident is caused by a 4 inch LOCA, with successful depressurization but failure of the internal refueling water storage tank (IRWST) to inject due to check valve failure. The core is uncovered at 2 hours and the vessel fails at appr ximately 12 hours. The debris is cooled in the reactor cavity to less than 800 F but temporarily reheats to 1340 F after the water present in the reactor cavity is boiled off. Condensation from the passive containment cooling system (PCCS) eventually results in the IRWST water overflowing into the reactor cavity, cooling the debris. Hydrogen produced from metal oxidation is controlled by igniters. The containment peak pressure and temperature are 47 psia and 368°F respectively, well under the design pressure of 60 psia. The conditions corresponding to Service Level C have been determined in the SSAR to be 104 psia at 400°F, and the ultimate capacity has been determined to be 135 psia at 400°F. Thus, the AP600 containment design provides substantial margin to loads which would be expected should a severe accident occur.

In addition, variations on the Base Case sequence as well as other sensitivity sequences were analyzed. The variations on the Base Case sequence were taken from dominant accident scenarios determined in the Level 1 PRA. These additional analyses address the sensitivity of the results to ex-vessel debris coolability, containment pressurization due to core concrete interaction, hydrogen igniter operation, creep rupture of reactor coolant piping system, and availability of PCCS water. A summary of the sequences analyzed and the corresponding containment pressures and temperatures are presented in Table 3-1.

Based on the results in Table 3-1, sequences involving low pressure core melt into an intact containment with containment systems functioning as designed meet the Service Level C limit with significant margin. Even the sensitivity sequences in Table 3-1, in which containment systems are assumed to have degraded performance, meet the 24-hour Service Level C criterion. Three of the sequences analyzed in the AP600 PRA are associated with the containment bypass and isolation failure release type. Passive design capability to preclude or accommodate these types of events has been provided. On the basis of this design capability, this release type is not considered credible. Further, its PRA frequency is roughly an order of magnitude less than the URD 10^{-7} per year threshold. It is also noted that the three sequences presented in Table 3-1 represent a bound of eight PRA sequences which have a range of release timing. The majority of these eight sequences has release beginning after 24 hours, with about half having release after 72 hours. The frequency of release before 24 hours is about 8×10^{-9} per year.

From this preliminary assessment, there is confidence that the AP600 will be able to meet the ASME Service Level C limits.

Table 3-1

SUMMARY OF AP600 SEVERE ACCIDENT SEQUENCE CONDITIONS

REPRESENTATIVE SEQUENCES	CONTAINMENT MAXIMUM PRESSURE AND TEMPERATURE Design Pressure: 60 psia Service Level C: 104 psia @ 400F Ultimate Capacity: 135 psia @ 400F	REMARKS	RELEASE TYPE	FREQUENCY OF RELEASE TYPE (Per Year)	SITE BOUNDARY DOSE LEVELS FOR RELEASE TYPE (24 HRS AFTER CORE DAMAGE) MEDIAN DOSE (*Rem*)
Base Case BC1: Loss of Coolant Accident (LOCA) with In-Containment Refueling Water Storage Tank Check Valve Failure	47.1 psia 368°F		Release associated with the leakage from an intact containment that is not pressurized above the design pressure.	2.5×10^{-7}	0.07
VRP1: Vessel Rupture	45 psia 296°F				
SLP: Small LOCA with Passive Residual Heat Removal (RHR), Core Makeup Tanks (CMT) Fail, Automatic Depressurization System (ADS) Not Actuated	26.1 psia 215°F				
MLP: Medium LOCA, Passive RHR Fails, ADS Fails	36.3 psia, 296°F				
IGN: Igniter Failure. BC1 + 67% of cladding is reacted in-vessel and hydrogen igniters are turned off.	47 psia spike 800°F spike 29 psia 260°F at 24 hrs	Peak pressure from hydrogen burn does not exceed design pressure.			
CC: Passive Containment Cooling System (PCCS) Water Failure. BC1 + failure of PCCS water on outside of shell, three out of four CMT and accumulators available.	68 psia spike at 12 hrs 80 psia 296°F at 48 hrs	Assuming constant rate of pressurization by non-condensable gases generated from CCI, containment failure is expected greater than 4.2 days.	Release associated with leakage from a containment which is overpressurized by noncondensible gas from CCI	7.6×10^{-10}	0.08

Table 3-1

SUMMARY OF AP600 SEVERE ACCIDENT SEQUENCE CONDITIONS
(Cont'd)

REPRESENTATIVE SEQUENCES	CONTAINMENT MAXIMUM PRESSURE AND TEMPERATURE Design Pressure: 50 psia Service Level C: 104 psia @ 400F Ultimate Capacity: 135 psia @ 400F	REMARKS	RELEASE TYPE	FREQUENCY OF RELEASE TYPE (Per Year)	SITE BOUNDARY DOSE LEVELS FOR RELEASE TYPE (24 HRS AFTER CORE DAMAGE) MEDIAN DOSE *(Rem)*
OKP: PCCS Failure, Coolable Debris. CC + four out of four CMT and accumulators available.	75.4 psia 314°F	Containment has reached a steady state at 27 hours, not expected to fail.	Release associated with leakage from an intact containment that has been pressurized above design but below Service Level C pressure.	5.6×10^{-8}	0.12
LFW1: Loss of Feedwater and Containment Isolation. Passive RHR, CMT and ADS Fail	25.7 psia 396°F		Release associated with the leakage from a containment that is bypassed or has not been isolated.	2×10^{-8}	>1
SGTR: Steam Generator Tube Rupture (SGTR), Steam Generator Safety Valve Stuck Open. Passive RHR actuated on high-temperature signal in hot leg rather than a low steam generator level. ADS fails.	23.5 psia 188°F	Core not predicted to become uncovered until after 72 hours.			
SG2: SGTR + Passive RHR failure, ADS fails.	50 psia spike 1340°F spike 22 psia 260°F at 32 hrs				

Table 3-1

SUMMARY OF AP600 SEVERE ACCIDENT SEQUENCE CONDITIONS
(Cont'd)

REPRESENTATIVE SEQUENCES	CONTAINMENT MAXIMUM PRESSURE AND TEMPERATURE Design Pressure: 60 psia Service Level C: 10[illegible] psia @ 400F Ultimate Capacity: 135 psia @ 400F	REMARKS	RELEASE TYPE	FREQUENCY OF RELEASE TYPE (Per Year)	SITE BOUNDARY DOSE LEVELS FOR RELEASE TYPE (24 HRS AFTER CORE DAMAGE) MEDIAN DOSE *(Rem)*
SENSITIVITY ANALYSES PERFORMED INDEPENDENTLY OF PROBABILITY OF OCCURRENCE					
CR: Creep Rupture Size Sensitivity. Half square-foot creep rupture in RCS assumed.	50 psia spike 1250°F spike 26 psia, 215°F at 28 hrs	Containment pressure reaches equilibrium below design pressure.		5×10^{-9}	
DRY: Passive Core Cooling System Failure. Case BC1 assuming failure of all passive core cooling system water sources.	45 psia spike 332°F spike 32.7 psia, 260°F at 25 hrs	Assuming constant rate of core-concrete interaction the basemat fails at 8.8 days; overpressurization occurs at 4.9 days.		$<10^{-10}$	
CHF: Debris Coolability Sensitivity. Case BC1 assuming the debris not coolable even though cavity is flooded	42 psia spike 550°F spike 34.8 psia, 280°F at 25 hrs	Basemat fails due to CCI at 26.5 days. Overpressurization occurs at 16.7 days.		Not Calculated	

A preliminary assessment of SBWR containment performance against ASME limits has also been performed by evaluating low pressure core melt sequences from the SBWR PRA. The two base sequences LPL-SN and LPE-SN are similar in nature. The initiating event is an inadvertently open relief valve which depressurizes the reactor. This initiating event is very similar to a LOCA and is used to determine the consequences from a LOCA. The reactor scrams and the Main Steam Isolation Valves (MSIVs) close. The feedwater pumps trip and the automatic depressurization system (ADS) opens the remaining safety relief valves (SRVs) and the depressurization valves (DPVs). All high and low pressure injection systems are assumed to fail. No credit is taken for operation of the Isolation Condenser (IC) units. At approximately 50 minutes into the event, core uncovery occurs which eventually leads to reactor vessel lower head penetrations failure at about 4.5 hours. Corium is deposited on the lower drywell floor which causes the flooder to open due to high local temperature. The debris is quenched and core concrete interaction does not occur. Steam generation in the lower drywell leads to further increase in the containment pressure until the PCCS heat removal capacity equals the decay heat generated by the core debris. The long-term containment pressure is about 0.56 MPa (80 psia) which is below the wetwell vent pressure setpoint of 0.93 MPa (135 psia). The containment temperature is approximately 530K (495°F). The conditions corresponding to ASME Unit Factored Load and Service Level C have been determined in the SSAR to be 118 psia at 500°F, and the ultimate capacity has been determined to be 215 psia at 500°F. Thus, the SBWR containment design provides substantial margin to loads which would be expected should a severe accident occur. The LPL-SN sequence is the same as LPE-SN except that one gravity drain cooling system pool injects water into the vessel delaying reactor vessel failure by about 8 hours. In both cases, normal containment leakage is the only mode of fission product release.

In addition, other sequences were analyzed to address the sensitivity of the results to vessel rupture, high pressure core melt, limited debris coolability, failure of the flooder, failure of containment heat removal and dominant release path. Seventeen additional sequences were evaluated. Table 3-2 presents a summary of the nineteen sequences and the corresponding containment pressures and temperatures.

Based on the SBWR SSAR analyses, sequences involving low pressure core melt into an intact containment with containment systems functioning as designed meet the Service Level C limit with significant margin.

Some of the sequences analyzed in the SBWR PRA include system failures beyond the failures in the two basic sequences, e.g., high pressure melt ejection and containment bypass. Passive

Table 3-2

SUMMARY OF SBWR SEVERE ACCIDENT SEQUENCE CONDITIONS

REPRESENTATIVE SEQUENCES	CONTAINMENT MAXIMUM PRESSURE AND TEMPERATURE Design Pressure: 0.48 MPa (70 psia) Service Level C: 0.81 MPa @ 533K, (118 psia @ 500F) Ultimate Capacity: 1.48 MPa @ 533K, (215 psia @ 500F)	REMARKS	RELEASE TYPE	FREQUENCY OF RELEASE TYPE *(Per Year)*
SEQUENCES WITH VESSEL FAILURE AT LOW PRESSURE				
LPE-SN: Inadvertently open relief valve (IORV), MSIVs close, feedwater pumps trip, ADS opens, high and low pressure injection fail, no credit taken for IC. Drywell sprays fail , flooders operate after vessel failure.	0.56 MPa (80 psia) 530K (495F)		Release associated with the leakage from an intact containment that is not pressurized above Service Level C	7×10^{-8}
LPL-SN: Same as LPE-SN except that one GDCS pool injects into vessel, delaying vessel failure by approximately 8 hours.	0.60 MPa (87 psia) ~530K (495 F)		Release associated with the leakage from an intact containment that is not pressurized above Service Level C	6.4×10^{-8}
LPE-SCV: Same as LPE-SN assuming that the debris is not coolable.	0.93 MPa (135 psia) before venting 600K (620F)	1.68m of concrete ablation in lower drywell after 80 hours	Scrubbed release from wetwell vent at 28.7 hrs	1.1×10^{-8}
LPE-SCD: Same as LPE-SCV assuming vent is not opened at 28.7 hrs.	1.0 MPa (145 psia) @ the time of head failure 600K (620F)		Release through a failed drywell head at 31.2 hrs	1.6×10^{-9}
LPL-SCV: Same as LPL-SN assuming that the debris is not coolable.	0.93 MPa (135 psia) before venting ~600K (620F)	1.46m of concrete ablation in lower drywell after 80 hours	Scrubbed release from wetwell vent at 36.6 hrs	1.1×10^{-8}

Table 3-2

SUMMARY OF SBWR SEVERE ACCIDENT SEQUENCE CONDITIONS
(Cont'd)

REPRESENTATIVE SEQUENCES	CONTAINMENT MAXIMUM PRESSURE AND TEMPERATURE Design Pressure: 0.48 MPa (70 psia) Service Level C: 0.81 MPa @ 533K, (118 psia @ 500F) Ultimate Capacity: 1.48 MPa @ 533K, (215 psia @ 500F)	REMARKS	RELEASE TYPE	FREQUENCY OF RELEASE TYPE *(Per Year)*
SEQUENCES WITH VESSEL FAILURE AT LOW PRESSURE - *(Cont'd)*				
LPL-SB: Same as LPE-SN except that one MSIV fails to close.		Containment Bypass		$<10^{-10}$
LPE-SWV: Same as LPE-SN assuming failure of containment heat removal (both PCCS and Suppression Pool Cooling)			Scrubbed release from wetwell vent at 40.2 hrs	2.3×10^{-10}
SEQUENCES WITH VESSEL FAILURE AT INTERMEDIATE PRESSURE				
MPE-SN: Same as LPE-SN except that DPVs fail to open leading to a medium pressure core melt.	0.73 MPa (106 psia) ~470K (400F)		Release associated with the leakage from an intact containment that is not pressurized above Service Level C	$<10^{-10}$
MPL-SN: Same as LPL-SN except that DPVs fail to open leading to a medium pressure core melt.	0.89 MPa (129 psia) ~470K (400F)		Release associated with the leakage from an intact containment that is pressurized below wetwell vent pressure	$<10^{-10}$
MPE-SCV: Same as LPE-SCV except that DPVs fail to open leading to a medium pressure core melt.		1.65m of concrete ablation in lower drywell after eighty hours	Scrubbed release from wetwell vent at 38.9 hrs	$<10^{-10}$
MPL-SCV: Same as LPL- SCV except that DPVs fail to open leading to a medium pressure core melt.			Scrubbed release from wetwell vent at 39.7 hrs	$<10^{-10}$
MPL-SCD: Same as LPL-SCV except that DPVs fail to open leading to a medium pressure core melt, and vent is not opened			Release through a failed drywell head at 38 hours	8.1×10^{-10}

Table 3-2

SUMMARY OF SBWR SEVERE ACCIDENT SEQUENCE CONDITIONS

(Cont'd)

REPRESENTATIVE SEQUENCES	CONTAINMENT MAXIMUM PRESSURE AND TEMPERATURE Design Pressure: 0.48 MPa (70 psia) Service Level C: 0.81 MPa @ 533K, (118 psia @ 500F) Ultimate Capacity: 1.48 MPa @ 533K, (215 psia @ 500F)	REMARKS	RELEASE TYPE	FREQUENCY OF RELEASE TYPE *(Per Year)*
SEQUENCES WITH VESSEL FAILURE AT HIGH PRESSURE				
HP-N: Loss of site power, MSIV close and reactor scrams, feedwater pump trip, high pressure injection fails, ADS fails, but SRVs cycle at setpoint pressure. IC inoperable. Flooders and drywell sprays operate after vessel failure.	0.63 MPa (92 psia) 450K (350F)		Release associated with the leakage from an intact containment that is not pressurized above Service Level C	7.7×10^{-9}
HP-SG: Same as HP-N except that drywell sprays fail.	0.68 MPa (99 psia) 800K (980F)		Leakage through the drywell head due to high temperature seal degradation at <24 hrs	5.5×10^{-10}
HP-SFG: Same as HP-N except that flooders and drywell sprays both fail.	0.57 MPa (83 psia) ~800K (980F)	Some CCI occurs due to flooders failure. 0.05m of concrete ablation in lower drywell before debris is quenched.	Leakage through the drywell head due to high temperature seal degradation @ 27.7 hrs	$<10^{-10}$
HP-SCG: Same as HP-N except that drywell sprays fail and debris is assumed not to be coolable.	0.72 MPa (105 psia) ~800K (980F)	0.13m of concrete ablation in lower drywell after eighty hours	Leakage through the drywell head due to high temperature seal degradation at <24 hrs	4.3×10^{-9}

Table 3-2

SUMMARY OF SBWR SEVERE ACCIDENT SEQUENCE CONDITIONS
(Cont'd)

REPRESENTATIVE SEQUENCES	CONTAINMENT MAXIMUM PRESSURE AND TEMPERATURE Design Pressure: 0.48 MPa (70 psia) Service Level C: 0.81 MPa @ 533K (118 psia @ 500F) Ultimate Capacity: 1.48 MPa @ 533K (215 psia @ 500F)	REMARKS	RELEASE TYPE	FREQUENCY OF RELEASE TYPE *(Per Year)*
VESSEL RUPTURE SEQUENCES				
VR-SN: Large LOCA in RPV lower head, reactor scrams, ADS fails, all modes of injection fail, flooders operate after core relocates in lower drywell.	0.36 MPa (53 psia) 500K (440 F) @ 80 hrs		Release associated with the leakage from an intact containment that is not pressurized above design pressure	3.9×10^{-9}
VR-SX: Same as VR-SN except that containment is assumed to fail when core debris is expelled from the RPV.	500K (440F) @ 80 hrs		Containment fails at < 24 hrs	$\leq 10^{-10}$
VR-SCV: Same as VR-SN except that debris is assumed not to be coolable.		1.8m of concrete ablation in lower drywell after eighty hours	Scrubbed release from wetwell vent at 34 hrs	6.3×10^{-10}

NOTES - MAAP-SBWR SEQUENCE NAMING CONVENTION:

First two or three characters (Base Sequence):

- LPE Low Pressure Core Melt with Loss of Short-Term Coolant Makeup
- LPL Low Pressure Core Melt with Loss of Long-Term Coolant Makeup
- MPE Medium Pressure Core Melt (depressurization through SRVs only) with Loss of Short-Term Coolant Makeup
- MPL Medium Pressure Core Melt (depressurization through SRVs only) with Loss of Long-Term Coolant Makeup
- VR Vessel Rupture
- HP High Pressure Core Melt

Characters in Between First Two and Last Characters (Failures):

- C Limited Debris Coolability
- F Failure of the Flooder
- S Failure of the Drywell Sprays to Operate
- W Failure of Containment Heat Removal (Both PCCS and Suppression Pool Cooling)

Last Character (Dominant Release Path):

- N Normal Containment Leakage
- V Suppression Chamber Vent
- G Leakage Through Drywell Head Seal
- D Drywell Head Failure
- X Early Containment Failure
- B Containment Bypass

design capability to preclude or accommodate these types of events has been provided. On the basis of this design capability these release types are not considered credible. Their frequencies are confirmed to be an order of magnitude or more below the 10^{-7} per year URD threshold.

3.1.3 Assessment of Uncontrolled Release

For AP600 core damage sequences with adequate cavity flooding and debris coolability, no containment overpressure is expected. Even for sensitivity sequences that are assumed to lead to overpressurization by noncondensable gases or to basemat penetration, failure is predicted to occur much later than 72 hours after the onset of core damage. Three additional sensitivity cases (CR, DRY, and CHF) were analyzed in this regard, even though these cases have negligible frequency of occurrence. They are presented at the bottom of Table 3-1.

Similarly, for SBWR core damage sequences with adequate cavity flooding and debris coolability, no containment overpressure is expected. For sensitivity sequences in which the ex-vessel debris is assumed to be non-coolable, overpressure is predicted to be reached at about 30 hours, at which time overpressure protection from the suppression pool vapor space (i.e., a scrubbed release) could be utilized if necessary.

3.2 DOSE CRITERION

The dose criterion limits the dose at 0.5 mile from the reactor from a physically-based source term to less than 1 rem for approximately 24 hours from the start of release of fission products into the containment.

Dose evaluations have been performed in the AP600 PRA. The Base Case sequence described in the PRA closely approximates the URD physically based source term with 100% noble gas release and 61% volatile fission product release. The containment leak rate is taken as the AP600 design leakage of 0.12 volume %/day. The containment leaks from the penetration area to the middle annulus between the primary and secondary containment shell which results in holdup of fission products and a reduction in offsite dose of about a factor of 20. The dose evaluation was performed using the MACCS code assuming that the release occurs at ground level and that 5% of the iodine release to containment is volatile and does not deposit. The median dose after 24 hours from the start of release of fission products from the fuel is 0.07 rem CEDE, well under the 1 rem level.

As stated in the AP600 PRA, variations on the Base Case and sensitivity sequences with isolated containment have fission product releases to the containment that are bounded by the URD physically-based source term. The release type associated with containment bypass and isolation failure sequences has dose greater than 1 rem, but as noted above and discussed in Section 2, such sequences are not considered credible as the passive plant has been designed to preclude such challenges. Offsite doses and frequencies of release for AP600 are presented in Table 3-1. This table summarizes the approach followed in the PRA in which four release types have been identified and quantified in terms of frequency of occurrence.

No evaluation of AP600 against the 5 rem thyroid limit specified in the URD emergency planning design criteria and methodology was included in the PRA. However, on the basis of ALWR Program evaluations, the 5 rem thyroid limit can be met by AP600. Also, experience indicates that given the 0.07 rem CEDE result, the thyroid dose will be under 5 rem. Westinghouse has committed to provide the thyroid evaluation, and the ALWR Program will track this item.

Dose evaluations against the emergency planning dose criterion were not included in the SBWR SSAR. However, on the basis of ALWR Program evaluations, the SBWR is capable of meeting both the 1 rem CEDE and the 5 rem thyroid dose for a physically-based source term. This is not unexpected since, as discussed above, the SBWR maintains containment load below appropriate ASME limits for credible accident sequences (i.e., low pressure core melts with containment intact) which should lead to low offsite doses. General Electric has committed to provide the dose evaluations for SBWR, and the ALWR Program will track this item.

3.3 SUPPORTING PRA REQUIREMENT

The supporting PRA requirement is to demonstrate that the core damage frequency is less than 10^{-5} per year, that the cumulative frequency for sequences resulting in a dose at 0.5 mile greater than 1 rem for 24 hours is less than 10^{-6} per year, and that the prompt accident qualitative health objective of the NRC Safety Goal Policy is met with no credit for offsite evacuation prior to 24 hours.

A PRA was performed for the AP600 in accordance with Volume III, Chapter 1, Appendix A of the URD. The total mean frequency of core damage was estimated to be 3.3×10^{-7} per year for internal events at power. For external events the core damage frequency for fires and internal floods was estimated to be less than 10^{-7} per year. Other external events are site specific, but on the basis of design characteristics and features provided to address such events the contribution of these events to core damage frequency is also expected to be negligible. For shutdown conditions the core damage frequency was estimated to be less than 10^{-7} per year. Thus, the total core damage frequency is expected to have significant margin to the 10^{-5} per year URD goal.

The AP600 complementary cumulative distribution function (CCDF) for offsite dose for 24 hours has been developed in the PRA. The cumulative frequency for sequences resulting in greater than 1 rem is approximately $3x10^{-8}$ per year, thus providing significant margin to the URD 10^{-6}, 1 rem goal.

A PRA was also performed for the SBWR as required by Volume III, Chapter 1, Appendix A of the URD. The total mean frequency of core damage was estimated to be $1.8x10^{-7}$ per year for internal events at power. For external events the core damage frequency for fires and internal floods was estimated to be less than 10^{-6} per year. Other external events are site specific, but on the basis of design characteristics and features provided to address such events the contribution of these events to core damage frequency is also expected to be negligible. For shutdown conditions the core damage frequency was estimated to be less than 10^{-7} per year. Thus, similar to the AP600, the total core damage frequency for SBWR is expected to have significant margin to the 10^{-5} per year URD goal.

The SBWR CCDF for sequences resulting in greater than 1 rem over the course of the accident is approximately $2x10^{-8}$ per year, thus providing significant margin to the 10^{-6}, 1 rem requirement.

The SBWR SSAR indicates that the prompt accident quantitative health objectives of the NRC Safety Goal Policy are met with several orders of magnitude margin. No evaluation of AP600 against these objectives has been provided as yet. However, on the basis of ALWR Program evaluations, this objective can be met for AP600 with no credit for evacuation. Westinghouse has committed to demonstrate that the quantitative health objective is met, and the ALWR Program will track this item.

3.4 CONCLUSIONS REGARDING PASSIVE PLANT CONFORMANCE TO ALWR REQUIREMENTS

Based on this preliminary assessment it is expected that the passive plant designs will be able to meet the emergency planning design criteria. Additional conformance assessment work may be appropriate as the design evolves and to assure that the containment systems being provided are well-engineered as described in Section 2.3.1. It is recognized that the URD, as well as the plant specific designs, have continued to evolve since the SSARs were issued. This design evolution is not expected to impact the conclusions of this assessment, and in fact may further enhance plant performance. In any case, the Plant Designers are responsible to demonstrate that their certified designs meet the emergency planning design criteria.

Section 4.0

CONCLUSIONS

The overall conclusion from the work performed to date on the technical aspects of ALWR emergency planning is that the likelihood and consequences of a severe accident for an ALWR are fundamentally different from that assumed in the technical basis for existing emergency planning requirements 15 years ago. Specific conclusions are as follows:

The updated emergency planning technical basis should be utilized for the ALWR. The primary reason for this is that the ALWR plant design capability, along with the greatly improved technical understanding of severe accident risk which has evolved over the last 15 years, result in significantly reduced ALWR radiological risk.

A strong technical basis for updated emergency planning exists in the URD. A set of deterministic criteria in the areas of severe accident containment performance and offsite dose, supplemented by PRA goals, have been developed for ALWR emergency planning and included in Volume III of the URD. For standard plant designs which demonstrate that these criteria are met, even in the extremely unlikely event of a severe accident the containment has been designed to maintain integrity and thus any radioactivity release will be very slow and small. A period of approximately 24 hours or more exists before reaching offsite dose levels at which the U.S. EPA recommends that actions be taken to protect members of the public.

ALWR designs have excellent potential to meet the design criteria. A preliminary assessment of AP600 and SBWR conformance with the ALWR emergency planning design criteria has been performed and indicates that the designs will meet the criteria. The Plant Designers have committed to provide demonstrations as part of design certification that their respective designs meet the criteria.

Section 5.0

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APPENDIX A

ALWR Emergency Planning Criteria and Methodology

and

Updated Containment Performance Requirements

(Reproduced from Reference 1)

Section A.1

ALWR Emergency Planning Criteria and Methodology

(Volume III, Chapter 1, new Section 2.6)

Paragraph No.	Requirement	Rationale	Rev.
2.5.3.4.7	**Engineering As-built Walkdown**	**Engineering As-built Walkdown**	3
	A detailed plant walkdown shall be performed after each ALWR plant is constructed to complete the SMA process. The selected primary and alternate success paths shall be walked down using the guidance given in EPRI Report NP6041 to verify that the assumptions made in the SMA are valid. If any equipment in the success paths is determined to have an actual HCLPF less than the SME, it shall be evaluated to determine that the HCLPF will exceed the SSE by a suitable margin or shall be strengthened. The walkdown process shall include review of construction drawings and documents.	An essential part of an SMA is the engineering walkdown to look for potential undesirable seismic conditions in the completed plant which cannot be identified during the design process. The SMA walkdown is performed to verify that the calculated margins have been achieved. During the walkdown, the review team will look for obvious deficiencies in the success path components selected for review and will be cognizant of potential systems interaction issues which cannot be identified during the design process. The designer should anticipate all concerns that will be addressed during the walkdown.	3
2.6	**CRITERIA AND METHODOLOGY FOR ALWR EMERGENCY PLANNING**	**CRITERIA AND METHODOLOGY FOR ALWR EMERGENCY PLANNING**	5
	The Passive ALWR shall be designed to allow simplification and standardization of emergency planning. The Plant Designer shall perform an evaluation of the plant design against two ALWR emergency planning technical criteria prescribed below for containment performance and site boundary dose. The methodology which is specified for demonstrating the criteria shall be utilized in this evaluation.	Technical criteria and methodology are provided so as to specify what a Plant Designer seeking approval of ALWR emergency planning for a particular plant design must demonstrate during design certification. It is intended that these criteria and methodology form the technical basis for any necessary regulatory action (e.g., a generic emergency planning rule in parallel with Passive ALWR design certification rulemaking). The criteria and methodology are intended to be used in an integrated manner and the criteria should not be applied without utilizing the methodology specified in this section.	5

Paragraph No.	Requirement	Rationale	Rev.
2.6	**CRITERIA AND METHODOLOGY FOR ALWR EMERGENCY PLANNING (CONTINUED)**	**CRITERIA AND METHODOLOGY FOR ALWR EMERGENCY PLANNING (CONTINUED)**	5
	The Plant Designer shall also perform a supplemental PRA evaluation in support of the evaluation against the two ALWR emergency planning criteria.	The criteria and methodology for containment performance and dose evaluation are primarily deterministic. The PRA evaluation is not a criterion itself but rather is intended to complement the two criteria. This is consistent with the NRC Severe Accident Policy which states that safety acceptability should be based on an approach which stresses deterministic engineering analysis, complemented by PRA.	5
		The requirements in this section are generally unique to emergency planning although the containment performance criterion draws heavily on containment performance requirements in other locations of the Utility Requirements Document. The requirements which are unique to emergency planning apply only to plants which are seeking approval of ALWR emergency planning and not to other plants.	5
2.6.1	**Containment Performance Criterion**	**Containment Performance Criterion**	5
	For ALWR emergency planning, the plant shall be provided with the capability to address severe accident containment challenges, including design features and characteristics to preclude core damage sequences which could bypass containment, and to withstand loads representing those associated with core damage sequences. The methodology in Section 2.6.4 below shall be used to evaluate that capability.	While ALWR accident prevention design features make the possibility of core damage extremely remote, specifying the capability to address severe accident containment challenges, including avoiding containment bypass and withstanding loads which are expected to envelope best estimate pressure and temperatures associated with severe accident conditions, provides confidence that the containment can withstand a severe accident.	5
	ASME limits specified in Chapter 5, Section 6.6.2.2 should not be exceeded for a period of approximately 24 hours after the start of release of fission products from the fuel.	Meeting ASME limits for approximately 24 hours provides low leakage for the period corresponding to the site boundary dose criterion.	5

Paragraph No.	Requirement	Rationale	Rev.
2.6.1	**Containment Performance Criterion (Continued)**	**Containment Performance Criterion (Continued)**	5
	Beyond approximately 24 hours, means for preventing uncontrolled fission product release from containment shall be provided in accordance with Chapter 5, Section 6.6.2.5.	Even if a core damage event should occur, the ALWR Program considers that it is very likely that the ALWR containment would be able to meet appropriate ASME limits for an indefinite time period, i.e., no containment overpressure would occur. This is based on LWR accident management capabilities and the TMI-2 accident experience which suggest that it is likely that core damage events will be recovered in-vessel, and on ALWR reactor cavity design features (e.g., debris spreading area, flooding of debris) which are designed to quench the ex-vessel debris. Nevertheless, for defense-in-depth purposes, a requirement has been specified for no uncontrolled release beyond approximately 24 hours to provide protection against long-term containment overpressure failure. Radioactive decay and removal of fission products in containment is such that a release at 16 hours, or even earlier depending on the plant design, would result in no acute health effects at the site boundary. Thus, the approximately 24-hour period provides significant margin to that time at which the acute health effects dose threshold could be exceeded.	5

Paragraph No.	*Requirement*	*Rationale*	*Rev.*
2.6.2	**Site Boundary Dose Criterion**	**Site Boundary Dose Criterion**	5
	Dose at the site boundary shall be evaluated per the methodology in Section 2.6.5 below and shall be shown not to exceed 1 rem for a period of approximately 24 hours from the start of release of fission products from the fuel.	The 1 rem value is the Protective Action Guide (PAG) dose level which is specified by the Environmental Protection Agency in a 1991 report as guidance for actions to protect the public in the early phase of a nuclear incident.	5
		As noted in NUREG-1338, based on experience for non-radiological emergencies, ad hoc evacuations take from two to eight hours, including time to notify the public. Not exceeding the PAG for approximately 24 hours would provide significant margin for ALWR accident detection, notification, and ad hoc evacuation.	5
2.6.3	**Supplemental PRA Evaluation**	**Supplemental PRA Evaluation**	5
	A PRA evaluation shall be performed per the methodology in Section 2.6.6 below to demonstrate that the following goals are met: • A core damage frequency $\leq 10^{-5}$/yr; • A cumulative frequency $< 10^{-6}$/yr for sequences resulting in greater than 1 rem over 24 hours at the site boundary.	The requirement to perform the supplemental PRA evaluation and the associated goals are intended to demonstrate the integrated effectiveness of the two emergency planning criteria (Sections 2.6.1 and 2.6.2 above). The supplemental PRA also serves as a tool for the Plant Designer for refining and optimizing the design. Finally, the supplemental PRA will provide confidence to the NRC in the overall safety of the plant and in the margin to NRC guidelines on core damage frequency and large release. Given the guidance in the NRC Severe Accident Policy Statement, it is not intended that the PRA goals be made part of design certification or of any rulemaking on emergency planning.	5
	In addition, it shall be demonstrated that ALWR designs are consistent with the prompt accident quantitative health objective of the NRC Safety Goal Policy with no credit for evacuation prior to 24 hours.	This requirement demonstrates that an acceptable level of radiological risk to the public, as defined by the prompt accident quantitative health objective of the NRC Safety Goal Policy, can be achieved with ad hoc evacuation, which as noted in Section 2.6.2, can be accomplished with significant margin within 24 hours.	5

Paragraph No.	Requirement	Rationale	Rev.
2.6.4	**Methodology for Demonstrating Containment Performance Criterion**	**Methodology for Demonstrating Containment Performance Criterion**	5
	The Plant Designer shall demonstrate that the pressure and temperature loads associated with core damage sequences are no more limiting than the peak LOCA plus hydrogen loads of Chapter 5, Section 6.6.2.2. For plant designs meeting the requirements of Chapter 5, Section 6.6.2.1, the characteristics of the core damage sequences shall be as follows:	Chapter 5, Section 6.6.2.2, requires that the peak LOCA plus hydrogen loads not exceed applicable ASME limits. The loads associated with core damage sequences must therefore be no more limiting than the LOCA plus hydrogen loads.	5
	• Containment is isolated and otherwise intact (i.e., no bypass has occurred); • Reactor coolant system is depressurized to < 100 psig; • Ample water is in the reactor cavity/lower drywell prior to or immediately upon vessel penetration for cooling ex-vessel core debris; • Passive containment heat removal is adequate; • BWR containments are inerted, and hydrogen control system is functioning.	Consistent with Chapter 5, Section 6.6.2, and the report, *Passive ALWR Requirements to Prevent Containment Failure*, (DOE/ID-10291), December, 1991, design characteristics and features are to be provided which address severe accident challenges, including bypass and loads from core damage sequences. An exhaustive set of severe accident challenges, regardless of the probability of occurrence of the challenge, have been addressed based on systematic consideration of past PRAs, operating experience, severe accident research, and unique design aspects of the ALWR. The conclusion from the technical work in support of this requirement is that if core damage should occur, it will be into an intact containment with the RCS at low pressure and with containment systems functioning as designed.	5
	Best estimate severe accident methods shall be utilized in evaluating the loads. Accepted industry computer codes such as MAAP shall be applied.	Best estimate methods are appropriate for the severe accident evaluation since the evaluation relates to matters beyond the design basis, i.e., the ALWR Safety Margin Basis, and since the ALWR plant features for addressing severe accident challenges significantly reduce the uncertainty in severe accident phenomena.	5

Paragraph No.	Requirement	Rationale	Rev.
2.6.5	**Methodology for Demonstrating Site Boundary Dose Criterion**	**Methodology for Demonstrating Site Boundary Dose Criterion**	5
	The demonstration that the site boundary dose criterion is met shall utilize a physically-based source term as defined in Chapter 5, Section 2.4.1, including fission product release into an intact containment, and fission product removal from the containment and the secondary building as applicable in the design. The methodology for the PAG dose evaluation shall consist of the following.	The physically-based source term is based on release and removal phenomena from actual core damage sequences and is expected to envelope potential source terms from the probabilistically significant sequences. The intact containment is based on ALWR containment performance requirements which have been specified such that severe accident challenges to containment are effectively precluded or can be accommodated, thus providing integrity of the containment.	5
2.6.5.1	**Approach**	**Approach**	5
	A probabilistic dose (PD) method (e.g., CRAC2 or MACCS) shall be used.	A PD method is chosen for consistency with the basis for existing emergency planning and the fact that PD methods have provision for the particulate component of the source term and thus are an appropriate method for calculating PAG comparison doses. The use of CRAC2, MACCS, or another similar code is consistent with current level 3 PRA evaluations and ALWR PRA Key Assumptions and Groundrules (KAG).	5
2.6.5.2	**Meteorological Database**	**Meteorological Database**	5
	The meteorological database shall be that provided in Annex B to Appendix A to Chapter 1 of the URD.	This meteorological database is that provided in the PRA KAG. It is an actual site meteorological database for which the RG 1.145 two-hour Exclusion Area Boundary X/Q is estimated to be greater than the X/Q for 80 to 90 percent of U.S. operating sites.	5

Paragraph No.	Requirement	Rationale	Rev.
2.6.5.3	**Direction-Dependent vs. Direction-Independent**	**Direction-Dependent vs. Direction-Independent**	5
	The dose calculation shall be direction-independent.	The calculations supporting existing emergency planning are direction-independent, i.e., the frequency of exceeding given dose levels is provided independent of direction. The NRC safety goals use a direction-independent approach as well. The use of a direction-independent approach is also consistent with the methods to be used in preparing the complementary cumulative distribution function (CCDF) for the exceedance frequency of off-site doses at the site boundary required by the PRA KAG.	5
2.6.5.4	**Statistical Measure of Dose to be Compared to PAG Values**	**Statistical Measure of Dose to be Compared to PAG Values**	5
	The dose to be compared to the PAG values for ALWR emergency planning shall be the median dose.	Existing emergency planning used the PD method and, based on WASH-1400 source terms and frequencies, established that "most" core melt accidents would not exceed the PAG. There were two sources of variability in the supporting calculations which determined the meaning of "most" in this analysis: the source term itself (magnitude, timing, and elevation/plume energy) and the meteorology. The ALWR physically-based source term already has significant margin compared to "most" core melt source terms since for "most" Passive ALWR core melt accidents, the containment is expected to remain intact and the physically-based source term is bounding. Thus the comparison to the PAG value for ALWR emergency planning is based on the 50th percentile (i.e., median) dose since "most" core melt accidents would result in doses equal to or less than the median value calculated using the PD method involving weather as the only other source of variability.	5

Paragraph No.	Requirement	Rationale	Rev.
2.6.5.5	**Whole Body Dose vs. Effective Dose Equivalent**	**Whole Body Dose vs. Effective Dose Equivalent**	5
	The effective dose equivalent (EDE) shall be used.	The October 1991 revision to *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents* (PAG Manual) calls for the use of EDE as the basis for determining off-site doses in relation to the 1 rem PAG. MACCS already employs this concept, as does the current 10CFR20.	5
2.6.5.6	**Comparison to Thyroid Dose PAG**	**Comparison to Thyroid Dose PAG**	5
	The thyroid dose shall not exceed 5 rem.	Since the October 1991 revision of the PAG Manual continues to consider the thyroid PAG, it is appropriate to meet that guideline as a condition for ALWR emergency planning.	5
2.6.5.7	**Inclusion of Organic Iodide in the PAG Calculation**	**Inclusion of Organic Iodide in the PAG Calculation**	5
	In calculating doses for comparison with the PAG values to justify ALWR emergency planning, the contribution from organic iodide can be neglected.	The I and HI are quite reactive and are likely to undergo natural deposition as rapidly (or more rapidly) than the particulate. Given that pH is controlled as specified in the Utility Requirements Document, the dose contribution from organic iodide is very small (a few percent of thyroid dose) and thus can be omitted from the dose calculation.	5
2.6.5.8	**Dose Commitment**	**Dose Commitment**	5
	A dose commitment of 50 years shall be included.	In the October 1991 revision of the PAG Manual, it is required that the EDE be a committed value or CEDE, where the commitment is assumed to the "lifetime". It is judged that a 50-year commitment is adequate on a generic basis to fulfill that requirement; it is also the duration used in the current 10CFR20.	5
		This differs from the PRA as specified in the KAG where the intent is to compare calculated doses to the 25 rem threshold for acute health effects (based on the current 25 rem whole body requirement in 10CFR100).	5

Paragraph No.	*Requirement*	*Rationale*	*Rev.*
2.6.5.9	**Radionuclides to be Included**	**Radionuclides to be Included**	5
	The radionuclides identified in Table II-2 of the CRAC2 User's Guide (NUREG/CR-2326) shall be the minimum list of radionuclides included in the calculation of doses for the purpose of meeting the limits for ALWR emergency planning.	There are 54 radionuclides identified in this list. In MACCS there are six additional radionuclides: Sr-92, Y-92, Y-93, Ba-139, La-141, and La-142. These are not critical for the PAG comparison calculation; the impact of the Sr, Y, Ba and La isotopes already included in the CRAC2 list is much greater, given their relative quantities, half-lives and dose conversion factors; therefore, the CRAC2 list is acceptable.	5
2.6.5.10	**Dose Conversion Factors**	**Dose Conversion Factors**	5
	External dose conversion factors (plume and ground exposure) shall be based on Kocher, D.C., "Dose Rate Conversion Factors for External Exposure to Photons, and Electron Radiation from Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities," *Health Phys., Volume 38*, pp. 543-621 (1980). Inhalation dose conversion factors shall be based on Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," *Office of Radiation Programs*, USEPA (1988).	Federal Guidance Report No. 11 is the document referenced by the October 1991 revision of the PAG Manual. However, in this guide, external dose conversion factors are provided only for noble gases. The external dose conversion factors used in MACCS for NUREG-1150 calculations are referenced in NUREG/CR-4551 to the specified Health Physics article. These are judged to be acceptable for the use described herein. The inhalation dose conversion factors provided in the guide are for a 50-year "lifetime" commitment, consistent with 2.6.5.8 above.	5

Paragraph No.	*Requirement*	*Rationale*	*Rev.*
2.6.5.11	**Plume Modeling**	**Plume Modeling**	5
	The model used to treat dispersion in the calculation of doses for the purpose of meeting the limits for ALWR emergency planning shall be a straightline Gaussian plume. Plume centerline doses shall be reported. The values of σ_y and σ_z that are used to characterize the Gaussian plume expansion shall be based on Pasquill-Gifford curves. If the analytical model used in the analysis employs a uniform approximation of the expansion in the crosswind (y) direction (e.g., CRAC2), the final result shall be increased by an appropriate factor to provide centerline doses. In the case of CRAC2 (which employs a 3-σ_y "top hat" approximation of the cross-wind Gaussian distribution), the factor shall be 1.2.	The plume modeling in MACCS differs somewhat from that in CRAC2. The differences have been resolved as follows: • To demonstrate that the PAGs will not be exceeded within the exclusion area boundary (EAB) radius, the peak centerline value is the value that should be reported. To obtain this value, the CRAC2 results must be multiplied by a factor of 1.2. In addition, to compensate for the initially more disperse plume in CRAC2 (which results from setting the initial σ_y equal to building width/3 instead of building width/4.3), it is necessary to set the CRAC2 building width at the input level to 70% of its actual value.	5
	The initial σ_y shall be the building width divided by 4.3 if some other factor is used to determine the initial σ_y (e.g., a factor of 3 in CRAC2), and the building width specifiction shall be changed at the input level to compensate (e.g., the building width for CRAC2 shall be input as 70% of its actual value).		5

Paragraph No.	Requirement	Rationale	Rev.
2.6.5.11	**Plume Modeling (Continued)**	**Plume Modeling (Continued)**	5
	The correlation for dispersion in the vertical direction (z) shall be the form $\sigma_z = ax^b + c$ where x is the distance the plume has traveled. The values for a, b and c shall be the fixed values in CRAC2. In the event a simpler form has been employed for calculational ease (e.g., $\sigma_z = ax^b$ in MACCS), the coefficients shall be set to provide the same value of σ_z at a site boundary of 0.5 mile and at a low population zone (LPZ) radius of two miles as would be calculated using the fixed values for a, b and c in CRAC2. Those values are as follows:	• In CRAC2, the expansion in the z-direction (vertical) is controlled by an expression for σ_z as a function of plume travel, x. The expression has the form $\sigma_z = ax^b + c$ with the constants fixed in the coding. In MACCS, a different correlation which does not use an additive constant ("c" term) has been employed, but only for the purpose of convenience. For specific radial intervals of interest, values of a and b can be defined to give the same values of X/Q as CRAC2 at the two specific radial distances that define the interval. This is what has been done in this methodology specification. The 0.5-mile site boundary and 2-mile LPZ were chosen simply as typical radial distances.	5

Stability	*a*	*b*
A	2.47E-4	2.118
B	0.078	1.085
C	0.144	0.911
D	0.368	0.6764
E	0.2517	0.6720
F	0.184	0.6546

Paragraph No.	Requirement	Rationale	Rev.
2.6.5.11	**Plume Modeling (Continued)**	**Plume Modeling (Continued)**	5
	The time base for plume meander for long duration releases shall be the fixed value in CRAC2, three minutes.	• For long release times (greater than a few minutes), plume meander becomes an important factor in determining peak centerline doses. In CRAC2, the time base for plume meander was fixed at 3 minutes; in MACCS, it is a user input with 10 minutes having been used in NUREG-1150 and appearing in the standard problem input file. The data base supporting the modeling of plume meander includes averaging times (i.e., the time base) of approximately 3 to 10 minutes. Since the important parameter for plume meander is the ratio of release duration to the time base and since the release duration being used in the PAG assessment is 10 hours, per 2.6.5.14, duration to time base is better approximated by using the low end of the averaging range (i.e., the fixed CRAC2 value of 3 minutes) than the high end.	5
2.6.5.12	**Release Height and Energy of Release**	**Release Height and Energy of Release**	5
	The release height and energy of release assigned to the physically-based source term shall correspond to a cold, ground-level release for the purpose of calculating the dose.	Current severe accident analysis practice is to use release height and energy values that are consistent with the containment failure size/location or leak rate and associated thermodynamic conditions. However, for the ALWR physically-based source term, containment is intact, releases are not credited through a stack, and best estimate meteorology is used. Thus a cold, ground level release is appropriate.	5

Paragraph No.	Requirement	Rationale	Rev.
2.6.5.13	**Duration of Exposure to Ground Contamination**	**Duration of Exposure to Ground Contamination**	5
	The duration of exposure to ground contamination shall be 24 hours from the start of release of fission products from the fuel.	The 24-hour period provides margin for ALWR accident detection, notification, and ad hoc evacuation. The 24-hour period is also consistent with the existing emergency planning basis.	5
2.6.5.14	**Duration of Release and Number of Plume Segments**	**Duration of Release and Number of Plume Segments**	5
	The release duration to be used in calculating doses for the Passive ALWR physically-based source term shall be 10 hours if a single plume segment is used or 24 hours if multiple plume segments are used.	The CRAC2 code has a limit on release duration of 10 hours and can employ only a single plume. The MACCS code will accept a release duration greater than 10 hours and can employ multiple plumes (i.e., different source terms in succession), this capability being most useful when the character of the release to the environment abruptly changes in the course of an accident. This is not the case for the Passive ALWR physically-based source term, where the difference in dose between a 10-hour release duration and a 24-hour release duration is only a few percent.	5
2.6.5.15	**Shielding Factors**	**Shielding Factors**	5
	Shielding factors shall be 0.75 for plume exposure and 0.33 for exposure to ground contamination.	The values given are those from NUREG-0396, Section F, "no immediate protective actions" and are consistent with the "normal activity" requirement of the PRA KAG.	5

Paragraph No.	Requirement	Rationale	Rev.
2.6.5.16	**Breathing Rate and Inhalation Protection Factors**	**Breathing Rate and Inhalation Protection Factors**	5
	The breathing rate shall be $3.3 \times 10^{-4} m^3/sec$. For codes with provision for an inhalation protection factor, this value shall be set at 0.4. For codes without an inhalation protection factor, the breathing rate shall be reduced by a factor of 2.5.	The breathing rate identified in the October 1991 revision of the PAG Manual is the value specified. In the MACCS code, there is provision to reduce the inhalation dose by a factor to account for differences between the plume concentration and the concentration actually being breathed. NUREG/CR-4551 (one of the supporting documents for NUREG-1150) suggests an annual average value of 0.4 for normal activity (0.2 for active sheltering). The use of a "normal activity" inhalation protection factor is consistent with the requirements of the PRA KAG.	5
2.6.5.17	**Dry Deposition Velocity**	**Dry Deposition Velocity**	5
	The dry deposition velocity shall be 1.0 cm/sec for iodine and 0.1 cm/sec for other particulates.	These values are those of the October 1991 revision of the PAG Manual. Current severe accident analysis practice is to use values of 1.0 cm/sec (NUREG-0396/CRAC2) to 0.3 cm/sec (NUREG-1150/MACCS); the PRA KAG does not establish a requirement for dry deposition velocity.	5

Paragraph No.	Requirement	Rationale	Rev.
2.6.6	**Methodology for Performing Supplemental PRA**	**Methodology for Performing Supplemental PRA**	5
	The supplemental PRA shall be performed in accordance with the Volume III, Chapter 1, Appendix A, PRA Key Assumptions and Groundrules (KAG) with the exception that the off-site dose exceedance limit is 1 rem, per Section 2.6.3 above.	The KAG is the ALWR methodology for PRA evaluations. The KAG specifies that the PRA address internal events plus external events with the exception of seismic risk which is to be addressed by the seismic margin approach per Chapter 1, Section 2.5.3.4, of the URD.	5
	The required demonstration on the NRC Safety Goal Policy shall use the following methodology: • The ALWR reference site parameters in Annex B to the KAG shall be used. • No evacuation shall be assumed prior to 24 hours. Subsequent to 24 hours, the evacuation parameters of the KAG, Annex B, shall be used. • To demonstrate the NRC Safety Goal Policy quantitative objective for risk to an average individual (less than 0.1% of the risk from all other accidents), ALWR accident risk shall be less than 4×10^{-7} per person per year.	The numbers specified for risk comparisons are based upon recent data from the National Safety Council (Accident Facts, National Safety Council, 1988). The quantitative objective for latent cancer risks, which is also part of the NRC Safety Goal Policy, is not included in this required demonstration of Safety Goal compliance because, as noted in NUREG-1150, emergency response in close-in regions does not contribute substantially to differences in latent cancer risk. It is expected, however, that ALWRs would have no difficulty in meeting the latent cancer risk quantitative objective.	5

Section A.2

Updated Containment Performance Requirements

(Volume III, Chapter 5, revised Section 6.6.2)

Paragraph No.	Requirement	Rationale	Rev.
6.6.2	**Containment Performance**	**Containment Performance**	0
	The ALWR containment performance requirement shall consist of a number of elements as specified below. The initial element shall include a matrix of plant design characteristics and features to address a comprehensive set of containment challenges from severe accidents. This matrix approach, together with the other elements of containment performance shall provide high assurance of containment integrity and low off-site dose in the event of a severe accident.	The elements below comprise a deterministic approach to containment performance. The deterministic approach is complemented by the PRA requirements, including meeting ALWR PRA goals. This deterministic approach, complemented by PRA, is consistent with NRC Severe Accident Policy Statement guidance and provides the set of containment performance requirements that are considered necessary to address severe accidents. The combined set of deterministic and PRA requirements satisfies the Commission response to SECY-90-016 for a deterministic alternative which provides at least comparable mitigation capability to the conditional containment failure probability (CCF) of 0.1 but does not discourage improvements in core damage prevention.	5
6.6.2.1	**Plant Features to Address Containment Challenges**	**Plant Features to Address Containment Challenges**	5
	The plant shall include design characteristics and features to address a comprehensive set of severe accident challenges to the containment. Design characteristics and features shall include:	Design characteristics and features to address a comprehensive set of severe accident containment challenges are necessary to provide severe accident protection for the ALWR consistent with the NRC Severe Accident Policy, ALWR safety policy, and to meet the ALWR requirements. A complete set of design characteristics and features and the adequacy of these characteristics and features is documented in the report, *Passive ALWR Requirements to Prevent Containment Failure* (DOE/ID-10291), December 1991. In the report, an exhaustive set of severe accident challenges, regardless of	5

Paragraph No.	Requirement	Rationale	Rev.
6.6.2.1	**Plant Features to Address Containment Challenges (Continued)**	**Plant Features to Address Containment Challenges (Continued)**	5
		probability, have been addressed based on systematic consideration of past PRAs, operating experience, severe accident research, and unique design aspects of the ALWR. The report concludes that the severe accident challenges have been effectively precluded or can be accommodated by the ALWR design characteristics and features specified in the Requirements Document.	5
	• Features to provide reliable shutdown of the reactor by rod insertion, e.g., Chapter 4, Section 5.3 (BWR) and Chapter 4, Section 8.2 (PWR) as well as diverse reactivity control capability in the form of SLC, Section 4.5 (BWR) and PSIS, Section 5.2 (PWR).	• Reliable reactivity control, through rod insertion and the capability to accom odate failure to scram in the form of diverse means of reactivity insertion, limits the challenges associated with ATWS.	5
	• Features to reliably depressurize the RCS, e.g., Sections 4.4 (BWR) and 5.4 (PWR).	• A reliable depressurization system minimizes the probability of high pressure core melts with subsequent potential for direct containment heating. Cavity configuration also limits the magnitude of containment pressure rise.	5
	• Features to limit the generation of non-condensible gases as a result of corium-concrete interaction, e.g., Section 6.6.3.	• Containment integrity could be challenged in the long term as a result of pressure buildup from production of non-condensible gases following corium-concrete interaction. Preventing or limiting this event enhances containment performance.	5
	• Features that provide passive containment cooling for decay heat removal, e.g., Sections 4.3 (BWR) and 8.3 (PWR).	• Long-term containment cooling is required to maintain containment pressure within design limits.	5

Paragraph No.	Requirement	Rationale	Rev.
6.6.2.1	**Plant Features to Address Containment Challenges (Continued)**	**Plant Features to Address Containment Challenges (Continued)**	5
	• Features to handle the pressure and temperature resulting from generation of combustible gases, e.g., Section 6.5.	• Features that control combustion and prevent detonation of hydrogen eliminate this threat to containment integrity following a severe accident.	5
	• Features to assure containment integrity including isolation and precluding steam generator tube rupture and other containment bypass scenarios, e.g., Chapter 3, Section 2, and Chapter 5, Sections 4.3, 6.2, and 7.2, for the BWR and Chapter 3, Sections 2 and 4, and Chapter 5, Sections 5.3 and 6.2, for the PWR.	• Challenges to containment integrity which result from failures which occur independent of or coincident with core damage (e.g., containment bypass events) must be avoided.	5
6.6.2.2	**Containment Performance Structural Evaluation**	**Containment Performance Structural Evaluation**	0
	The Plant Designer shall demonstrate that the containment system pressure boundary, when subjected to the pressure and temperature loads from LOCA plus hydrogen described below, combined with the appropriate dead loads, meets the following ASME Code, Section III criteria:	The ASME Section III Code referenced structural integrity criteria satisfy the intended minimum requirements of 10CFR50.34(f)(3)(v). Also, any gross distortions and subsequent large strains in pressure boundary material due to potential shell buckling modes are precluded. The LDB requirements (Section 2.4.2) are expected to be limiting for inerted containments while the SMB requirements are expected to be limiting for containments which are not inerted.	4
	• For Class MC free standing steel vessels and for the steel portions of Class CC reinforced concrete vessels which are not backed up by concrete, the following requirements shall apply:		5
	- Paragraph NE-3221, Service Level C Limits on stress intensity values.		0

Paragraph No.	*Requirement*	*Rationale*	*Rev.*
6.6.2.2	**Containment Performance Structural Evaluation (Continued)**	**Containment Performance Structural Evaluation (Continued)**	0
	- For regions of ellipsoidal or torispherical shell surfaces of containment, the allowable compressive stress due to internal pressure shall not exceed 60 percent of the value of critical buckling stress determined by one of the methods given in ASME Subparagraph 3222.1(a).	- Compressive stress in ellipsoidal or torispherical shell heads due to internal pressure loading is a localized stress field which does not represent a challenge to overall containment stability; thus a lower factor of safety against buckling than otherwise permitted by Subparagraph 3222.2 is appropriate in these regions. The value of 60 percent of the critical buckling stress results in a safety factor of 1.67, which is consistent with the requirements of Code Case N-284 for local buckling.	5
	• For the steel liner portions of Class CC vessels which are backed by concrete, the factored load limits on liner strains established in Subarticle CC-3720 shall apply.		0
	• For those portions of other ASME Code class components which also constitute a portion of the containment systems pressure boundary, the corresponding ASME Section III Service Level C Limits shall apply.		0

Paragraph No.	Requirement	Rationale	Rev.
6.6.2.2.1	**Inerted Pressure Suppression Containments**	**Inerted Pressure Suppression Containments**	5
	The analysis of LOCA plus hydrogen loads shall assume: • Pool temperature equal to the peak temperature associated with the DBA LOCA within 24 hours from the accident initiation. • All nitrogen in the drywell is located in the wetwell airspace. • The total hydrogen equivalent to 100% active fuel cladding metal water reaction is located in the wetwell airspace.	The assumptions maximize the pressure and temperature loads in the containment in the performance of the 10CFR50.34(f)(3)(v) analysis.	5
6.6.2.2.2	**Non-Inerted Containments**	**Non-Inerted Containments**	5
	The analysis of LOCA plus hydrogen loads shall assume: • Peak pressure associated with the DBA LOCA; • Accumulation of hydrogen associated with 75% active fuel cladding metal water reaction; • Adiabatic isochoric complete combustion of this accumulated quantity of hydrogen. If containment is found to be steam inerted at the peak DBA pressure, then combustion shall be assumed to occur at the time steam condensation reduces the mole fraction of steam to combustible levels (~50% mole fraction steam).	The Licensing Design Basis analysis required by 10CFR50.34(f)(3)(v) would credit a hydrogen control system as hydrogen is generated. The Safety Margin Basis analysis requirement contained in this section postulates the peak DBA pressure and a realistic upper bound to total hydrogen concentration, i.e., that associated with 75% active clad oxidation, before crediting a hydrogen control system or ignition sources. This yields a higher peak pressure than that required by 10CFR50.34(f)(3)(v). Burning is assumed to occur at the highest potential containment pressure if inerting initially precludes combustion.	5

Paragraph No.	Requirement	Rationale	Rev.
6.6.2.3	**Severe Accident Sequence Selection for Reporting Containment Response**	**Severe Accident Sequence Selection for Reporting Containment Response**	0
	The Plant Designer shall report containment performance during severe accidents. Analysis of severe accident sequences shall be performed to confirm that the containment provides substantial margin with respect to severe accident challenges. Accident sequences from the PRA shall be selected for analysis of containment performance. PRA sequences shall be grouped into functional sequence types for the purpose of determining the mean total frequency of all accident sequences with approximately the same type of challenge. The sequence types shall be those resulting from the failure of any one of the following functions: • Reactivity insertion; • RCS depressurization; • Core or core debris coolant inventory control; • Containment pressure/temperature control; • Combustible gas control; • Containment isolation and containment bypass control; • Other functions, the failure of which could lead to containment challenge.	The primary means of addressing severe accident containment challenges is the deterministic matrix of design characteristics and features of Section 6.6.2.1 and the deterministic analyses of Section 6.6.2.2. This deterministic approach addresses an exhaustive list of containment challenges, regardless of probability. The probabilistic requirement of Section 6.6.2.3 complements the deterministic approach as required in the NRC Severe Accident Policy. The difficulty of assigning accurate numerical estimates notwithstanding, use of PRA in this manner provides valuable design insights and added confidence that containment margin exists for severe accidents and that important risk contributors have been addressed. This set of functions is considered necessary to assure containment integrity based on the report, *Passive ALWR Severe Accident Containment Performance Requirements*, January 1992. This report concludes that the only potentially significant severe accident challenges to a standard ALWR plant design which implements the provisions in the Requirements Document are those associated with core damage events that occur into an intact containment with the RCS at low pressure with containment systems functioning as designed.	5

Paragraph No.	*Requirement*	*Rationale*	*Rev.*
6.6.2.3	**Severe Accident Sequence Selection for Reporting Containment Response (Continued)**	**Severe Accident Sequence Selection for Reporting Containment Response (Continued)**	0
	Functional sequence types with mean frequency greater than approximately 10^{-7}/yr shall be analyzed for containment response.	The approximately 10^{-7}/yr threshold for functional sequence types to be analyzed for containment response is consistent with the NUREG-1420 10^{-7}/yr limit for insignificant risks and is consistent with Standard Review Plan guidance to evaluate potential accidents from hazards in the plant vicinity which exceed approximately 10^{-7}/yr. Also, NUREG-1150 uses a cutoff of 10^{-7}/yr for accident progression analysis. NUREG-1338 stated that any sequence appearing to have a frequency down to about 10^{-7}/yr will be examined from the standpoint of residual risk. Finally, consideration of functional sequence types greater than approximately 10^{-7}/yr provides assurance that the cumulative effects of such sequence types will not exceed the 10^{-6}/yr probability goal for off-site consequences.	5
	Functional sequence types with frequency less than 10^{-7} per year shall be reported for discussion: • Identifying the design features and operating characteristics credited to reach this low frequency; • Singling out the frequency of those sequence types which may result in early containment failure.	The purpose of this requirement is to assure that there is understanding of those features designed to preclude containment failure resulting from a severe accident. It is also expected to show that those phenomena which could lead to exceeding the capacity of containment early in a postulated severe accident event are a small fraction of the ALWR PRA goals for core damage frequency and consequences.	5
	The loads resulting from any analyzed functional sequence types shall be no more limiting than the peak LOCA plus hydrogen loads of Section 6.6.2.2 for approximately 24 hours after the start of fission product release from the fuel.	If the loads resulting from the analyzed severe accident sequence types are enveloped by the conditions determined for LOCA plus hydrogen in accordance with Section 6.6.2.2, the comparison of these severe accident loads may be made directly with the LOCA plus hydrogen loads. In the event the loads exceed those determined in accordance with Section 6.2.2.2, it is expected the Plant Designer will be able to	

Paragraph No.	Requirement	Rationale	Rev.
6.6.2.3	**Severe Accident Sequence Selection for Reporting Containment Response (Continued)**	**Severe Accident Sequence Selection for Reporting Containment Response (Continued)**	0
		demonstrate that the containment still meets the functional criteria for Service Level C or Unity Factored Load as permitted by 10CFR50.34(f)(3)(v) and provide confidence that the structural integrity and leak tightness of the passive plant containment will be maintained following a severe accident.	5
		Should any functional sequence type selected for analysis result in loads which exceed the functional criteria for Service Level C or Unity Load permitted by 10CFR50.34(f)(3)(v) or result in containment bypass, the Plant Designer should identify the reasons for the high loads or the bypass and explain why the accident sequence frequencies cannot be further reduced, and provide recommendations for an alternate basis on which confirmation of acceptable containment performance can be justified.	5

Paragraph No.	Requirement	Rationale	Rev.
6.6.2.4	**Containment Ultimate Capacity Analysis**	**Containment Ultimate Capacity Analysis**	0
	The Plant Designer shall perform an analysis to determine the ultimate structural capability of the containment. For steel containments, the ultimate capacity shall be defined as the pressure and temperature loadings which correspond to the collapse load defined by the method detailed in paragraph II-1430 of the ASME Code, Section III, Appendix II. For concrete containments, the ultimate structural capacity shall be defined as the pressure and temperature loading which produces liner plate strains equal to the liner strain limits of the ASME Code Section III, Subarticle CC-3720 for the Factored Load Category. The analysis shall consider the penetrations and their interaction with the containment, the shield building, and other structures internal or external to the containment, which might cause localized failure prior to the limit load for the overall pressure boundary. Results from testing of prototype details or models of prototype details may be used to augment such analyses. The failure mode associated with the ultimate structural capability shall be identified.	An analysis of containment ultimate capacity is required by Standard Review Plans 3.8.1 and 3.8.2, including the determination of pressure retaining capacity of localized areas. The failure analysis criteria included here are identical to or more conservative than those developed during NRC/IDCOR issue resolution (see ARSAP Technical Task 2.3 report) or are more realistically based on recent experimental tests for concrete containments by Sandia National Laboratories. These tests have indicated that concrete containment capability may be limited by leakage resulting from liner plate tears. EPRI report NP-6261 describes computer modeling techniques used to predict the failure mode of the scale model concrete containment tested by Sandia. Interaction of the containment penetrations with the shield building or other structures may produce leakage paths.	0

Paragraph No.	Requirement	Rationale	Rev.
6.6.2.5	**Long-term Containment Overpressure Protection**	**Long-term Containment Overpressure Protection**	5
	Protection of the containment for overpressurization beyond 24 hours shall be provided. Overpressure protection beyond 24 hours may be provided simply by the size and strength of the containment by demonstrating that the ASME limits specified in Chapter 5, Section 6.6.2.2, are not exceeded for approximately two to three days after the beginning of the accident.	Containment overpressure protection provides additional defense-in-depth to protect the containment from long-term catastrophic failure. The analysis shall credit design features for containment heat removal and debris cooling on the basis of Passive ALWR requirements directed at decay heat removal and providing water to the debris. The analysis should utilize best estimate analysis methodologies including realistic assumptions. On the order of two to three days is judged to be adequate time for actions by the plant staff to bring the accident under control.	5
6.6.3	**Cavity/Pedestal-Drywell Configuration**	**Cavity/Pedestal-Drywell Configuration**	0
6.6.3.1	**Retention of Core Debris**	**Retention of Core Debris**	0
	The reactor cavity/pedestal drywell shall be evaluated to confirm that quantities of core debris sufficient to jeopardize containment integrity will not be transported from the cavity/drywell after RPV failure and then either mix with the containment atmosphere while in a finely particulated form or establish direct contact with the containment boundary. For passive ALWRs, the evaluation shall address low-pressure (nearly complete depressurization) conditions prior to vessel failure unless a higher pressure sequence is identified as risk-significant in the PRA for a specific passive ALWR design.	The specified evaluation will confirm that direct containment heating is not an issue for passive ALWR designs, based primarily on the assured provisions for RCS depressurization, but also considering the specific proposed cavity/pedestal drywell geometry. The PRA will define the extent to which incomplete depressurization will be considered for a specific design.	0

APPENDIX B

Summary of ALWR Requirements to Address

Severe Accident Containment Challenges

(Reproduced from Reference 12)

Table 4

SUMMARY OF REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES THAT ARE INDEPENDENT OF OR COINCIDENT WITH CORE DAMAGE

CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE	ALWR BASIS#	KEY PASSIVE ALWR REQUIREMENTS: LIMIT POTENTIAL FOR CHALLENGE*	KEY PASSIVE ALWR REQUIREMENTS: ACCOMMODATE CHALLENGES*
1. Containment Isolation	Isolation	PWR/BWR	2	A Reduced fluid line penetrations Isolation provisions and leakage rate testing per standards. Valves capable of closure with possible flow and full containment pressure. Control room position indication for automatic and remote manual valves. A Manual valve configuration permits locking only in closed position. A Closed systems penetrating containment evaluated for ex-vessel severe accidents. Fail closed or DC powered isolation valves. A Capability for periodic gross check of containment integrity.	P Passive Residual Heat Removal minimizes core damage risk given isolation failure (with RHR on line even without DC power).
2. Interfacing System LOCA	Bypass	PWR/BWR	2	A Reduced interfaces between the Reactor Coolant System (RCS) and low pressure systems. A High to low pressure interfaces provided with isolation valves leak testing capability, isolation valve position indicator in control room, and high pressure alarm. Interlocks prevent isolation valve opening when RCS pressure exceeds RSDC system design pressure (PWR). A RSDC designed for full reactor pressure (BWR). Double isolation.	Pressure Relief A Design pressure such that full RCS pressure is below rupture pressure and no leaks will occur which exceed RCS makeup capacity.

The acceptability of ALWR requirements to address containment challenges was based on the following criteria:
1. Current LWR resistance to challenge acceptable for ALWR.
2. Sufficient ALWR design features added to increase resistance to challenge by reducing the severity and/or ensuring containment.

* Passive plant design features which exceed requirements for current LWRs are identified with A (common to all ALWRs) or P (passive ALWRs only).

Table 4 (continued)

SUMMARY OF REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES THAT ARE INDEPENDENT OF OR COINCIDENT WITH CORE DAMAGE

CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE	ALWR BASIS#	KEY PASSIVE ALWR REQUIREMENTS: LIMIT POTENTIAL FOR CHALLENGE*	KEY PASSIVE ALWR REQUIREMENTS: ACCOMMODATE CHALLENGES*
3. Blowdown Forces	Containment Pressure Control	PWR/BWR	1	Design and ISI in accordance with ASME BPV Code. Leak Before Break.	Design containment for double ended guillotine break of largest pipe.
4. Pipe Whip and Jet Impingement	Bypass	PWR/BWR	1	Design and ISI in accordance with ASME BPV Code. Leak Before Break. Use of only proven materials and fabrication processes. Use of EPRI water chemistry guidelines.	Protection from jet/pipe whip where leak before break is not demonstrated.
5. Steam Generator Tube Rupture	Bypass	PWR	2	Improved water chemistry. Proven materials. A Mechanical design of tubes, tube supports, and tube sheets reduce likelihood of SGTR. A Improved design features facilitate SG cleaning and replacement.	P Operator actions can terminate leakage prior to ADS actuation for design basis leak. P Automatic Depressurization System (ADS) operation terminates tube leakage automatically. P Passive RHR plus additional features prevent secondary side relief following SGTR.
6. ATWS	Reactivity Control	BWR	2	A Diverse Reactor Protection System (RPS). A Diverse means of rod insertion.	Standby Liquid Control (SLC). A Checkerboard pattern of scram group rods maximizes group worth.
		PWR	2	A Diverse RPS (or capability to ride out ATWS).	Borated Safety Injection (SI). A Negative moderator temperature coefficient over entire fuel cycle improves ATWS response.

The acceptability of ALWR requirements to address containment challenges was based on the following criteria.
1. Current LWR resistance to challenge acceptable for ALWR.
2. Sufficient ALWR design features added to increase resistance to challenge by reducing the severity and/or ensuring containment.

* Passive plant design features which exceed requirements for current LWRs are identified with A (common to all ALWRs) or P (passive ALWRs only).

Table 4 (continued)

SUMMARY OF REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES THAT ARE INDEPENDENT OF OR COINCIDENT WITH CORE DAMAGE

				KEY PASSIVE ALWR REQUIREMENTS	
CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE	ALWR BASIS#	LIMIT POTENTIAL FOR CHALLENGE*	ACCOMMODATE CHALLENGES*
7. Suppression Pool Bypass	Containment Pressure Control	BWR	2	Vacuum Breakers: potential loads accounted for, position indication, minimal leakage. P No high energy lines in wetwell airspace.	ADS use of SRVs which discharge to suppression pool and thus ensure vapor suppression despite leakage. P Passive RHR (including PCCS)
8. Catastrophic RPV Failure	Internal Containment Loading	PWR/BWR	2	A $RT_{NDT} \leq 10^{o}F$; initial $RT_{NDT} \leq 20^{o}F$ for PWR core beltline; low fluence at vessel wall. A No welds in beltline region. A Relief valves prevent overpressure, backed up by depressurization system and low head injection. Design in accordance with ASME code. Design features to avoid relief valve opening for expected plant transients.	
9. Internal Vacuum	Containment Pressure Control	PWR/BWR	1		Vacuum Breakers. Design for external pressure loads.
10. Internal (Plant) Missiles	External Containment Loading	PWR/BWR	2	Turbine overspeed protection. A Improved turbine integrity/one-piece rotors.	Turbine orientation avoids missile contact with containment. Missile protection for any safety related components in missile path (SRP 3.5.1.3)
11. Tornado and Tornado Missiles	External Containment Loading	PWR/BWR	2	Conformance with ANSI 2.12 and ANSI 51.5.	P Passive core cooling systems located within containment.

The acceptability of ALWR requirements to address containment challenges was based on the following criteria:
1. Current LWR resistance to challenge acceptable for ALWR.
2. Sufficient ALWR design features added to increase resistance to challenge by reducing the severity and/or ensuring containment.

* Passive plant design features which exceed requirements for current LWRs are identified with A (common to all ALWRs) or P (passive ALWRs only).

Table 4 (continued)

SUMMARY OF REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES THAT ARE INDEPENDENT OF OR COINCIDENT WITH CORE DAMAGE

				KEY PASSIVE ALWR REQUIREMENTS	
CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE	ALWR BASIS#	LIMIT POTENTIAL FOR CHALLENGE*	ACCOMMODATE CHALLENGES*
12. Man-Made Site Proximity Hazards	External Containment Loading	PWR/BWR	2	Conformance with ANSI 2.12	P Passive core cooling systems located within containment
13. Seismic	External Containment Loading	PWR/BWR	2	Siting requirements exclude the most vulnerable sites.	A SSE at 0.3g A Evaluation at > SSE with PRA or margins assessment as part of design process A Address vulnerabilities from past experiences, e.g., provide common basemat

The acceptability of ALWR requirements to address containment challenges was based on the following criteria:

1. Current LWR resistance to challenge acceptable for ALWR.
2. Sufficient ALWR design features added to increase resistance to challenge by reducing the severity and/or ensuring containment.

* Passive plant design features which exceed requirements for current LWRs are identified with A (common to all ALWRs) or P (passive ALWRs only)

Table 4a

SUMMARY OF REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES RESULTING FROM CORE DAMAGE

CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE	ALWR BASIS#	KEY PASSIVE ALWR REQUIREMENTS: LIMIT POTENTIAL FOR CHALLENGE*	KEY PASSIVE ALWR REQUIREMENTS: ACCOMMODATE CHALLENGES*
14. High Pressure Melt Ejection (HPME)	Reactor Pressure Control	BWR	2	P Diverse depressurization systems. P Passive RHR can aid depressurization.	Suppression pool cools heated gases. Inerted containment (no combustion heat addition).
		PWR	2	P Diverse depressurization systems. P Passive RHR can aid depressurization.	A Cavity configuration to limit transport of fragmented core debris.
15. Hydrogen Generation to Detontable Limits	Combustible Gas Control	BWR	1	Inerted.	A Evaluation required if local detonation is possible.
		PWR	2	A Limit H_2 generation with design features, such as ADS and cavity flooding. A Hydrogen control system (e.g., non safety related igniters) designed to keep hydrogen concentration below 10% for 100% active clad equivalent reaction. A Containment size prevents global detonable H_2 concentration (< 13%) for generation up to 75% active clad equivalent reaction. A Design ensures convective mixing and minimizes DDT-prone geometry.	A Evaluation required if local detonation is possible.

The acceptability of ALWR requirements to address containment challenges was based on the following criteria:
1. Current LWR resistance to challenge acceptable for ALWR.
2. Sufficient ALWR design features added to increase resistance to challenge by reducing the severity and/or ensuring containment.

* Passive plant design features which exceed requirements for current LWRs are identified with A (common to all ALWRs) or P (passive ALWRs only).

Table 4a (continued)

SUMMARY OF REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES RESULTING FROM CORE DAMAGE

CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE	ALWR BASIS#	KEY PASSIVE ALWR REQUIREMENTS: LIMIT POTENTIAL FOR CHALLENGE*	KEY PASSIVE ALWR REQUIREMENTS: ACCOMMODATE CHALLENGES*
Hydrogen Deflagration	Combustible Gas Control	BWR	1	Inerted	A Demonstrated accommodation of generation equivalent to 100% active clad reaction A Structural evaluation for LOCA plus hydrogen loads (75% active clad reaction).
		PWR	2	A Deflagration likely at low concentrations (< 10%) given hydrogen control system (IRWST and PCCS limit steam inerting potential).	A Demonstrated accommodation of generation equivalent to 100% active clad reaction with multiple burns. A Structural evaluation for LOCA plus hydrogen loads, including global burn of hydrogen equivalent to 75% active clad reaction.
16. In-Vessel Debris-Water Interaction	Internal Containment Loading	BWR/PWR	1	Large-scale phenomena limited in probability. In-vessel geometry limits interacting quantities and size of any interaction	Rugged reactor vessel contains forces; as backup, rugged lower drywell/reactor cavity contains lower head failure.
17. Ex-Vessel Debris-Water Interaction	Internal Containment Loading	BWR/PWR	2	Large-scale phenomena limited in probability. Ex-vessel geometry limits interacting quantities and size of any interaction.	A Rugged lower drywell/reactor cavity confirmed by evaluation. Containment design accommodates steam generation

The acceptability of ALWR requirements to address containment challenges was based on the following criteria:
1. Current LWR resistance to challenge acceptable for ALWR.
2. Sufficient ALWR design features added to increase resistance to challenge by reducing the severity and/or ensuring containment.

* Passive plant design features which exceed requirements for current LWRs are identified with A (common to all ALWRs) or P (passive ALWRs only).

Table 4a (continued)

SUMMARY OF REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES RESULTING FROM CORE DAMAGE

CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE	ALWR BASIS#	KEY PASSIVE ALWR REQUIREMENTS: LIMIT POTENTIAL FOR CHALLENGE*	KEY PASSIVE ALWR REQUIREMENTS: ACCOMMODATE CHALLENGES*
18. Noncondensible Gas Generation	Fuel/Debris Cooling	BWR/PWR	2	A Features limiting concrete erosion (see item 19) limit noncondensible gas generation as well. A Sacrificial concrete specified as low gas generation type. A Overlying pool cools gases from core concrete interaction.	Containment size and pressure retention capability.
19. Basemat Erosion and Vessel Support Degradation	Fuel Debris Cooling	BWR/PWR	2	A Reactor cavity/lower drywell spreading area of $0.02 m^2/MWt$ promotes core debris cooling. A Lower drywell/cavity flooding. A Lower drywell flooding thermally actuated direct from BWR gravity drain tank or suppression pool. A Overflow from containment reflux via PWR IRWST prefloods reactor cavity. A Backup capability for water addition from sources external to containment.	A Sacrificial concrete where debris on floor contacts boundary structures (which are the passive BWR vessel support)
20. Core Debris in Sump	Fuel/Debris Cooling	BWR/PWR	2	A Special cavity sump design prevents localized unterminated core-concrete interaction. A Sump drainline configuration precludes gravity transport of debris ex-containment. A Reactor cavity/lower drywell flooding	

\# The acceptability of ALWR requirements to address containment challenges was based on the following criteria:
1. Current LWR resistance to challenge acceptable for ALWR.
2. Sufficient ALWR design features added to increase resistance to challenge by reducing the severity and/or ensuring containment.

* Passive plant design features which exceed requirements for current LWRs are identified with A (common to all ALWRs) or P (passive ALWRs only).

Table 4a (continued)

SUMMARY OF REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES RESULTING FROM CORE DAMAGE

CHALLENGE	AFFECTED SAFETY FUNCTION	PLANT TYPE	ALWR BASIS#	KEY PASSIVE ALWR REQUIREMENTS: LIMIT POTENTIAL FOR CHALLENGE*	KEY PASSIVE ALWR REQUIREMENTS: ACCOMMODATE CHALLENGES*
21. Core Debris Contact With Liner	Fuel/Debris Cooling	BWR/PWR	2	A Liner protected by concrete. A Lower drywell/cavity flooding. A Design features to limit debris dispersal including ADS.	
22. Decay Heat Generation	Containment Pressure Control	BWR	2	Main Condenser. A Reactor Water Cleanup System P Passive RHR (RCS heat removal mode)	P Passive Containment Cooling
		PWR	2	Steam Generators/Main Feedwater (MFW)/Backup Feedwater. Reactor Shutdown Cooling	P Passive Containment Cooling P Passive Heat Removal through containment shell without PCCS water limits containment pressure.
23. Tube Rupture from Hot Gases	Bypass	PWR	2	Steam Generators/MFW/Backup Feedwater A Depressurization System.	

The acceptability of ALWR requirements to address containment challenges was based on the following criteria:
1. Current LWR resistance to challenge acceptable for ALWR.
2. Sufficient ALWR design features added to increase resistance to challenge by reducing the severity and/or ensuring containment.

* Passive plant design features which exceed requirements for current LWRs are identified with A (common to all ALWRs) or P (passive ALWRs only)

APPENDIX C

ALWR Design Characteristics And Features Which Address Dominant WASH 1400 And Subsequent PRA Accident Sequences And Failure Modes

Appendix C

ALWR Design Characteristics and Features which Address Dominant WASH 1400 and Subsequent PRA Accident Sequences and Failure Modes

LOCA

- No recirculation piping in BWR; minimal number of welds in RCS piping in PWR
- RCS depressurization system allows low pressure systems to be effective regardless of the break size.
- It is unnecessary to switch to recirculation since passive containment heat removal condenses steam released into containment and returns it to the vessel by gravity.
- Safety system dependencies essentially eliminated (include only dc power for the purpose of depressurization).

Vessel Rupture

- Reduced RCS peak pressure for plant transients.
- Improved materials:
  - Less than .012% phosphorus, weld and base metal
  - Less than .03% copper, PWR base metal
  - Less than .05% copper, BWR base metal
  - Less than .08% copper, weld metal
  - Less than .05% vanadium, weld metal
- Initial ductility transition reference temperature less than 10°F (less than -20°F for PWR core belt region), reference temperature shift less than 30°F over plant life.
- Low fluence at vessel wall.
- No welds in beltline region.

Interfacing System LOCA

- Low pressure systems normally isolated from the RCS are provided with interlocks to prevent their exposure to RCS pressure and are enunciated should high pressure conditions occur.
- The ultimate rupture strength of potential interfacing systems is capable of withstanding full RCS pressure.

Transient (loss of injection)

- Core passive residual heat removal system automatically actuates on loss of ac power. Passive system is fail safe and can operate independent of any support system.

- Automatic depressurization and gravity injection are capable of providing adequate core cooling independent of normal makeup systems and passive residual heat removal system.

Transient (station blackout)

- Core passive residual heat removal system automatically actuates on loss of ac power. Passive system is fail safe and can operate independently from any support system.

- Automatic, backup ac power systems.

- Battery capacity in excess of 72 hours.

- Canned rotor reactor coolant pumps are provided in the PWR, eliminating the potential for seal LOCA (the BWR is natural circulation and has no recirculation pumps).

ATWS

- PWR capability to ride out an ATWS.

- PWR negative moderator temperature coefficient over entire operating cycle.

- PWR borated safety injection.

- BWR capability to mitigate short term ATWS effects and shutdown automatically by diverse means:

  - Safety relief valve capacity > 100% power
  - Motor drives diverse from hydraulic drive mechanisms
  - Auxiliary Rod Insertion system diverse from reactor protective system
  - Automatic Standby Liquid Control independent of all support systems except dc power

Shutdown Risk

- Permanent, operable, redundant water level instrumentation designed for use during shutdown conditions.

- Antisiphon provisions in refueling pool cooling and cleanup system piping to prevent pool drain down.

- Features to prevent or mitigate the effects of losing suction to decay heat removal pumps during shutdown condition (e.g., piping design to minimize vortexing and air entrainment).

- Features to assure required net positive suction head is always available to decay heat removal pumps.

- Passive decay heat removal systems are capable of removing decay heat and preventing RCS overpressure.

- Detailed requirements for analyses of mid-loop operation (PWRs) and low-level operation (BWRs) to provide assurance that known loss of shutdown cooling problems have been addressed and that information to operate the plant safely during shutdown has been developed.

- Provision of a separate power supply circuit to the plant permanent nonsafety leads for use in the event of extended unavailability of the normal power supply such as may occur during shutdown.

- Capability of closing valves for draining the reactor vessel or RCS without reliance on ac power.

- Limitations on boron dilution flow in PWRs such that the operator has at least 30 minutes after indication of dilution to terminate the incident prior to any recriticality.

<u>Overpressure (steam)</u>

- Passive containment cooling systems transfer heat directly from containment without dependence on support systems, the BWR through a heat exchanger in a water pool, the PWR directly through the containment steel shell.

<u>Overpressure (noncondensables) and Basemat Penetration</u>

- Reactor cavity/lower drywell configured to promote spreading of core debris to increase coolability.

- Ample water is available to cool debris in the reactor cavity/lower drywell passively, by means independent of potential causes for core damage.

<u>In-Vessel Steam Explosion</u>

- Containment failure due to in-vessel steam explosion was unlikely in WASH 1400, and has been reexamined several times since and is now considered to be extremely unlikely[25]. This is due to improved understanding of steam explosion phenomena, particularly the extent to which water depletion in the debris-water interaction zone (due to high heat transfer rates from debris fragments to the water and to the dispersive effect of the subsequent high steaming rates on the surrounding water pool) limits molten debris premixing and mechanical energy yield.

<u>Hydrogen Combustion</u>

- The BWR containment is inerted.

- The PWR containment is required to have a hydrogen control system. Even without crediting this system, the PWR containment is capable of withstanding a burn associated with hydrogen generated from oxidation of as much as 75% of the active fuel cladding without exceeding ASME Service Level C limits.

Containment Isolation

- The passive plants have fewer penetrations as a result of safety systems being located inside containment and other changes to reduce the number of penetrations.
- Most penetrations are isolated during power operation.
- Penetrations which may be open during power operation are fail safe or dc powered making them effectively independent of support systems.
- A periodic, on line leakage monitor is specified to avoid pre-existing opening.

Liner Melt-through

- Reactor cavity and lower drywell are configured to protect the containment boundary from direct contact by core debris.

Ex-Vessel Steam Explosion

- Similar to in-vessel steam explosions, water depletion in the debris-water interaction zone limits ex-vessel molten debris premixing and mechanical energy yield; also, voiding (i.e., steam content in the debris-water-steam system) limits pressure pulse propagation to structures.
- A rugged BWR reactor vessel foundation design is provided together with a URD requirement to demonstrate that ex-vessel debris water interactions will not cause loss of reactor vessel structural support.
- A shield is provided in the BWR lower drywell to protect the containment boundary from the effects of debris-water interactions.

Direct Containment Heating

- Both PWR and BWR have an automatic RCS depressurization system containing redundant trains and diversity in valve designs to prevent common cause failures. The depressurization systems require only dc power for operation.
- Passive decay heat removal systems are capable of reducing and maintaining the RCS at low pressures.
- Cavity/lower drywell configuration is such that much of the debris will be trapped as opposed to being entrained in the steam flow. Also, recent work suggests that any debris which is entrained is exposed to only a small fraction of the steam flow from the RCS, thus greatly limiting the potential for thermal/chemical interactions [26].

Overtemperature

- Automatic RCS depressurization system and RCS passive decay heat removal system minimize high pressure melt ejection and resulting core debris transport into upper drywell

- Ample water available in lower drywell to cool debris and avoid high temperatures

- BWR drywell spray to reduce temperatures

Steam Generator Tube Rupture (SGTR)

- Reduced primary coolant temperatures to reduce corrosion

- Improved water chemistry and tube materials (i.e., NiCrFe alloy 690 TT).

- Improved mechanical design of tubes and tube bundles.

- Passive RHR prevents need for secondary side relief and steam generator overfill.

- Automatic RCS depressurization terminates tube leakage with no operator action.

- Depressurized RCS minimizes convection of hot gases which could cause tube rupture.

APPENDIX D

Assessment of AP600 Design Conformance with ALWR Containment Requirements

Table D-1

ASSESSMENT OF AP600 DESIGN CONFORMANCE WITH ALWR REQUIREMENTS WHICH ADDRESS CONTAINMENT CHALLENGES

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS			
		LIMIT POTENTIAL FOR CHALLENGE[1]		ACCOMMODATE CHALLENGE[1]	
1. Containment Isolation	Isolation	P	• Reduced fluid line penetrations [6.2.3.2.1 & 6.2.3.1.3-A].	P	• Passive Residual Heat Removal (RHR) minimizes core damage risk given isolation failure (with RHR on-line even without DC power) [6.3].
			• Isolation provisions and leakage rate testing per standards [6.2.5.2.2].		
			• Valves capable of closure with maximum flow and full containment pressure [6.2.3.1.3-F].		
			• Control room position indication for automatic and remote manual valves [6.2.3.1.3-H,I].		
		P	• Manual valve configuration permits locking only in closed position [6.2.3.1.3-J].		
		P	• Closed systems penetrating containment evaluated for ex-vessel severe accidents [6.2.3.1.1-H].		
			• Fail closed or DC powered isolation valves [6.2.3.1.3-K].		
		P	• Capability for periodic gross check of containment integrity [2].		
2. Interfacing System LOCA	Bypass	P	• Reduced interfaces between the Reactor Coolant System (RCS) and low pressure systems [PRA App. A.3.2].		• Pressure Relief [5.1.2 & 5.2.2].
		P	• High to low pressure interfaces provided with isolation valve leak testing capability [6.2.5.2.2 & for RHR, Fig. 5.4-7], isolation valve position indicator in control room [6.2.3.1.3-H, I & for RHR see note 2], and high pressure alarm [RHR, 7.6.1.1.1].	P	•Design pressure such that full RCS pressure is below rupture pressure and no leaks will occur which exceed RCS makeup capacity [5.4.7.2 & PRA App. A.3.2].
			• Interlocks prevent isolation valve opening when RCS pressure exceeds RSDC system design pressure [5.4.7.2.2].		
			• Double isolation [5.4.7.2.2].		

[1] Passive plant design features which exceed requirements for current LWRs are identified with a P.
[2] No reference in SSAR; however, Westinghouse has committed to this capability.

Table D-1
(Cont'd)

ASSESSMENT OF AP600 DESIGN CONFORMANCE WITH ALWR REQUIREMENTS WHICH ADDRESS CONTAINMENT CHALLENGES

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS			
		LIMIT POTENTIAL FOR CHALLENGE[1]		ACCOMMODATE CHALLENGE[1]	
3. Blowdown Forces	Containment Pressure Control		• Design and ISI in accordance with ASME BPV Code [5.2.1.1]. • Leak Before Break [5.1.3.4 & 3.6.1.1-P].		• Design containment for double-ended guillotine break of largest pipe [6.2.1.1.1].
4. Pipe Whip and Jet Impingement	Bypass		• Design and ISI in accordance with ASME BPV Code [5.2.1.1]. • Leak Before Break [5.1.3.4 & 3.6.1.1-P]. • Use of only proven materials and fabrication processes [5.2.3.1]. • Use of EPRI water chemistry guidelines [5.4.2.4.1].		• Protection from jet/pipe whip where leak before break is not demonstrated [3.6.1.1-C; 3.6.2.3.4.2 & 3.6.2.4.1].
5. Steam Generator Tube Rupture	Bypass	 P P	• Improved water chemistry [5.4.2.4.3]. • Proven materials [5.4.2.4.1]. • Mechanical design of tubes, tube supports, and tube sheets reduce likelihood of SGTR [5.4.2.3.3, 5.4.2.3.4 & 5.4.2.4.2]. • Improved design features facilitate SG cleaning and replacement [5.4.2 & 5.4.2.5].	P P P	• Operator actions can terminate leakage prior to ADS actuation for design basis leak [15.6.3]. • Automatic Depressurization System (ADS) operation terminates tube leakage automatically [15.6.3]. • Passive RHR prevent secondary side relief following SGTR [15.6.3].
6. ATWS	Reactivity Control	P	• Diverse RPS (or capability to ride out ATWS [4.3.1.7]) [PRA App. C12].	 P	• Borated Safety Injection (SI) [5.4.13]. • Negative moderator temperature coefficient over entire fuel cycle improves ATWS response [4.2.2.3].
7. Suppression Pool Bypass	Containment Pressure Control		NOT APPLICABLE		
8. Catastrophic RPV Failure	Internal Containment Loading	P P P	• $RT_{NDT} \leq 10°F$; initial $RT_{NDT} \leq -20°F$ for PWR core beltline; low fluence at vessel wall [5.3.3.1]. • No welds in beltline region [5.3.4.1]. • Relief valves prevent overpressure, backed up by depressurization system and low-head injection [5.4.9]. • Design in accordance with ASME code [5.3.1.1]. • Design features to avoid relief valve opening for expected plant transients [6.3.1.1.1 & 15.2.8.3].		

[1] Passive plant design features which exceed requirements for current LWRs are identified with a P.

Table D-1
(Cont'd)

ASSESSMENT OF AP600 DESIGN CONFORMANCE WITH ALWR REQUIREMENTS WHICH ADDRESS CONTAINMENT CHALLENGES

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS			
			LIMIT POTENTIAL FOR CHALLENGE[1]		ACCOMMODATE CHALLENGE[1]
9. Internal Vacuum	Containment Pressure Control				•Design for external pressure loads [3.8].
10. Internal (Plant) Missiles	External Containment Loading	P	• Turbine overspeed protection [10.2.2.3.6]. • Improved turbine integrity/one-piece rotors [10.2.3].		•Turbine orientation avoids missile contact with containment [3.5.1.3]. • Missile protection for any safety related components in missile path (SRP 3.5.1.3) [3.5].
11. Tornado and Tornado Missiles	External Containment Loading		• Conformance with ANSI 2.12 and ANSI 51.5 (in accordance with ASCE 7-88, "Minimum Design Loads for Buildings and other Structures," formerly ANSI A58.1-82) [3.3.1; 3.5.2 & 3.5.3].	P	• Passive core cooling systems located within containment [Fig. 6.3-5, 6.3-6, 6.3-7].
12. Man-Made Site Proximity Hazards	External Containment Loading		• Conformance with ANSI 2.12 [2].	P	• Passive Core cooling systems located within containment [Fig. 6.3-5, 6.3-6, 6.3-7].
13. Seismic	External Containment Loading		• Siting requirements exclude the most vulnerable sites [no effect on design].	P P P	• SSE at 0.3g [3.7.1]. • Evaluation at > SSE with margins assessment as part of design process [PRA App. H]. • Address vulnerabilities from past experience, e.g., provide common basemat [3.8.5.1].
14. High Pressure Melt Ejection (HPME)	Reactor Pressure Control	P P	• Diverse depressurization systems [5.1.2]. • Passive RHR can aid depressurization [6.3]. [See also PRA 10.2.2 & App. L.2.5].	P	• Cavity configuration to limit transport of fragmented core debris [PRA 10.2.3].

[1] Passive plant design features which exceed requirements for current LWRs are identified with a P.
[2] No reference in SSAR; however, Westinghouse has committed to this capability.

Table D-1
(Cont'd)

ASSESSMENT OF AP600 DESIGN CONFORMANCE WITH ALWR REQUIREMENTS WHICH ADDRESS CONTAINMENT CHALLENGES

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS			
		LIMIT POTENTIAL FOR CHALLENGE[1]		ACCOMMODATE CHALLENGE[1]	
15a. Hydrogen Generation to Detonable Limits	Combustible Gas Control	P P P P	• Limit H_2 generation with design features, such as ADS and cavity flooding [5.4.6 & 3.8.3.1.5]. • Hydrogen control system designed to keep hydrogen concentration below 10% for 100% active clad equivalent reaction [6.2.4]. • Containment size prevent global detonable H_2 concentration (<13%) for generation up to 75% active clad equivalent reaction [2]. • Design ensures convective mixing and minimizes DDT-prone geometry [6.2.4.1.1; PRA 10.2.5 & App. O].	P	• Evaluation required if local detonation is possible [PRA 14.0; 15.0; App. N & App. O].
15b. Hydrogen Deflagration	Combustible Gas Control	P	• Recombination or deflagration likely at low concentrations (<10%) given hydrogen control system (IRWST and PCCS limit steam inerting potential) [PRA App. N & App. O].	P P	•Demonstrated accommodation of generation equivalent to 100% active clad reaction with multiple burns [PRA App. N]. • Structural evaluation for LOCA plus hydrogen loads, including global burn of hydrogen equivalent to 75% active clad reaction [PRA App. N.4.8].
16. In-Vessel Debris-Water Interaction	Internal Containment Loading		• Large-scale phenomena limited in probability [PRA 10.2.1]. • In-vessel geometry limits interacting quantities and size of any interaction [PRA 10.2.1].		• Rugged reactor vessel contains forces [PRA 10.2.1]; as backup, rugged reactor cavity contains lower head failure [PRA 10.2.1].
17. Ex-Vessel Debris-Water Interaction	External Containment Loading		• Large-scale phenomena limited in probability [PRA 10.2.1]. • Ex-vessel geometry limits interacting quantities and size of any interaction [PRA 10.2.1].	P	• Rugged reactor cavity confirmed by evaluation [PRA 10.2.1]. • Containment design accommodates steam generation [PRA 10.2.1].
18. Noncondensible Gas Generation	Fuel/Debris Cooling	P P	• Features limiting concrete erosion (see item 19) limit noncondensible gas generation as well. • Overlying pool cools gases from core-concrete interaction [PRA 10.2.4].		• Containment size and pressure retention capability [PRA 10.2.4 & App. I].

[1] Passive plant design features which exceed requirements for current LWRs are identified with a P.
[2] No reference in SSAR; however, Westinghouse has committed to this capability.

Table D-1
(Cont'd)

ASSESSMENT OF AP600 DESIGN CONFORMANCE WITH
ALWR REQUIREMENTS WHICH ADDRESS CONTAINMENT CHALLENGES

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR OR PRA SECTIONS			
		LIMIT POTENTIAL FOR CHALLENGE[1]		ACCOMMODATE CHALLENGE[1]	
19. Basemat Erosion and Vessel Support Degradation	Fuel/Debris Cooling	P P P P	• Reactor cavity/lower drywell spreading area of $0.02 m^2/MWt$ promotes core debris cooling [PRA 10.2.4]. • Reactor cavity flooding [PRA 10.2.2]. • Overflow from containment reflux via PWR IRWST prefloods reactor cavity [PRA 10.2.2]. • Backup capability for water addition from sources external to containment [PRA App. C.4.4.1].	P	• Sacrificial concrete where debris on floor contacts boundary structures [3.8.2.1.2].
20. Core Debris in Sump	Fuel/Debris Cooling	P P P	• Special cavity sump design prevents localized unterminated core-concrete interaction [10.2.4]. • Sump drainline configuration precludes gravity transport of debris ex-containment [PRA 10.2.4]. • Reactor cavity flooding [PRA 10.2.2].		
21. Core Debris Contact with Liner	Fuel/Debris Cooling	P P P	• Liner protected by concrete [3.8.2.12]. • Reactor cavity flooding [PRA 10.2.2]. • Design features to limit debris dispersal including ADS [5.4.6 & 3.8.3.1.5].		
22. Decay Heat Generation			• Steam Generators/Main Feedwater (MFW)/Startup Feedwater [10.4.9]. • Normal Residual Heat Removal System [5.4.7].	P P	• Passive Containment Cooling [6.2.2]. • Passive Heat Removal through containment shell without PCCS water limits containment pressure [PRA App. L.3.1 & L.3.2].
23. Tube Rupture from Hot Gases		 P	• Steam Generators/MFW/Startup Feedwater [10.4.9]. • Depressurization System [5.1.2]		

[1] Passive plant design features which exceed requirements for current LWRs are identified with a P.

Table D-2

Exceptions for AP600 Design Conformance With ALWR Requirements

1. No SSAR provision exists for periodic gross check of containment integrity. However, Westinghouse has committed to the ALWR Program to provide this capability in AP600. The ALWR Program will track this item.

2. No SSAR requirement exists for a high pressure alarm on the high-to-low pressure interface for the Primary Sampling System and the Chemical Volume and Control System (CVCS). The ALWR Program will track this item.

3. An inconsistency exists between SSAR Section 7.6.1.1.1, which identifies a high pressure alarm on the low pressure side of the RHR System, and Figure 5.4-7 which does not show it. Westinghouse has confirmed in response to an NRC Request for Additional Information that the high pressure alarm is part of the system and that Figure 5.4-7 will be corrected.

4. The existence of isolation valve position indication for the RHR System is not mentioned in the SSAR, but Westinghouse confirmed that this capability is provided in the design.

5. No SSAR commitment to ANSI 2.12[27] exists for man-made site proximity hazards. However, Westinghouse has stated that AP600 will conform to ANSI 2.12. The ALWR Program will track this item.

6. No explicit statement is made in the SSAR regarding containment size being large enough to limit dry hydrogen concentration to less than 13% given 75% active clad oxidation. However, the ALWR Program has evaluated hydrogen concentration based on AP600 zircaloy mass and containment volume, and has concluded that the 13% requirement is met.

7. The SSAR does not currently specify low gas generation concrete in the reactor vessel cavity. However, based on sensitivity studies for ex-vessel debris coolability, the intent of the requirement is met, i.e., avoid rapid containment overpressure due to noncondensable gas generation, even under very conservative molten core concrete interaction assumptions.

APPENDIX E

Assessment of SBWR Design Conformance with ALWR Containment Requirements

Table E-1

ASSESSMENT OF SBWR DESIGN CONFORMANCE WITH
ALWR REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR SECTIONS			
		LIMIT POTENTIAL FOR CHALLENGE[1]		ACCOMMODATE CHALLENGE[1]	
1. Containment Isolation	Isolation	P P P P	• Reduced fluid line penetrations [2]. • Isolation provisions and leakage rate testing per standards [6.2.4.1]. • Valves capable of closure with maximum flow and full containment pressure [6.2.4.2.5]. • Control room position indication for automatic and remote manual valves [6.2.4.2]. • Manual valve configuration permits locking only in closed position [3]. • Closed systems penetrating containment evaluated for ex-vessel severe accidents [19.B.5.2.1]. • Fail closed or DC powered isolation valves [6.2.4.1]. • Capability for periodic gross check of containment integrity [3.6.1.4].	P	• Passive Residual Heat Removal (Isolation Condenser System [ICS]) minimizes core damage risk given isolation failure (with ICS on-line even without DC power) [5.4.6 & 5.4.6.2].
2. Interfacing System LOCA	Bypass	P P P	• Reduced interfaces between the Reactor Coolant System (RCS) and low pressure systems [7.6.1.1, 19H.2.25 & 19H.2.44]. • High to low pressure interfaces provided with isolation valve leak testing capability [9.1.3.2.2 for LPCI], isolation valve position indicator in control room [9.1.3.2.2 for LPCI], and high pressure alarm [3]. • RWCU/SDC designed for full reactor pressure [5.4.8.1.2]. • Double isolation [7.6.1.2].	 P	• Pressure Relief [6.3.3]. • Design pressure such that full RCS pressure is below rupture pressure [9.1.3.2.2 for LPCI]

[1] Passive plant design features which exceed requirements for current LWRs are identified with a P.

[2] No reference in SSAR; however, General Electric has committed to this capability

[3] No reference in SSAR

Table E-1

ASSESSMENT OF SBWR DESIGN CONFORMANCE WITH
ALWR REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES
(Cont'd)

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR SECTIONS			
		LIMIT POTENTIAL FOR CHALLENGE[1]		ACCOMMODATE CHALLENGE[1]	
3. Blowdown Forces	Containment Pressure Control		• Design and ISI in accordance with ASME BPV Code [5.2.1]. • Leak Before Break [3.6.3].		• Design containment for double-ended guillotine break of largest pipe [6.2.1.1.3].
4. Pipe Whip and Jet Impingement	Bypass		• Design and ISI in accordance with ASME BPV Code [5.2.1]. • Leak Before Break [3.6.3]. • Use of only proven materials & fabrication processes [Tables 5.2.1 & 5.2-4] • Use of EPRI water chemistry guidelines [5.2.3.2.2].		• Protection from jet/pipe whip where leak before break is not demonstrated [3.6.1.3 & 3.6.3].
5. Steam Generator Tube Rupture	Bypass		NOT APPLICABLE		
6. ATWS	Reactivity Control	P P	• Diverse Reactor Protection System and Alternate Rod Injection [4.6.1.2.5, 7.2.1, 7.4.5]. • Diverse means of rod insertion [4.6.1.2.1].	P	• Standby Liquid Control (SLC) [9.3.5]. • Checkerboard pattern of scram group rods maximizes group worth [2].
7. Suppression Pool Bypass	Containment Pressure Control	P	• Vacuum Breakers: potential loads accounted for, position indication, minimal leakage [6.2.1.1.2 & 1.9.4.4.1.11]. • No high energy lines in wetwell airspace [Figures 5.2-1 and 21.1.2-2].	P	•ADS use of SRVs which discharge to suppression pool and thus ensure vapor suppression despite leakage [5.2.2 & 6.3.3.2]. • Passive ICS (including PCCS) [5.4.6 & 6.2.2].
8. Catastrophic RPV Failure	Internal Containment Loading	P P P	• $RT_{NDT} \leq 10°F$; low fluence at vessel wall [5.3.2.1] • No welds in beltline region [5.3.3.3]. • Relief valves prevent overpressure, backed up by depressurization system and low-head injection [5.2.2, 6.3.2, & 6.3.3]. • Design in accordance with ASME code [5.3.1.1]. • Design features to avoid relief valve opening for expected plant transients [19AE.8.3.2].		
9. Internal Vacuum	Containment Pressure Control		• Containment internal design loads specifications, Table 6.2.1 through 6.2.6.		• Vacuum Breakers [6.2.1.1.2]. • Design for external pressure loads [6.2.1.1.2 & 6.2.1.1.3].

[1] Passive plant design features which exceed requirements for current LWRs are identified with a P.
[2] No reference in SSAR; however, General Electric has committed to this capability.

Table E-1

ASSESSMENT OF SBWR DESIGN CONFORMANCE WITH ALWR REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES

(Cont'd)

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR SECTIONS			
		LIMIT POTENTIAL FOR CHALLENGE[1]		ACCOMMODATE CHALLENGE[1]	
10. Internal (Plant) Missiles	External Containment Loading	P	• Turbine overspeed protection [10.2.2.4]. • Improved turbine integrity/one-piece rotors [10.2.3 & 10.2.3.4].		• Turbine orientation avoids missile contact with containment [3.5.1.1.1]. • Missile protection for any safety related components in missile path (SRP 3.5.1.3) [3.5.1 & 3.5.3].
11. Tornado and Tornado Missiles	External Containment Loading		• Conformance with ANSI 2.12 & ANSI 51.5 (in accordance with ANSI A58.1 and ASCE Paper Number 3269) [ ].	P	• Passive core cooling systems located within containment [5.4.6 & 6.2.2].
12. Man-Made Site Proximity Hazards	External Containment Loading		• Conformance with ANSI 2.12 [2].	P	• Passive core cooling systems located within containment [5.4.6 & 6.2.2].
13. Seismic	External Containment Loading		• Siting requirements exclude the most vulnerable sites [no effect on design].	P P P	• SSE of 0.3g [2.5.2 & 3.7.1.1]. • Evaluation at > SSE with margins assessment as part of design process [19D]. • Address vulnerabilities from past experience, e.g., provide common basemat [3.8.4.1].
14. High Pressure Melt Ejection (HPME)	Reactor Pressure Control	P P	• Diverse depressurization systems [6.3.3 & 19.4.4.1.5]. • Passive ICS can aid depressurization [5.4.6.3].		• Suppression pool cools heated gases [6.2.1.1.2 & 19.4.4.1.4]. • Inerted containment (no combustion heat addition) [9.4.8 & 19.4.4.1.10].
15a. Hydrogen Generation to Detonable Limits	Combustible Gas Control		• Inerted [9.4.8, 19.4.3.4 & 19.4.4.1.10].	P	• Evaluation required if local detonation is possible [19.4.3.4].

[1] Passive plant design features which exceed requirements for current LWRs are identified with a P.

[2] No reference in SSAR; however, GE has committed to this capability.

Table E-1

ASSESSMENT OF SBWR DESIGN CONFORMANCE WITH
ALWR REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES
(Cont'd)

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR SECTIONS			
		LIMIT POTENTIAL FOR CHALLENGE[1]		ACCOMMODATE CHALLENGE[1]	
15b. Hydrogen Deflagration	Combustible Gas Control		• Inerted. [9.4.8 & 19.4.4.1.10].	P P	• Demonstrated accommodation of generation equivalent to 100% active clad reaction [19B.3.2.5 & 19.G.2.45]. • Structural evaluation for LOCA plus hydrogen loads (100% active clad reaction) [19B.3.2.5 & 19.G.2.45].
16. In-Vessel Debris-Water Interaction	Internal Containment Loading		• Large-scale phenomena limited in probability [19BB]. • In-vessel geometry limits interacting quantities and size of any interaction [19BB].		• Rugged reactor vessel contains forces; as backup, rugged lower drywell contains lower head failure [19.4.2.2].
17. Ex-Vessel Debris-Water Interaction	External Containment Loading		• Large-scale phenomena limited in probability [19.4.3.3 & 19BB.5]. • Ex-vessel geometry limits interacting quantities and size of any interaction [19BB.2.1].	P	• Rugged lower drywell confirmed by evaluation [19.4.2.2, 19B.3.2.5 & 19BB.3.4]. • Containment design accommodates steam generation [19BB.5.4].
18. Noncondensible Gas Generation	Fuel/Debris Cooling	P P	• Features limiting concrete erosion (see item 19) limit noncondensible gas generation as well. • Overlying pool cools gases from core-concrete interaction [19AE.7.1].		• Containment size and pressure retention capability [19B.6.2.1 & 19B.6.2.2].
19. Basemat Erosion and Vessel Support Degradation	Fuel/Debris Cooling	P P P P	• Lower drywell spreading area of $0.02 m^2/MWt$ promotes core debris cooling [19.4.4.1.8]. • Lower drywell flooding [6.2.1.1.2 & 19.4.4.1.3]. • Lower drywell flooding thermally actuated directly from gravity drain tank or suppression pool [6.3.2.2 & 19.4.4.1.3]. • Backup capability for water addition from sources external to containment [9.1.3.1].	P	• Sacrificial concrete where debris on floor contacts boundary structures (which are the passive BWR vessel support) [19.4.4.1.7].

[1] Passive plant design features which exceed requirements for current LWRs are identified with a P.

Table E-1

ASSESSMENT OF SBWR DESIGN CONFORMANCE WITH
ALWR REQUIREMENTS TO ADDRESS CONTAINMENT CHALLENGES
(Cont'd)

CHALLENGE	AFFECTED SAFETY FUNCTION	KEY ALWR REQUIREMENTS AND ASSOCIATED SSAR SECTIONS			
		LIMIT POTENTIAL FOR CHALLENGE[1]		ACCOMMODATE CHALLENGE[1]	
20. Core Debris in Sump	Fuel/Debris Cooling	P P P	• Special cavity sump design prevents localized unterminated core-concrete interaction [2]. • Sump drainline configuration precludes gravity transport of debris ex-containment [2]. • Lower drywell flooding [6.2.1.1.2 & 19.4.4.1.3].		
21. Core Debris Contact with Liner	Fuel/Debris Cooling	P P P	• Liner protected by concrete [19.4.4.1.7]. • Lower drywell flooding [6.2.1.1.2 & 19.4.4.1.3]. • Design features to limit debris dispersal including ADS [6.3.3 (ADS), 19B.10.2.4 & 19BB.3.3 (corium shield)].		
22. Decay Heat Generation	Containment Pressure Control	 P P	• Main Condenser [5.4.7]. • Reactor Water Cleanup System (RWCS) [5.4.7 & 5.4.8]. • Passive RHR (RCS heat removal mode) [5.4.6 & 5.4.7].	P	• Passive Containment Cooling [6.2.2 & 19.4.4.1.2].
23. Tube Rupture from Hot Gases	Bypass		NOT APPLICABLE		

[1] Passive plant design features which exceed requirements for current LWRs are identified with a P.

[2] No reference in SSAR; GE is currently considering this requirement.

Table E-2

Exceptions for SBWR Conformance Assessment With ALWR Requirements

1. The fact that the number of containment fluid line penetrations has been reduced is not explicitly mentioned in the SSAR, but General Electric has stated that this is the case in the SBWR design.

2. There is no SSAR provision that manual containment isolation valves permit locking only in the closed position. The ALWR Program will track this item.

3. No SSAR reference to isolation valve leak testing capability and position indication in control room was identified for sampling lines. General Electric indicated that this capability exists. The ALWR Program will confirm this item as part of conformance assessment.

4. No SSAR reference to high pressure alarms for high to low pressure interfaces was identified. The ALWR Program will track this item.

5. There is no SSAR requirement for a checkerboard pattern of control rods within a scram group to maximize group worth, but General Electric has stated that this provision is in the SBWR design (it became standard practice in recent operating BWRs).

6. Conformance with ANSI 2.12[27] for man-made site proximity hazards was not identified in the SSAR, although the design approach appears consistent with ANSI 2.12. The ALWR Program will track this item.

7. No provision currently exists in the SSAR for cavity sump and sump drainline design to prevent localized core concrete interaction and ex-containment gravity transport of core debris in the event of ex-vessel core damage. General Electric is considering design features in this regard. The ALWR Program will track this item.