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Electric Power
Research Institute ______ Leadership in Science and Technology

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Subject: **Project 669**

In response to your April 24, 1992 letter transmitting the Draft Safety Evaluation Report (DSER) for the Passive Plant Utility Requirements Document, Enclosure 1 provides a data base of the issues identified for Chapters 1, 1A and 12. This data base provides a "road map" for issue resolution. Hand-marked pages to Volume III of the Requirements Document are included for issues where changes are proposed to address NRC concerns.

Enclosure 2 provides a list of the issues in Chapters 1 and 1A which have responses outstanding (all Chapter 12 responses are provided). Our remaining responses for the DSER issues will be provided in the future. Please call John D. Trotter at 415/812-2810 if you have any questions.

Sincerely,

G Bockhold

George Bockhold Jr.
Senior Manager
Advanced LWR Program

cc: J. [illegible] Wilson/NRC

Enclosures: 2

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Enclosure 1

Passive Plant DSER Issue Data Base

ALWR/NRC OPEN ISSUES

P.1.O-1

Status: Open

Next Action: ALWR

scope of mitigation requirements (2.1, 2.4)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.2-1) "It is not clear to the staff where EPRI places its "significant additional emphasis" on mitigation. There are many examples throughout the Passive Requirements Document about which the staff concludes that the passive plant design criteria place less emphasis on the mitigation of design-basis accidents (DBAs) than do criteria for current plants." (DSER, p 1.2-16) "However, it does not include requirements for fission-product control or hydrogen control during design-basis loss-of-coolant accidents. EPRI should either include these two issues or justify their exclusion."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/SPLB J. Kudrick

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.O-2 | III.A | Status: Open | Next Action: ALWR/NRC

regulatory treatment of non-safety-related systems (2.3.1, 4.3.1, 7, 10, 12.2.1, 12.2.3, Appendix B)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.2-3) "the requirement regarding a non-safety-related onsite ac power source affects the reliability and availability of those non-safety-related active systems that provide defense-in-depth functions. This issue is enveloped for the passive designs under the issue pertaining to the regulatory treatment of non-safety-systems."	See Policy Issue III.A	(DSER) See Abstract	See Policy Issue III.A

(DSER, p 1.4-2) "Since these important non-safety-related systems are not required by EPRI to meet safety-grade criteria, the staff is trying to establish functional performance requirements, acceptance criteria, and other appropriate design guidelines to ensure that such systems have adequate functional capability and will remain operable when called on. Therefore, the staff's positions on quality group classifications of specific structures, components, and equipment may not be available until the above criteria have been established."

(DSER, p 1.7-2) "The staff concludes that pertinent quality assurance provisions should be applied to these activities and items. This issue is part of the overall issue regarding the regulatory treatment of non-safety-related systems for passive plants ..."

(DSER, p 1.12-3) "The specific staff positions on the inservice testing requirements for the essential non-safety-related components will be determined when the staff completes its review of the issue of regulatory treatment of non-safety-grade systems."

(DSER, p 1.12-11) "EPRI stated that the passive ALWR will not have safety-related pumps and that the staff's positions contained in the RAIs should not apply to non-safety-related pumps. As discussed in Section 12.2.1 of Chapter 1 of this report, the specific staff position on the inservice testing requirements for the essential non-safety-related components will be determined when the staff completes its review of the regulatory treatment of non-safety-related systems."

NRC Review

NRR/EMCB
NRR/EMEB
NRR/ESGB
NRR/LHFB
NRR/LOLB
NRR/LPEB
NRR/PDST
NRR/PEPB
NRR/PRAB
NRR/PRPB
NRR/RSGB
NRR/SELB
NRR/SICB
NRR/SPLB
NRR/SRXB
RES

Last Updated: 7/14/92

ALWR/NRC OPEN ISSUES

P.1.O-3		Status: Open	Next Action: NRC

automatic standby liquid control system for passive BWR design (2.3.2, Appendix B)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.2-9) "In its December 6, 1991, letter, EPRI stated that it has determined that automatic actuation of the SLCS was appropriate for evolutionary designs, and that it was modifying the Requirements Document for evolutionary plant designs to reflect that position. Although EPRI has not submitted its position regarding the automatic actuation of the SLCS for passive designs, the staff expects EPRI to provide design requirements that are consistent with those for the evolutionary designs regarding this matter."	The URD will be modified to require automatic SLCS. See Issue P.5.V-5	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/SRXB M. Rubin

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LICENSING AND REGULATORY REQUIREMENTS AND GUIDANCE

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2	PLANT OPTIMIZATION SUBJECTS	0
2.1	INTRODUCTION	0
	Plant optimization subjects are items for which the ALWR provides alternative criteria to satisfy the underlying basis for the [illegible], based on currently available technology. The technical bases for plant optimization subjects are provided in plant optimization subject papers including the ALWR position, discussion and assessment. The requirements identified in the Requirements Document are consistent with the positions taken in the plant optimization subject papers contained in this section.	0

LISTING OF OPTIMIZATION SUBJECTS FOR THE PASSIVE ALWR — 2

Section	*Lead Chapter*	*Title*	Rev.
			0
2.1.1	Chapter 1	Operating Basis Earthquake and Dynamic Analysis Methods	0
2.1.2	Chapter 1	Tornado Design	2
2.1.3	Chapter 1	Off-site Emergency Planning	2
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2.5.4	Chapter 5	Dedicated Containment Vent Penetration	0
~~2.5.5~~	~~Chapter 5~~	~~ATWS Mitigation for the Advanced BWR~~	~~0~~
2.5.6	Chapter 5	Safe Shutdown	0

Section No		Rev.
2.5.5	**ATWS MITIGATION FOR THE ADVANCED BWR**	0
2.5.5.1	**ALWR POSITION**	0
	The requirements for the Advanced BWR comply with the regulatory requirements for ATWS mitigation with the exception that the Standby Liquid Control System is manually initiated instead of automatically initiated. The Requirements Document provides requirements for the Advanced BWR that address the prevention and mitigation of ATWS as required by 10CFR50.62. The Advanced BWR is required to provide an alternate rod insertion system (ARI) that uses sensors and logic diverse and independent from the reactor protection system. In addition to the ARI, the Advanced BWR is required to provide both an electric and a hydraulic insertion capability for the control rod drives. A standby liquid control system initiated by manual action is required to be provided. The Passive BWR operates with natural circulation, thereby eliminating the need for recirculation pumps. Since there are no recirculation pumps, the requirements for an automatic pump trip are not applicable.	0
2.5.5.2	**DISCUSSION**	0
	A number of requirements for the Advanced BWR have been developed explicitly for the purpose of increased margins and reduced demands of engineered safety systems. Combined with a highly reliable reactor protection system and a CRD system with enhanced reliability, the significance of ATWS and its potential effects have been substantially reduced in the Advanced BWR. These requirements include:	0
	• More robust reactor coolant system design (e.g., larger water inventory) to accommodate transient conditions, thereby resulting in fewer transients and challenges requiring actuation of the RPS.	0
	• The reactor is required to have a negative power coefficient under all conditions.	0
	• Large relief capacities are required to be provided for all plant conditions, including safety grade depressurization.	0
	• A hydraulic CRD scram system that does not require a scram dump volume is required for the Advanced BWR.	0
	• The CRD hydraulic control units and associated rod pairs are divided into four independent scram groups. The combination of rods within each group are arranged so that hot shutdown can be achieved even in the event of failure to scram of an entire rod group.	0

Section No.		Rev.
2.5.5.2	**DISCUSSION (CONTINUED)**	0

- The electric insertion function is divided into three independent motor groups and the rods within the groups are arranged in a checker board pattern so that hot shutdown can be achieved even in the event of failure of any one motor group. 0

In the event of a transient requiring scram in the Advanced BWR, diverse means of achieving reactor shutdown have been provided. These include a hydraulic rod insertion system and an electric motor drive insertion system each of which can be initiated by the RPS or the ARI. The ARI provides signals to insert rods that are independent and diverse from the electrical RPS signals. The motor drives provide a mechanical diversity for rod insertion independent of the hydraulic portion of the CRD system. The ARI is in accordance with the requirement of 10CFR50.62(c)(3). The motor driven function provides CRD mechanical diversity to the hydraulic scram in addition to the electrical scram initiation diversity provided by the RPS and the ARI. 0

An additional diverse means of negative reactivity insertion is provided by the Standby Liquid Control (SLC) system. This system fulfills the requirements of General Design Criteria 26 for an independent reactivity control system of different design principals and has capacity in accordance with 10CFR50.62(c)(4). The need for initiating this system is significantly reduced by the enhanced reliability provided by the CRD system mechanical and electrical requirements for independence and diversity. The importance of automatic SLC initiation is significantly reduced by the enhanced reliability of the CRD system design features, thereby assuring the adequacy of manual initiation while still providing protection against the adverse consequences of an inadvertent actuation. 0

2.5.5.3 ASSESSMENT 0

The ALWR program emphasis on nuclear safety, design simplification, man-machine interface, and increased margins assures that ATWS event frequency and consequences are very low. 0

The regulatory requirements of General Design Criteria 26 and 10CFR50.62 are fully addressed by the Advanced BWR requirements for independence and diversity in both the electrical and mechanical capabilities of the CRD system and the Standby Liquid Control System. 0

Due to the increased margins, reduced demands on the reactor protection system, and the enhanced reliability of the CRD system by the provisions of the mechanical and electrical independence and diversity, the automatic initiation of the SLC as required by 10CFR50.62(c)(4) is not necessary for the Advanced BWR. 0

Section No.		Rev.
4.3.3	**ALWR Passive Plant Compliance**	0
	The specific requirements of CFR 50.62 are based on the assumption of specific designs. The reactor protection and auxiliary systems affecting reactor shutdown are substantially different in the passive ALWR design than in current LWR designs. Examples of features substantially different in the passive ALWR and which are referenced by CFR 50.62 are:	0
	• The PWR design does not include an auxiliary feedwater system.	0
	• The BWR design does not include a recirculation system; it is designed for natural circulation.	0
	• The BWR control rod drive system has been completely redesigned with an electric motor rod drive system for normal reactor control and a hydraulic scram system without scram discharge volumes.	0
	~~• Although an SLCS will be provided for the BWR, the system is designed to be manually initiated. See Section 2.5.5.~~	~~0~~
	The passive ALWR integrated design considered ATWS as a basic design requirement and has provided a design which addresses the intent of CFR 50.62 by providing highly reliable and diverse reactivity control systems and an automatically actuated SLCS.	0

Paragraph No.	Requirement	Rationale	Rev.
3.4	**COMMON REQUIREMENTS OF PSI AND PDHR SYSTEMS (CONTINUED)**	**COMMON REQUIREMENTS OF PSI AND PDHR SYSTEMS (CONTINUED)**	0
	• The designer shall identify all valves which are to be locked in position and/or provided with position indication in the control room.	• Certain critical valves will require locking and/or control room position indication.	0
	• Valve motors shall generally not be automatically stopped due to an electric overload except during valve operational testing.	• To provide for the highest availability when required while still providing equipment protection during more frequent operational testing.	0
3.5	**DIVERSE REACTIVITY CONTROL SYSTEM**	**DIVERSE REACTIVITY CONTROL SYSTEM**	0
	A diverse reactivity control system that meets the applicable requirements of General Design Criterion (GDC) 26 shall be provided.	This function in the ALWR is provided by the standby liquid control system (SLCS) in the BWR and by the safety injection system (SIS) in the PWR. The normal reactivity control system is specified in Chapter 4.	0
		Requirements for ATWS events as specified in 10CFR50.62 prescribe the SLCS features for BWR plants with the currently used locking-piston drives; the BWR ALWR will utilize an electro-hydraulic drive. (See Chapter 4.) The electric motor drives ~~will be~~ are used as ~~the~~ backup scram mechanisms to ~~satisfy ATWS requirements.~~ reduce the probability of an ATWS event and thereby reduce the probability of a required SLCS actuation.	0

Paragraph No.	*Requirement*	*Rationale*	*Rev.*
4.4.2	**Performance Requirements**	**Performance Requirements**	0
4.4.2.1	The depressurization system shall be capable of depressurizing the reactor vessel to the extent that drywell and reactor pressure reach equilibrium.	The PSIS requires the reactor vessel to be depressurized to drywell pressure for PSIS injection.	0
4.4.2.2	The rate of depressurization shall not be excessively rapid so as to exceed the design limit on reactor vessel blowdowns and/or cause excessive carryover of moisture and still assure PSIS injection before core uncovery can occur.	The effects of a rapid depressurization rate (i.e., vessel mechanical stress and water carryover in the steam discharge) must be balanced against the additional complexity of the depressurization system and the safety requirements of maintaining core coverage.	0
4.4.3	**System and Equipment Requirements**	**System and Equipment Requirements**	0
4.4.3.1	The depressurization system shall work in conjunction with the PSIS and shall have the same system initiating signals.	The PSIS requires the reactor vessel to be depressurized to drywell pressure for PSIS injection.	0
4.4.3.2	The DPS shall have appropriate redundancy of components and features. The performance requirement of no core damage, as specified in Section 2.3.6, shall be met, assuming an initiating event and the limiting single failure.	Redundancy and ability to perform system requirements for core cooling are required by criterion 35 of 10CFR50 Appendix A.	0
4.4.3.3	The depressurization valves shall work in conjunction with the SRVs and ATWS initiation signals to provide additional reactor vessel relief capacity if needed during an ATWS event.	Additional reactor vessel relief capacity may be needed during an ATWS event.	0
4.4.3.~~4~~ 5	Depressurization system actuation circuit continuity testing capability during power operation shall be provided.	The ability to test actuation circuitry during power operation enhances overall plant availability.	0
4.4.3.~~5~~ 6	Depressurization system actuating devices shall be testable. If removal of the actuation device is required for testing, it shall ... sy removal and replacement.	Surveillance testing is required to assure system function and availability.	0

INSERT

4.4.34 Requirement

An automatic, DPS inhibit shall be provided to prevent dilution of the boron injection flow during an ATWS event. The inhibit shall be initiated by signals unique to an ATWS event and shall be compatible with the automatic SLCS injection specified in section 4.5.

Rational:

While a DPS inhibit is not preferred it should be provided when low pressure injection flow from the PSIS would excessively dilute the SLCS boron flow and

Pg 5.4-16

Paragraph No.	Requirement	Rationale	Rev.
4.4.3	**System and Equipment Requirements (Continued)**	**System and Equipment Requirements (Continued)**	0
4.4.3.~~6~~ 7	The SRV and DPS valves shall be of designs that are sufficiently independent to avoid significant vulnerability to common cause failure.	Given the requirements of Section 4.4.3.2 for redundancy within the depressurization system, assuring independence in the manner in which the valves within each group functions will ensure negligible risk of failure to depressurize.	0

Paragraph No.	Requirement	Rev.
4.5	**BWR STANDBY LIQUID CONTROL (SLC) SYSTEM**	0
4.5.1	**Definition**	0
4.5.1.1	**Scope**	0
	This section, together with the applicable portions of Sections 2 and 3 of this chapter, provides the requirements for the BWR SLC system of the ALWR plant.	0
	The SLC system consists of a high pressure accumulator tank containing a liquid control chemical, piping, valves, and controls and instrumentation as shown in Figure 5.4.3.	0
4.5.1.2	**Functions**	0
	The SLC system provides the capability for a means of inserting negative reactivity into the reactor core which is separate and diverse from the control rod system. It provides reactor shutdown from full power operation to cold subcritical, with no assistance from control rod movement, by mixing a neutron absorber with the primary reactor coolant. The system is ~~used in the event that~~ actuated if a sufficient number of control rods cannot be inserted to maintain subcriticality. ~~The SLCS is not required to shut down the reactor or to be a backup system for the CRDS.~~	~~0~~
4.5.1.3	**Interfaces**	0
	• The principal piping interface with the SLC is the reactor vessel (Chapter 4) into which the liquid control is injected; entry may be via one of the reactor coolant injection lines.	0
	• Plant electrical dc power systems (Chapter 11) are used to actuate valves.	0
	• Monitoring of the system status and actuation signals are provided via the instrumentation and control systems in Chapter 10.	0
	• The filter demineralizers of the reactor water cleanup system (Chapter 3) shall be isolated coincident with an SLC injection initiation to avoid removal or dilution of pentaborate in the reactor.	0
	• The demineralized water supply system (Chapter 2) provides water for the initial mixing of boron chemicals.	0
	• The plant high pressure nitrogen supply system (Chapter 7) provides clean gas for mixing of the boron solution.	0

Paragraph No.	Requirement	Rationale	Rev.
4.5.2	**Performance Requirements**	**Performance Requirements**	0
4.5.2.1	SLCS design shall meet the requirements for safety-related systems covered in Sections 2 and 3 of this chapter.	These requirements are ALWR design basis requirements.	0
4.5.2.2	The system shall have the capability for controlling the reactivity difference between the steady-state power operating condition at any time in core life and the cold shutdown condition.	The requirements are to assure the proper functioning of the system.	0
4.5.2.3	The minimum injection flow capability shall be sufficient to bring the reactor from full rated condition to cold shutdown with margin and hold it there while allowing for xenon decay. The injection flow rate and pressure, selection of injection location, and distribution system, if required, shall also ensure adequate mixing and distribution within the reactor for all design basis conditions.	The flow rate, boron content, and injection system design to meet the system functional requirements under conditions of the licensing design basis event will be established by the Plant Designer because it depends on plant size unique parameters.	0
4.5.2.4	Assuming failure of normal scram and the back-up electric motor drives while at normal operation, the system with injection initiated ~~by the operator~~ shall be capable of maintaining (1) the reactor vessel pressure below the emergency limit, (2) containment below its design pressure, and (3) a coolable fuel geometry.	The SLCS provides a further back-up to the electric drives in the event of multiple failures (e.g., failure of normal scram, ARI and electric drives) and for severe accident protection. Automatic feedwater runback as described in Chapter 2 Section 4.2.3.7 is provided as an additional ATWS mitigation feature.	0
4.5.2.5	The minimum liquid control storage capacity shall be sufficient to provide adequate margin when considering reactor coolant temperature, voids, Doppler effect, equilibrium and shutdown margin. An additional margin of 25 percent to be confirmed by the analyses specified in Section 4.5.2.6 shall be provided above calculated value to allow for mixing and distribution within the reactor system. Also, when determining the actual amount of sodium pentaborate needed, consideration shall be given to dilution by the shutdown cooling system.	The basis given for the additional capacity margin is used in current designs and considers maldistribution of boron in the reactor system, the time required to homogeneity, and natural convection of the mixture as a function of its concentration and temperature.	0

Paragraph No.	Requirement	Rationale	Rev.
4.5.3.3	**Arrangement**	**Arrangement**	0
	The accumulator tank and nitrogen supply, including nitrogen control valves, relief valves, and remotely actuated injection valves shall be located outside containment in an accessible area.	To facilitate testing and maintenance and for access during emergency situations.	0
4.5.3.4	**Testing**	**Testing**	0
4.5.3.4.1	The system shall include provisions for functional testing of components without contaminating the reactor system with boron solution during each refueling or planned outage.	The requirements are included to support the refueling outage schedule and plant capacity factors specified in Chapter 1.	0
4.5.3.4.2	Provisions shall be made for sampling and chemical analysis of the liquid control solution during plant operation and shutdown.	Verification of solution concentration is required to ensure adequate shutdown margin. Sampling requirements in more detail are in Chapter 3.	0
4.5.3.5	**Instrumentation and Control**	**Instrumentation and Control**	0
4.5.3.5.1	The SLCS shall be capable of initiation by manual operator action through a ~~key-locked~~ switch in the main control room specifically designed to preclude inadvertent actuation.	The need for automatic operation required by 10CFR50.62 is being met by the independent insertion capability of the electric motor drives of the electro-hydraulic control rod drives. (See Chapter 4.) The key lock is provided to prevent inadvertent operations.	0
4.5.3.5.2	The system shall be capable of operation in the event of loss of ac power, with a coincident most limiting single component failure. Electric power for operating components, including controls and instrumentation, shall be obtained from appropriately independent buses that are connectable to emergency dc power sources.	These requirements are necessary to meet the regulatory requirements for reliability. Heaters, if required to maintain the chemical solution above saturated temperature, would not be considered operating components and therefore would not need to be supplied from emergency power.	0

Replace

4.5.3.5.1 The SLCS shall be automatically actuated on signal(s) which are unique to an ATWS event. The automatic actuation logic shall be highly reliable and designed to avoid inadvertent operation. Manual initiation capability by direct operator action in the MCR shall be provided.

Automatic operation is required by 10 CFR 50.62. The design of the logic should balance the need for reliable actuation with the competing need to avoid spurious boron injection.

Paragraph No.	Requirement	Rationale	Rev.
8.6.3	**Integration and Coordination**	**Integration and Coordination**	0
	The M-MIS for the Depressurization System shall be integrated and coordinated with the M-MIS for other plant systems as required by 8.2.4. In particular, this includes:	It is expected that requirements on separation and segmentation will result in very few direct connections between the Depressurization System M-MIS and other plant M-MIS. The performance requirements of Chapter 5 lead to the integration and coordination defined in this requirement.	0
	• The conditions in the Reactor Coolant System;		0
	• The M-MIS for the Passive Safety Injection System;		0
	• For BWRs, the M-MIS for the overpressure protection of the Reactor Coolant System, i.e., the safety relief valves;		0
	• For PWRs, the M-MIS for the backup reactor trip portion of the Reactor Protection System, if provided.		0
8.7	**BWR STANDBY LIQUID CONTROL (SLC) SYSTEM**	**BWR STANDBY LIQUID CONTROL (SLC) SYSTEM**	0
8.7.1	**Functions**	**Functions**	0
	The M-MIS for the SLC System shall provide the monitoring and control necessary to inject a solution containing a neutron absorber into the reactor coolant so that there is sufficient negative reactivity to bring the reactor to a cold subcritical condition without the control rods.	This allocation of functions is consistent with Section 4.5 of Chapter 5.	0
8.7.2	**Control and Monitoring Strategies**	**Control and Monitoring Strategies**	0
	The SLC system shall be automatically initiated, ~~only~~ by direct operator action in the MCR. Capability of manual initiation shall be provided. This operator action shall involve protective features which effectively preclude inadvertent actuation and assure that the Shift Supervisor concurs in the system actuation.	This is consistent with Section 4.5 of Chapter 5. The inadvertent injection into the system would require substantial cleanup effort.	0

Paragraph No.	Requirement	Rationale	Rev.
8.7.3	**Integration and Coordination** Automatic initiation and	**Integration and Coordination**	0
	The M-MIS for the SLC system will be integrated with the M-MIS design of other plant systems only as necessary to assure that adequate information is available from the neutron monitoring and rod control systems for the operator to decide to use the SLC system. In addition, per Section 7.12.2.3, the RWCU system will isolate automatically upon actuation of the SLC system to ensure it does not remove the neutron absorber from the reactor coolant.	It is expected that there will be no direct connection between the SLC system M-MIS and other plant M-MIS except for the RWCU system M-MIS and that required for automatic SLC initiation.	2
8.8	**CONTAINMENT ISOLATION M-MIS**	**CONTAINMENT ISOLATION M-MIS**	0
8.8.1	**Functions**	**Functions**	0
	The Containment Isolation M-MIS provides the control and monitoring necessary to isolate the containment to minimize the release of radioactivity to the environment.	This allocation of functions is consistent with Section 6.2 of Chapter 5.	0
8.8.2	**Control and Monitoring Strategies**	**Control and Monitoring Strategies**	0
8.8.2.1	**Confirmation of Isolation Action**	**Confirmation of Isolation Action**	0
	The containment isolation shall be initiated and accomplished without operator action. The operators shall be provided with a comprehensive operator aid (display) and appropriate controls which will allow them expeditiously and efficiently to:	Although the isolation is automatic, the operators provide valuable backup. There are, however, many components involved in an isolation. Unless special steps are taken to aid the operators, they will not provide an effective backup.	0
	• Confirm that the required isolation has been completed and to take manual action, if necessary, to complete the isolation;		0
	• Confirm that the initiation of the isolation was based on valid information;		0
	• Take manual control to return isolated systems to service when conditions permit.		0

ALWR/NRC OPEN ISSUES

P.1.O-4 | III.B | Status: Open | Next Action: NRC

check valve categorization (2.3.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.2-10) "Treatment of check valves as active components would cause these valves to be evaluated more stringently than they were in previous licensing reviews."	The ALWR program endorses ANSI-ANS 58.9-1981, "Single Failure Criteria for Light Water Reactor Safety Related Fluid Systms" which considers check-valves to be active components when they are required to change state to perform their safety function. The standard gives examples when the proper function of a component can be demonstrated despite any credible condition. It requires documentation of the exemptions in the single failure analysis. Thus, the consideration of check valves in either active or passive failure will be made on a case by case basis by the plant designer. (Chapter 5, section 4.2.3.1.1 will be made consistent with Chapter 1, section 2.2 to clarify this point.)	(DSER) See Abstract	NRC review pen & ink change (Ch 5)

NRC Review

NRR/EMEB J. Brammer

Last Updated: 8/6/92

Paragraph No.	Requirement	Rationale	Rev.
4.2.2	**Performance Requirements**	**Performance Requirements**	0
	The PSIS shall meet performance requirements as specified in Sections 2 and 3.	Section 2 provides top level *prevention* requirements; Section 3 provides PSIS requirements common to the BWR and PWR designs.	0
4.2.3	**System and Equipment Requirements**	**System and Equipment Requirements**	0
4.2.3.1	**Arrangement**	**Arrangement**	0
4.2.3.1.1	The PSIS shall be divided into an appropriate number of redundant components and features. As a minimum, the following shall be provided:	Separation of the PSIS into redundant components and features is required to meet regulatory requirements which require accomplishment of the licensing design basis function assuming an initiating event and the limiting failure.	0
	• Multiple piping shall be provided for the core coolant makeup from the PSIS pool(s) and the suppression pool to the reactor vessel. They shall have sufficient redundancy and mechanical separation as specified in Section 2.		0
	• Redundancy of components(i.e., valves, controls and instrumentation) shall be provided as necessary to meet Section 2 requirements.	• Redundant components provide single failure protection. Passive safety systems may include active features such as ~~check valves,~~ instrumentation, and single-action valves which initiate systems operation (see Section 1.2.1.1).	0
	• A single PSIS pool may be utilized. The total pool volume may be provided by a number of segments which are connected in such a way that they perform as a single unit.		0

ALWR/NRC OPEN ISSUES

P.1.O-5	II.F	Status: Open	Next Action: NRC/ALWR

tornado wind speeds (4.5.2.5)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-21) "In the draft Commission paper on passive plant policy issues dated February 27, 1992, the staff stated that it will accept the tornado design basis of 300 mph recently proposed by EPRI. Table 1.4-1 shows the design-basis tornado parameters that are considered acceptable to the staff. However, until the staff resolves this issue with the Commission, it considers this to be an open issue ..."	The industry expects Commission endorsement of the wind-speed decision. Furthermore, we concur with site-specific evaluations of explosion and specific air-traffic patterns. However, criteria for general aircraft impact should be developed on a generic basis.	(DSER) See Abstract	NRC (Commission) review Policy on wind-speed. NRC/ALWR develop generic criteria for general aircraft impact

NRC Review

NRR/PRPB J. Lee

Last Updated: 7/17/92

ALWR/NRC OPEN ISSUES

P.1.O-6	II.D	Status: Open	Next Action: ALWR

leak before break (4.5.5)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-26) "The NRC staff concludes that EPRI must commit to and reference NUREG-1061 in the requirement section of Sections 4.5.5.2.2 and 4.5.5.2.4."	See Policy II.D	(DSER) See Abstract	See Policy II.D

NRC Review

NRR/EMEB D. Terao
NRR/EMCB

Last Updated: 7/14/92

ALWR/NRC OPEN ISSUES

P.1.O-7

Status: Open

Next Action: NRC

seismic evaluation and design of small-bore piping (4.7.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-42) "Pending completion of this review (of NCIG-14, EPRI NP-6628), the staff's position is that the methodology in EPRI NP-6628 is not acceptable."	NRC should complete review	(DSER) See Abstract	NRC to complete review

NRC Review

NRR/EMEB J. Brammer
NRR/ESGB

Last Updated: 7/2/92

Printed on: 8/18/92

P.1.O-8		Status: Open	Next Action: NRC

use of IEEE Standard 323 (4.8.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-49) "IEEE Standard 323, 1983 version, has not been found acceptable by the staff. Where differences exist between IEEE Standard 323 and 10 CFR 50.49, the designer must follow the NRC regulation, or identify and justify the differences for the staff to review. Therefore, the above statement in Section 4.8.2.1 of the Requirements Document is not acceptable. The phrase "as outlined in IEEE Standard 323" should be deleted from the sentence."	For all Codes and Standards the designer is expected to invoke the latest edition unless written justification is provided to use a different edition. Differences between the NRC and the designer (on the use of the latest edition) would be covered under such documentation requirements.	(DSER) See Abstract	NRC to review this response

NRC Review

NRR/SPLB G. Hubbard

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.O-9		Status: Open	Next Action: NRC

method of environmental qualification of mechanical and electrical equipment (4.8.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-50) "Section 4.8.2.4 of Chapter 1 of the Requirements Document states that qualification will be accomplished by physical test or by experience, demonstrating the equipment's similarity to previously qualified equipment or to equipment which has been exposed to other more severe environments. The staff finds that the above statement can easily be misinterpreted, and therefore, it needs to be clarified by stating that the method of qualification should be in accordance with 10 CFR 50.49 (f)."	We agree. A URD change will be made to Section 4.8.2.4.	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/SPLB G. Hubbard

Last Updated: 8/6/92

Printed on: 8/18/92

Paragraph No.	Requirement	Rationale	Rev.
4.8.2	**Environmental Qualification of Mechanical and Electrical Equipment**	**Environmental Qualification of Mechanical and Electrical Equipment**	0
4.8.2.1	The Plant Designer shall insure that mechanical and electrical equipment is qualified for use in the operating environment under which it will be required to perform its design function. Mechanical equipment qualification shall meet the requirements of applicable industry standards for the class of equipment involved. Class 1E electrical equipment shall be environmentally qualified in accordance with 10CFR50.49, as outlined in IEEE Standard 323.	This program is required to demonstrate that the equipment will perform its design function on demand to meet system performance requirements when subjected to the design environmental conditions. Establishment of mechanical environmental qualification requirements is in a developmental stage. Industry standards current at the time of equipment qualification should be used.	0
4.8.2.2	The Plant Designer shall make maximum use of provisions in other parts of Section 4.0 to remove excessive conservatisms from environmental analysis and to provide environmental envelopes which closely match the calculated design conditions. Analyses performed to define the environmental envelope shall be completed after analysis variables have been finalized.	The most significant gains to be made in the area of environmental qualification are in the definition of more realistic environmental envelopes for equipment qualification. Unreasonably harsh environmental envelopes have greatly increased the costs of environmental qualification without a corresponding increase in safety.	0
4.8.2.3	Pertinent environmental qualification parameters include, but are not necessarily limited to temperature, pressure, humidity, radiation, chemical spray and aging.	Elimination of unnecessarily conservative design scenarios will lessen such problems.	0
4.8.2.4	Qualification shall be accomplished by physical test or by experience, demonstrating the equipment's similarity to previously qualified equipment or to equipment which has been exposed to other more severe environments.	The use of proven equipment is the optimum approach to equipment qualification. Equipment which has undergone complete qualification tests and has demonstrated reliable service should be the primary choice.	0

For Class 1E electrical equipment, such qualification shall follow the methodology stated in 10 CFR 50.49 (f) outlined below:

(see next page)

(1) Testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

(2) Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable.

(3) Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

(4) Analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

ALWR/NRC OPEN ISSUES

P.1.O-10

Status: Open **Next Action:** NRC

limits on nitrites, nitrates, and total halogens as chlorine (5.2.8)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.5-7) "the staff concludes that EPRI should revise the Passive Requirements Document to include limits on nitrites, nitrates, and total halogens as chlorine. In addition, a total limit on total chlorine + total sulfur + total nitrite + total nitrate expressed as mole-equivalents of chlorine should also be included."	We are unaware on any technical basis for limits on nitrites and nitrates. NRC should explain technical requirement.	(DSER) See Abstract	NRC to review response and continue dialog.

NRC Review

NRR/EMCB G. orgiev

Last Updated: 7/2/92

ALWR/NRC OPEN ISSUES

P. .O-11 | | Status: Open | Next Action: NRC

PWR water chemistry (5.5.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.5-26) "for PWR water chemistry, Section 5.5.2.4 of the Passive Requirements Document should reference EPRI NP-7077, Revision 2, instead of EPRI NP-5960, Revision 1."	We agree. The URD will be revised to reference Revision 2, EPRI NP-7077, "PWR Primary Water Chemistry Guidlines" in Section 5.5.2.4.	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/EMCB G. Georgiev

Last Updated: 8/6/92

Printed on: 8/18/92

Paragraph No.	Requirement	Rationale	Rev.
5.5.2	**Water Chemistry Design Basis** (Continued)	**Water Chemistry Design Basis (Continued)**	0
5.5.2.2	The hydrogen water chemistry specific control values given in the EPRI HWC Guidelines relative to recirculating piping (e.g., 230 MV) shall apply to nozzles, components, and other non-replaceable components in the reactor vessel lower plenum.	The current guidelines (1987 Revision) is directed toward optimizing protection to the recirculation piping; in the ALWR, which has no such piping, the maximum protection should be directed to the reactor vessel pressure boundary.	0
5.5.2.3	A special evaluation shall be made when considering carbon and low alloy material for reactor coolant service with less than 10 ppm oxygen as the result of HWC. The evaluation shall include erosion, corrosion, radiation buildup, and pitting at shutdown.	The side effects of HWC must be considered in selection of materials. For example, in one plant, higher radiation levels (due to crud and cobalt) have been observed in some carbon steel piping.	0
5.5.2.4	**PWR Water Chemistry Design Basis**	**PWR Water Chemistry Design Basis**	0
	The water chemistry design basis for the PWR shall be in accordance with: • *PWR Primary Water Chemistry Guidelines:* ~~[illegible]~~0 *, and its subsequent revisions; • *PWR Secondary Water Chemistry Guidelines: Revision 2*, EPRI NP-6239, and its subsequent revisions; and as supplemented by the guidelines provided in Table 1.5-4.	The guidance for auxiliary systems in Table 1.5-4 are provided to assure compliance with the EPRI guidelines. Based on experience and the system and component design features specified in the ALWR Requirements Document, the guidelines are judged to be achievable at least 90 percent of the time. The guidelines are intended to be the basis for system design and not the objectives for excellent plant performance.	0

* : Revision 2, EPRI NP-7077

ALWR/NRC OPEN ISSUES

P.1.O-12

Status: Open

Next Action: NRC

reliability assurance program framework (6.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.6-4) "The staff concludes that a RAP should contain and define the basic framework (scope, purpose, and objective) and the program elements, and should describe how the elements would be applied to the plant structures, systems, and equipment." (DSER, p 1.6-7) "The staff concludes that a RAP should contain and define the basic framework (scope, purpose, and objective) and the program elements, and should describe how the elements would be applied to the plant structures, systems, and equipment."	We believe the staff's concern was addressed by Revision 3 of Chapter 1, Section 6.	(DSER) See Abstract	NRC to review Ch. 1, Section 6, Revision 3

NRC Review

NRR/LPEB R. Correia

Last Updated: 7/2/92

ALWR/NRC OPEN ISSUES

P.1.O-13		Status: Open	Next Action: NRC

quantitative reliability and availability goals (6.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.6-5) "The staff concludes that a RAP should contain overall reliability and availability design goals based on safety requirements that have associated with them core-damage frequencies or probabilities. The staff agrees that plant availability, outage duration, and outage frequency are appropriate quantitative design requirements. However, quantitative reliability and availability requirements that will be responsive to those safety requirements that have associated with them core-damage frequencies or probabilities should also be included as design requirements for non-safety-related systems."	We believe the staff's concern was addressed by Revision 3 of Chapter 1, Section 6.	(DSER) See Abstract	NRC to review Ch 1, Section 6

NRC Review

NRR/LPEB R. Correia

Last Updated: 7/2/92

P.1.O-14

Status: Open

Next Action: NRC

integration of reliability engineering techniques (6.2, 6.3, 6.4)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.6-6) "The staff concludes that a RAP should contain reliability engineering techniques performed during the design phase to ensure the overall design reliability and availability goals that are based on plant safety are met. The staff acknowledges that the techniques described in Section 6.2.3 of the Requirements Document are reliability program elements that will be applicable in the design phase of a plant and are acceptable. However, how these techniques will relate to the overall reliability program and how they will be integrated and considered in the RAP is not specified."	We believe the staff's concern was addressed by Revision 3 of Chapter 1, Section 6.	(DSER) See Abstract	NRC to review Ch 1, Section 6, Rev 3

(DSER, p 1.6-7) "The staff concludes that a RAP should contain reliability engineering techniques performed during the design phase to ensure the overall design reliability and availability goals will be met. As discussed above, the staff acknowledges that the techniques described in Section 6.2.3 of Chapter 1 of the Requirements Document are reliability program elements that are applicable in the design phase of a plant and are acceptable. However, as stated above, how these techniques will be integrated and considered in the overall reliability program is not specified."

(DSER, p 1.6-8) "During its review of the Requirements Document, the staff found many of the elements of a RAP contained in other sections and chapters, such as Sections 2.3.3.7 and 2.3.3.8 of Chapter 1, Sections 2.2.12 and 3.4.5 of Chapter 5, and Section 6.1.6.3 of Chapter 10. However, as discussed above, EPRI did not state how these elements will be integrated and considered in the total reliability program."

NRC Review

NRR/LPEB R. Correia

Last Updated: 7/2/92

ALWR/NRC OPEN ISSUES

P.1.O-15

Status: Open

Next Action: NRC

relationship of system requirements to overall plant safety reliability and availability goals (6.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.6-6) "The staff concludes that the designer should establish a set of system reliability and availability goals to ensure that the overall reliability and availability goals that are based on plant safety will be met. The staff finds the requirements specified in Section 6.2.4 acceptable; however, the connection of the system requirements to overall plant safety reliability and availability goals is not specified."	We believe the staff's concern was addressed by Revision 3 of Chapter 1, Section 6.	(DSER) See Abstract	NRC to review Ch 1, Section 6, Rev 3

NRC Review

NRR/LPEB R. Correia

Last Updated: 7/2/92

ALWR/NRC OPEN ISSUES

P.1.O-16

Status: Open

Next Action: NRC

difference between reliability assurance program for safety- and non-safety-related systems (6.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.6-???) [COULD NOT FIND SPECIFIC REFERENCE JDT 4/30/92)	NRC should identify what the problem is.	UNKNOWN	NRC to clarify issue

NRC Review

NRR/LPEB R. Correia

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ALWR/NRC OPEN ISSUES

P.1.O-17		Status: Open	Next Action: NRC

human factors considerations for operation and maintenance provisions (8.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, P 1.8-2) "In Section 8.2.B.4 of the DSER for Chapter 1, the staff recommended that IEEE P1023/D5, "Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations," and EPRI 2360, "Human Factors Methods for Assessing and Enhancing Power Plant Maintainability," be referenced in this section. The staff concludes that this recommendation is applicable to passive plant designs. These documents are not referenced in the Requirements Document."	The URD will be changed (Section 8.2.1.2) to reference IEEE 1023 - 1988. Chapter 10, Section 3.7.7 contains a reference to EPRI NP-4350, "Human Engineering Design Guidelines for Maintainability", which in turn references the older report EPRI 2360, "Human Factors Methods for Assessing and Enhancing Power Plant Maintainability".	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/LHFB D. Smith

Last Updated: 8/19/92

Paragraph No.	Requirement	Rationale	Rev.
8.2	**PROVISIONS TO ENHANCE OPERABILITY AND MAINTAINABILITY**	**PROVISIONS TO ENHANCE OPERABILITY AND MAINTAINABILITY**	0
8.2.1	**Solution to Known Operations and Maintenance Problems**	**Solution to Known Operations and Maintenance Problems**	0
8.2.1.1	The Plant Designer and Constructor shall document known operations and maintenance problems specific to the ALWR design and their solutions in a report available for Utility review prior to plant commitment. This report shall be based on review of the data sources listed in Table 1.8-1 and shall cover, as a minimum, the problem areas listed in Tables 1.8-2 and 1.8-3. The report shall cover all issues of plant performance and not be restricted to nuclear or non-nuclear equipment and shall be performed at a detailed enough level so that root causes can be determined and appropriate solutions prepared.	Utility inputs on the ALWR Program have repeatedly stressed the need to systematically identify and resolve problems that exist in present plants. The Plant Designer is also encouraged to look to foreign data sources (in addition to Table 1.8-1). Based on a review of experience, a number of operation and maintenance problems that exist in the present nuclear plants can already be identified, and are listed in Tables 1.8-2 and 1.8-3.	0
8.2.1.2	Human factors design principles shall be consistently applied throughout the design process for each operation or maintenance work space in the ALWR plant to reduce operation and maintenance errors during all plant modes.	Human errors that affect plant performance may be system-, design-, or human-induced. Human factors applications focus on eliminating from the ALWR the causes of human errors that exist in the present plants. (Information on human errors can be found in many of the references of Table 1.8-1.)	0
8.2.2	**Procedures and Training**	**Procedures and Training**	0
8.2.2.1	Consistent with the standard plant design described in Section 11.5, procedures and training for operation and maintenance shall be standardized. A standard set of operating and maintenance procedures and training shall be developed for each ALWR design. In addition, standardization between ALWR designs should be addressed to the extent practical.	A standardized set of procedures and training should permit achieving high quality and performance in operation and maintenance activities. It is recognized that there will be differences due to design-unique factors; however, standardization to some degree between ALWR designs should be achievable through the utility/operations review described in Section 11.11.	0

The IEEE 1023-1988, Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations will be utilized by the Plant Designer.

The IEEE Guide 1023-1988 ~~[illegible]~~ provides comprehensive guidelines on the design process for taking into account human factor.

ALWR/NRC OPEN ISSUES

P.1.O-18 | | Status: Open | Next Action: NRC

computer security reference (11.12)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.11-6) "Section 5.9 of Attachment 1 to Section 11 of Chapter 1 states: "The plant designer will certify the IMS security system in accordance with Guidelines for Computer Security and Accreditation (GUIDE 83)." GUIDE 83 does not appear to be a correct citation."	The URD will be changed to eliminate the reference to "Guide 83"	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/RSGB R. Dube

Last Updated: 8/6/92

Volume III

ATTACHMENT 1 TO SECTION 11

Paragraph No.	Requirement	Rev.
5.8	**QUALITY ASSURANCE**	0
	The IMS shall be capable of automatically maintaining complete records of the design as it evolves. A method utilized to achieve this provision is described in Section 5.7.	0
	Rationale:	0
	Since design verification is achieved by peer design reviews which are recorded in the IMS, and the IMS is constructed to provide a single verified official traceable information source, the quality of the design can be demonstrated by appropriate IMS queries at any work station at any time.	0
5.9	**COMPUTER AND NETWORK SECURITY**	0
	An IMS security level, together with the methods utilized to achieve the desired risk level, shall be proposed by the Plant Designer for review and approval by the Plant Owner. The Plant Designer shall certify the IMS security system in accordance with Guidelines for Computer Security and Accreditation (GUIDE 83). The Plant Designer shall be responsible to maintain the security of the IMS at the approved and certified level.	0
	Rationale: Delete (NBS FIPS PUB 102.) Add	0
	This requirement is intended to cover all aspects of security including unauthorized disclosures of information and data loss or contamination. There are a number of commercially applied methods to minimize computer system vulnerability to data loss or contamination. Since no system is absolutely safe, risk management involves analysis of risks, cost of recovery, cost of risk reduction, and acceptance of residual risk. Factors to be considered in a risk analysis are; the necessity of the system to function at a given level of performance, maintaining data accuracy and preserving continuity of operation. Plant Owner management must decide what performance level is required relative to what level of residual risk is acceptable. The Plant Designer is responsible to demonstrate that the proposed methods achieve the acceptable security level. Reference 2 provides more information on the subject of computer and network security.	2

ALWR/NRC OPEN ISSUES

P.1.C-1

Status: Confirm

Next Action: NRC

tornado wind speeds (4.5.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-21) "During a January 30, 1992, meeting with the staff, EPRI indicated that it would delete the reference to the tornado recurrence interval from the Requirements Document. EPRI should revise Table 1.2-6 accordingly."	Agree. Changes will be made in Revision 4 of Volume III	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/PRPB J. Lee

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Fuse 9.5.

Table 1.2-6

ENVELOPE OF ALWR PLANT SITE DESIGN PARAMETERS[1]

	Rev.
EXTREME WIND:	0
Basic wind speed:110 mph[2]	0
Importance factors: 1.0[3]/1.11[4]	0
TORNADO[5]	0
• Maximum tornado wind speed: ~~200~~ 300 mph	0
• ~~Translational velocity: 57 mph~~ Maximum Rotational Speed: 240 mph	0
• ~~Radius: 453 ft~~ Maximum Translational Speed: 60 mph	0
• ~~Maximum atm dP: 1.40 psid~~ Radius of Maximum Rotational Speed: 150ft	0
• ~~Rate of pressure change: .27 psi/sec.~~ Maximum Pressure Drop: 2.0 psi	0
• Missile Spectra: Spectrum A of SRP 3.5.1.4	0
• ~~Missile Velocity: 35% of Maximum Horizontal Windspeed~~ Rate of Pressure Drop 1.2 psi/sec	0
• ~~Missile Altitude: 30 ft above grade for large soft and large rigid missiles; all elevations for the small rigid missile.~~	0
SOIL PROPERTIES[8]	0
• Minimum Bearing Capacity demand: ≥ 15 ksf	0
• Minimum Shear Wave Velocity: ≥ 1000 fps	0
• Liquefaction Potential: None	0
(at Site-Specific SSE Level)	0
SEISMOLOGY	0
• SSE PGA[9]: 0.30g[10][11]	0
• SSE Design Response Spectra: per Reg. Guide 1.60	1
• SSE Time History: Envelope SSE Response Spectra	0

Table 1.2-6

ENVELOPE OF ALWR PLANT SITE DESIGN PARAMETERS[1]

Rev.

ATMOSPHERIC DISPERSION (Chi/Q)[13] 3

Downwind Distance	0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day	Rev.
						3
0.5 mile	1.0 E-3					3
2.0 mile	–	1.35 E-4	1.0 E-4	5.4 E-5	2.2 E-5	3

NOTES: 0

(1) Further definition on application of site design parameters may be found in subsequent chapters covering various areas of design. For example, Volumes II and III: 0

Chapter 6 – Building Design and Arrangement
Chapter 8 – Plant Cooling Water Systems
Chapter 9 – Site Support Systems.

(2) 50-year recurrence interval. 0

(3) Importance factor to be used for non-safety-related structures as defined in ANSI A58.1-1982. 0

(4) Importance factor to be used for safety-related structures as defined in ANSI A58.1-1982. 0

(5) Probable maximum flood level (PMF), as defined in ANSI/ANS 2.8-1983, *Determining Design Basis Flooding at Power Reactor Sites*. Minimum value to be basis of standard plant design with provisions as defined in Chapter 6 for accommodation of flood levels up to maximum value. 0

(6) Maximum value for 1 hour 1 sq. mile PMP with ratio of 5 minutes to 1 hour PMP of .32 as found in National Weather Service Publication HMR No. 52. 0

~~(7) 1,000,000-year tornado recurrence interval with associated parameters based on ANSI/ANS 2.3-1983. Pressure effects associated with potential off-site explosions are assumed to be non-controlling for the design. See ALWR optimization paper on tornado design for additional information.~~ ~~0~~

7 ~~(8)~~ Values of bearing capacity and shear wave velocity are included in this table to assure wide application of a standard mat-type foundation design. Design must be evaluated parametrically against ranges of possible soil properties to verify wide application. ~~0~~

Table 1.2-6

ENVELOPE OF ALWR PLANT SITE DESIGN PARAMETERS(1)

Rev.

NOTES (CONTINUED): 0

8 ~~(9)~~ PGA = Peak Ground Acceleration. 0

9 ~~(10)~~ Free-field at plant grade elevation. 0

10 ~~(11)~~ Envelopes all present U.S. nuclear sites except those on California coastline. 0

11 ~~(12)~~ The indicated ambient temperatures are to be used in accordance with requirements specified in the appropriate sections of Chapters 8 and 9. 2

12 ~~(13)~~ The Chi/Q values are to be used for the 10CFR100 dose evaluation and were determined using meteorological data representative of an 80-90th percentile U.S. site. The Chi/Q values were calculated following guidance in Regulatory Guide 1.145 considering ground level release, building wake (building area of 33,800 ft^2), and lateral plume meander under stable atmospheric conditions. 3

ALWR/NRC OPEN ISSUES

P.1.C-2		Status: Confirm	Next Action: NRC

internal flooding design criteria (4.5.5)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-27) "The staff will confirm that EPRI will add information about internal flooding to Table 1.2-4 of Chapter 1 of the Requirements Document as committed to in a letter dated May 22, 1991."	"Flooding" added to Table 1.2-4 in Rev. 3 of Volume III	(DSER) See Abstract	NRC to review Rev.3

NRC Review

NRR/SPLB C. Li

Last Updated: 7/2/92

ALWR/NRC OPEN ISSUES

P.1.C-3 | **Status:** Confirm | **Next Action:** NRC

compliance with Regulatory Guides 1.26 and 1.29 (9)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.9-2) "the commitment to Regulatory Guides 1.26 and 1.29 is shown in Table B.1-2 as an optimization subject regarding the BWR main steamline isolation valves (MSIVs) and leakage control system, with Chapter 3 of the Requirements Document being listed as the lead chapter. The staff concludes that Table B.1-2 should clearly state that both of these regulatory guides will be complied with, with the exception of the MSIVs and leakage control system."	The clarification of what the optimization issue aplies to is found in the text of Section 2.3.1 of Appendix B. That Section is referenced in Table B.1-2.	(DSER) See Abstract	NRC to review this response.

NRC Review

NRR/LPEB S. Magruder

Last Updated: 7/17/92

ALWR/NRC OPEN ISSUES

P.1.V-1

Status: Closed(Cert) **Next Action:** none

implementation of design characteristics intended to enhance accident resistance (2.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.2-2) "Section 2.2 of Chapter 1 of the Passive Requirements Document specifies the ALWR design characteristics that are intended to enhance accident resistance, such as emphasizing simplification, providing ample design margin, using the best available materials and water chemistry, using the best proven diagnostic monitoring techniques, and maintaining a negative overall power reactivity coefficient under all conditions. Improved design margin is attained by the use of a 15-percent fuel design margin; lower core power densities; a larger reactor vessel, pressurizer, and steam generator secondary side; a longer transient response time; and sufficient margins to limiting conditions for operation and reactor trip setpoints. The staff concludes that these design characteristic requirements are acceptable in principle. However, the staff will evaluate the acceptability of the implementation of each specific design characteristic during its review of an individual application for final design approval and design certification."	agree	(DSER) See Abstract	

NRC Review

NRR/SRXB M. Rubin

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ALWR/NRC OPEN ISSUES

P.1.V-2

Status: Open

Next Action: none

bounding analysis by standard site design parameters (2.3.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.2-4) "Among the general requirements, Section 2.3.1.8 of Chapter 1 addresses plant siting. Table 1.2 6 lists the envelope of standard site design parameters, for which the requirements are discussed in greater detail in Section 4.5.2 of Chapter 1. These siting parameters are intended to cover most, but not all, potential sites for future ALWRs in the United States. As such, the Passive Requirements Document requires that the plant designer review the conditions at the plant owner's site against the standard design siting parameters in order to assess the possible need for modification of any design parameter. Further, it requires that the final design parameters to be used for the particular site be approved in writing by the plant owner. The results of staff's review of the envelope of standard site design parameters are given in Section 4.5.2 of this DSER chapter. In the DSER for Chapter 1 of the Requirements Document for evolutionary plant designs, the staff identified an open issue concerning worst-case site parameters. As for the final site design parameters to be used for any particular site, approval by the plant owner only is not sufficient. Approval by the NRC staff is also required. In addition, if one or more than one site-specific design parameter exceeds the standard site design parameters at some potential nuclear plant site, the plant owner should conduct a plant-specific evaluation against these parameters and submit a detailed review to the staff for approval."	Agree	(DSER) See Abstract	

NRC Review

NRR/PRPB J. Lee
NRR/ESGB

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ALWR/NRC OPEN ISSUES

P.1.V-3

Status: Open **Next Action:** NRC

selection of initiating events and their frequency categorization (2.3.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.2-6) "The events and accidents in Table 1.2-1 of Chapter 1 are generally consistent with the initiating events described in Chapter 15 of the SRP with the following exceptions. Item 3.1, "reactor coolant pump trips (PWR)," should include both partial and complete reactor coolant pump trips (both are moderate-frequency events). Category 2, "decrease in heat removal by secondary system," should include steam pressure regulator failure and turbine trip without bypass as moderate frequency events. Item 6.1, "inadvertent opening of a pressurizer safety valve in a PWR or a safety or relief valve in a BWR," should continue to be categorized as a moderate-frequency event rather than an infrequent event. There are also certain acceptable exceptions because of the design features unique to the ALWR passive plants. For example, the recirculation pump trip and pump shaft seizure events for BWRs are not included in the list because there are no recirculation pumps in the passive BWRs. Control rod assembly malfunctions as passive BWRs may also be excluded if proper justification can be provided for excluding them because unique design features, interlocks, and routine surveillance. However, because the Passive Requirements Document does not present an actual design, the staff concludes that there is insufficient justification to delete consideration of the rod drop event as an accident. The plant designer must justify the exclusion of BWR control rod assembly malfunctions from consideration in safety analyses during the design-specific review."	Table is not an inclusive listing of events to be considered in the safety analysis.	(DSER) See Abstract	NRC review this response.

NRC Review

NRR/SRXB M. Rubin

Last Updated: 3/18/92

ALWR/NRC OPEN ISSUES

P.1.V-4

Status: Closed(Cert) **Next Action:** none

acceptance criteria for transient and accident analysis (2.3.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.2-8) "Section 2.3.2.7 of Chapter 1 states that the plant designer will perform a consequence analysis for moderate-frequency and infrequent events with coincident single failures and specifies the acceptance criteria for these events, including limiting faults as summarized in Table 1.2-2c. In Revision 2 of the Passive Requirements Document, EPRI added a footnote to Table 1.2-2c to specify fuel cladding failure criteria for input to the radiological consequence analyses. That is, for the PWR, the fuel cladding failure criteria will be less than the 95/95 departure from nucleate boiling ratio (DNBR) limit, and for the BWR, it will be less than the minimum critical power ratio (MCPR), except for (1) a LOCA event or (2) a fuel handling and cask drop event. For a LOCA, EPRI specifies that the vendor should use the source term as defined in Section 2.5.2 of Appendix B to Chapter 1, and for a fuel handling and cask drop event, the vendor should use the number of assemblies involved. These failure criteria are consistent with the SRP, except for the source term for LOCA consequence analysis. Table 1.2-2c also specifies limits based on 10 CFR Part 20 and Appendix I to 10 CFR Part 50 for moderate-frequency and infrequent events, and specifies that, for PWRs, the radiological consequences of infrequent events may exceed the guidelines of 10 CFR Part 20 but cannot be such that they to interrupt or restrict public use of those areas beyond the exclusion areas. The staff concludes that the EPRI-proposed criteria are not specific enough to determine if they are consistent with the staff's review criteria. The plant designer should specify the exact acceptance criteria and identify deviations from those in the SRP, if any, and the bases for the deviations. The staff will address this matter during its review of an individual application for final design approval and design certification."	Agree	(DSER) See Abstract	

NRC Review

NRR/SRXB M. Rubin

Last Updated: 7/14/92

ALWR/NRC OPEN ISSUES

P.1.V-5

Status: Closed(Cert) **Next Action:** None

passive plant anticipated transient without scram response analysis (2.3.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.2-9) "Section 2.3.2.2 of Chapter 1 of the Passive Requirements Document specifies that analysis and acceptance criteria for events involving failures of multiple active components associated with anticipated transients without scram (ATWS) and station blackout will be in accordance with 10 CFR 50.62 and 10 CFR 50.63, respectively. However, 10 CFR 50.62 does not specify analysis and acceptance criteria, except for the prescriptive equipment design requirements that were based on analyses of the current generation of LWRs. The staff will require each plant designer to perform an analysis for the ATWS events to demonstrate that passive plant ATWS response is consistent with that considered by the staff in its formulation of the 10 CFR 50.62 design requirements for current plants. The staff will address this matter during its review of an individual application for final design approval and design certification."	Agree	(DSER) See Abstract	

NRC Review

NRR/SRXB M. Rubin

Last Updated: 7/14/92

P.1.V-6	III.A	Status: Open	Next Action: ALWR/NRC

operator actions 72 hours after accident (2.3.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.2-11) "Not only is the 72-hour timeframe used as the design basis for the safety systems, it is also used as a justification for not requiring support systems to be designed for safety-related functions. These design bases are consistent with the policies for passive ALWRs stated in Volume I of the Requirements Document. However, the Passive Requirements Document does not provide sufficient justifications or bases to substantiate these policies and objectives. Since the safety systems are designed for 72-hour duration after the initiation of accidents and may rely on other non-safety-grade active systems to perform core-damage prevention and mitigation functions thereafter, the staff is concerned about the reliability and availability of these non-safety-grade active support systems. Section 2.4.2.8 of Chapter 5 of the Passive Requirements Document specifies that non-safety-related equipment necessary for plant recovery after the assumed 72-hour accident duration will be designed for the expected environment during the 72-hour period. However, the Passive Requirements Document does not identify which equipment will be needed for (1) plant recovery after 72 hours and (2) continued accident mitigation. Although Chapter 3 of the Passive Requirements Document specifies that the active support systems will provide defense-in-depth functions, such as reactor coolant makeup and decay heat removal functions, they are not required to meet the requirements for safety-grade systems. In addition, the Passive Requirements Document specifies that only simple, unambiguous operator actions and easily accomplished offsite assistance will be necessary after 72 hours to prevent fuel damage. However, there is no clear definition of what constitutes 'simple, unambiguous operator action and easily accomplished offsite assistance.' "	• One of the concerns of the staff's is "the reliability and availability of non-safety systems". See the Policy Issue for further discussion. • The other concern is about the plant recovery after 72 hours. The staff should consider the modifications made in the RAI response dated 8/16/91 to sections 2.2.11 and 2.2.15 of Chapter 5.	(DSER) See Abstract	See III.A and also NRC to consider 8/16/91 response

NRC Review

NRR/LOLB
NRR/LPEB
NRR/PDST
NRR/PEPB
NRR/PRAB
NRR/PRPB
NRR/RSGB
NRR/SELB
NRR/SICB
NRR/SPLB
NRR/SRXB
RES

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-7	III.A	Status: Open	Next Action: ALWR/NRC

use of 72-hour design basis (2.3.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.2-12) "Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," requires that a safety-related ultimate heat sink consisting of at least two sources of water be provided with the capability to provide sufficient cooling for "at least 30 days" to permit simultaneous safe-shutdown and cooldown of all nuclear units that it serves and to maintain them in a safe-shutdown condition. Since PDHR systems provide a safety-grade means to transfer decay heat to the ultimate heat sinks, the requirement of a 72-hour capability for these systems is inconsistent with the 30-day capability of the safety-related ultimate heat sink specified in the regulatory guide. The staff will evaluate use of a 72-hour safety system design basis (i.e., requiring safety systems to be designed with 72-hour capability without reliance on operator action and other support system assistance) during its review of an individual specific application for final design approval and design certification. Each applicant must justify the use of the 72-hour timeframe design basis, the type of equipment and its quality and its availability for plant recovery after 72 hours, and demonstrate that available non-safety-grade systems will provide for long-term cooling."	See issue P.1.V-6	(DSER) See Abstract	see P.1.V-6

NRC Review

NRR/LOLB
NRR/LPEB
NRR/PDST
NRR/PEPB
NRR/PRAB
NRR/PRPB
NRR/RSGB
NRR/SELB
NRR/SICB
NRR/SPLB
NRR/SRXB
RES

Last Updated: 7/1 2

ALWR/NRC OPEN ISSUES

P.1.V-8		Status: Closed(Cert)	Next Action: None

technical basis for severe-accident management program and emergency operating procedures guidelines (2.3.4)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.2.16) "The staff concludes that EPRI's statements that the EOPs (or guidelines) will be provided by the plant designer as an integral part of the total plant design, will be based on detailed analyses of the functions and tasks of the operators, and will be verified and validated by active simulation is acceptable. If there are situations in which there is the potential for a human error of commission, the operators will be provided with procedural guidance and, if that is not sufficient, physical interlocks and inhibits will be incorporated in the design. Also, the procedures will not preclude operator action after the initiation of a passive system if that action can mitigate the consequences of an event; however, any such actions will have to be consistent with the need to permit safety system actuation to proceed to completion. Furthermore, such procedural guidance will have to be fully verified and validated.	Agree	(DSER) See Abstract	

In addition, the normal plant design configuration limiting the ability to initiate overrides, along with the inclusion of unambiguous symptoms for manual control of safety systems, will limit the contribution of errors of commission to plant risk.

The staff concludes that these design objectives are acceptable. However, it will evaluate the technical bases for the severe accident management and EOP guidelines, and the design of acceptable means (i.e., hardware or software inhibits) to prevent operator override of safety system functions without unduly restricting the operators' ability to respond to system failures or unanticipated event progression during its review of an individual application for final design approval and design certification."

NRC Review

NRR/PRAB R. Palla

Last Updated: 7/14/92

ALWR/NRC OPEN ISSUES

P.1.V-9

Status: Closed(Cert) **Next Action:** none

acceptability of analytical codes and methodologies for safety analysis (2.5)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.2-17) "Acceptable analytical codes used for ALWR analyses must have the capability to handle system performance and phenomena unique to passive ALWR designs, such as operation of gravity-fed safety injection systems, natural circulation heat exchangers at both high and low pressures and in the presence of noncondensible gases, critical flow in the automatic depressurization valves over the range of expected operating conditions, boron transport under gravity-driven system conditions, and natural circulation stability of the simplified BWR design. The codes must also be capable of handling interactions among passive safety systems as well as active systems. The analytical methods and codes must be validated with appropriate test data, including separate effects and integral effect tests with consideration of scaling effects. In its letter dated July 1, 1991, regarding the need for large-scale, full-height, and full-pressure integral testing of the passive safety systems, EPRI indicated that the ALWR Utility Steering Committee has established an analysis and testing review team, which is reviewing the detailed analysis and test plans for the passive plant designs and providing appropriate guidance. The staff will evaluate the acceptability of the analytical codes and methodologies used for safety analyses as well as validation data, including test plans and facilities, during its review of an individual application for final design approval and design certification."	Agree	(DSER) See Abstract	

NRC Review

NRR/SRXB M. Rubin

Last Updated: 7/14/92

ALWR/NRC OPEN ISSUES

P.1.V-10

Status: Closed(Cert)   Next Action: NRC/ALWR

defense-in-depth analysis (2.5, 3.5)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.3-3) "Section 3.5.3 of Chapter 1 requires that the plant responses to reactor trips, which are not complicated by failures beyond those that caused the trip, do not result (1) in the initiation of the emergency core cooling system, the primary safety or relief valve, or the backup feedwater system and (2) in the uncovering of uncovering the pressurizer heaters. Section 3.5.4 requires the plant to be capable of a turbine trip from 40 percent or less (BWR) and 100 percent or less (PWR) of the rated power without reactor trip and the lifting of the main steam safety valves. Section 3.5.5 requires that the inadvertent closure of the mainstream isolation valves (MSIVs) while at full power not result in the actuation of the safety/relief valves for BWRs. Section 3.5.6 requires that the loss of a running main feedwater or condensate pump while at full power not result in a reactor trip. Section 3.5.7 requires the systems and equipment to be designed to withstand a complete loss of bulk ac power for at least 2 hours without exceeding equipment design limits. The staff considers these design objectives as important defense-in-depth goals. Each of these requirements is acceptable provided the designer performs design-specific analyses to demonstrate that the specified design limits are met. The staff will evaluate these analyses during its review of an individual application for final design approval and design certification."	The URD asks that design limits for the plant's infrequent and moderate frequency vents are not exceeded without reliance on safety systems. This is not a licensing concern and should not be a SER issue.	(DSER) See Abstract	NRC to consider removal of this issue or continue discussion as needed.

NRC Review

NRR/SRXB M. Rubin

Last Updated: 7/14/92

ALWR/NRC OPEN ISSUES

P.1.V-11		**Status:** Closed(Cert)	**Next Action:** ALWR/NRC

60-year plant life (3.3, 4.8.2, 8.2, 11.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.3-1) "As stated in SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors," the staff will review the ALWR designs for a 60-year life notwithstanding the fact that a 40-year license term limitation is specified in the Atomic Energy Act and NRC's regulations. Although the Commission paper was directed to the evolutionary designs, the staff concludes that it is equally applicable to the passive designs. It is the applicants' responsibility to identify the components and systems that are affected. The staff will address plant life during its review of an individual applications for final design approval and design certification. These applications will have to provide information and programs to support design life and the staff's reviews of such issues as fatigue, corrosion, and thermal aging." (DSER, p 1.4-50) "It should be noted that ALWRs are designed for 60 years of operation, while the current plants are designed for 40 years. The staff concludes that the plant designer should ensure that plant equipment important to safety must be qualified for its intended service, and be able to perform its safety functions throughout its design life. The staff will address this issue during its review of an individual application for final design approval and design certification."	Section 11.3 was revised (Rev. 3) to better express the industry position. A design life plan shall be provided, which will include a design life classification system, condition monitoring and plant environmental monitoring. There is no contradiction with what the NRC staff is saying so there appears to be no issue.	(DSER) See Abstract	ALWR/NRC discussion to clarify issue.

NRC Review

NRR/PDST

Last Updated: 7/14/92

ALWR/NRC OPEN ISSUES

P.1.V-12 | Status: Open | Next Action: NRC

operation of PWR with a secured reactor coolant pump (3.5)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.3-2) "Section 3.5.2 of Chapter 1 requires the PWR plant to be capable of operating at reduced power with a secured coolant pump to enhance the availability of the plant and to reduce reactor trips. Appendix B to Chapter 1 indicates EPRI's commitment to comply with Generic Letter (GL) 86-09, "Technical Resolution of Generic Issue No. B-59 - (N-1) Loop Operation in BWRs and PWRs." GL 86-09 states that (N-1) loop operation is acceptable provided acceptable evaluation results are shown for certain plant-specific design characteristics, such as the impact of the down loop on instrumentation and control systems, human factors, operational systems, safety systems, status of valves, core-flow distribution, and potential for cold water reactivity insertion. Since these characteristics are highly dependent on the specific design of the plant, acceptability of operation with one secured reactor coolant pump is subject to plant-specific evaluation to address the concerns delineated in GL 86-09. The staff will evaluate that analysis during its review of an individual application for final design approval and design certification."	The requirement about "being capable of operating at reduced power with a secured coolant pump" has been deleted (Rev 3). This issue should be closed.	(DSER) See Abstract	NRC to review Rev. 3

NRC Review

NRR/SRXB G. Hsii

Last Updated: 7/14/92

ALWR/NRC OPEN ISSUES

P.1.V-13		Status: Closed(Cert)	Next Action: none

fuel burnup requirements (3.6)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.3-3) "Section 3.6 of Chapter 1 requires ALWR core designs that have a capability of up to a 24-month fuel cycle and fuel mechanical designs that have a peak bundle burnup of at least 45,000 and 55,000 megawatt days per metric tons of uranium (MWD/MTU) for BWRs and PWRs, respectively. These minimum fuel burnup requirements are inconsistent with the EPRI-proposed requirements of 50,000 and 60,000 MWD/MTU specified in Sections 4.2.2.2 and 7.2.2.2 of Chapter 4 for BWRs and PWRs, respectively. In addition, although these specific values are inconsistent with each other, they are greater than NRC-approved fuel burnup levels. To support this high fuel burnup operation, each ALWR design applications will need to include sufficient high fuel burnup data to demonstrate fuel integrity in the areas of fission gas release, cladding corrosion due to oxidation and hydriding, and reduction in cladding material strength."	Agree	(DSER) See Abstract	

NRC Review

NRR/SRXB G. Hsii

Last Updated: 7/14/92

ALW 'NRC OPEN ISSUES

P.1.V-14		Status: Closed(Cert)	Next Action: none

extended operating life of control blades and control rod assemblies (3.6)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.3-3) "Section 4.2.6.2 of Chapter 4 requires the BWR control blades used for maximum core insertion to be designed with a minimum exposure capability of 4.0E+21 neutrons/m2 (nvt) with a target of 8.0E+21 nvt, and the blades not used for maximum core insertion to be designed for an operating life of 13 or 20 reactor full-power years (RFPYs), which may be selected by the plant owner. Section 7.2.3 of Chapter 4 requires the PWR control rod assemblies to be designed for a minimum operating lifetime of 15 RFPYs with an objective of 20 RFPYs. These requirements are beyond the operating experience data of the current LWRs. To support the desired extended operating life of the control blades and control rod assemblies, each ALWR design application will need to include sufficient performance data to demonstrate that irradiation effects, including material hardening, absorber depletion, and swelling, will not impair structural integrity."	Agree	(DSER) See Abstract	

NRC Review

NRR/SRXB G. Hsii

Last Updated: 7/14/92

P.1.V-15 | **III.A** | **Status:** Open | **Next Action:** ALWR/NRC

safety classification (4.3.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-2) "General Design Criterion (GDC) 1, "Quality Standards and Records," of 10 CFR Part 50, Appendix A, requires that nuclear power plant structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Regulatory Guide (RG) 1.26, "Quality Group Classification and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants," is the principal document used by the staff in its review of this subject. However, the ALWR Requirements Document proposes the use of ANSI/ANS 51.1, "Nuclear Safety Criteria for the Design of Stationary PWRs," and ANSI/ANS 52.1, "Nuclear Safety Criteria for the Design of Stationary BWRs" as an alternative way of complying with RG 1.26. The staff has not completely endorsed these two industry standards for the evolutionary ALWR and concludes that, because of the unique designs of the passive ALWRs, neither of these standards will be applicable to these plants. As discussed in Section 4.31 of Appendix B to Chapter 1 of the Passive Requirements Document, SRP Section 3.2.2, "System Quality Group Classification," also may not be completely applicable. Many of the passive systems that EPRI has classified as non-safety-related are similar to systems in current LWR designs are classified as safety-related. However, these passive plant non-safety-related systems are relied on to provide defense-in-depth capabilities to serve as the first line of defense in the event of transients and plant upsets to reduce challenges to the passive safety systems. Since these important non-safety-related systems are not required by EPRI to meet safety-grade criteria, the staff is trying to establish functional performance requirements, acceptance criteria, and other appropriate design guidelines to ensure that such systems have adequate functional capability and will remain operable when called on. Therefore, the staff's positions on quality group classifications of specific structures, components, and equipment may not be available until the above criteria have been established. In addition, in order for the staff to develop positions on quality group classifications, applicable piping and instrumentation diagrams should be available. However, this level of detail is not appropriate for the Requirements Document."	See Policy Issue III.A	(DSER) See Abstract	See Policy Issue III.A

NRC Review

NRR/EMEB

Last Updated: 7/14/92

ALWR/NRC OPEN ISSUES

P.1.V-16

Status: Open

Next Action: NRC

seismic qualification by experience (4.3.2, 4.8.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-48) "Current NRC guidance (Regulatory Guide 1.100, Revision 2) recognizes the use of experience data as a means of seismic qualification of equipment. However, the earthquake experience data base methodology described in IEEE-344-1987 -which, as stated in Regulatory Guide 1.100, Revision 2, is to be evaluated by the staff on a case-by-case basis - is different from the detailed criteria and approach provided in the SQUG Generic Implementation Procedure (GIP). The staff does not accept the GIP as a qualification procedure. Rather, it is a verification procedure and is intended to be used only by the older operating plants under the A-46 resolution. Since the staff does not accept the GIP as a qualification procedure, it is not applicable to newer operating reactors or future ALWR plants. The development of the GIP verification procedures and criteria was not necessarily based on the required elements of IEEE 344-1987 or staff requirements for newer operating reactors. Thus, a significant portion of the data base in the A-46 methodology is not applicable to future ALWRs. Therefore, consistent with RG 1.100, Revision 2, the staff will evaluate the use of experience data on a case-by-case basis for plant-specific applications (see Section 3 of Chapter 2 of this report for a specific application of experience data)."	The URD has been updated (Rev 2). In section 4.8.1.8, there is no longer a reference to SQUG but to Reg. Guide 1.100, Rev 2 and to the fact that the experience data method may require NRC evaluation for acceptance on a case by case basis. It is planned that an industry guideline document for using experience data will be developed under EPRI management. It will be submitted for NRC review and approval. This will then become the basis for approval in lieu of case by case approval.	(DSER) See Abstract	NRC review Rev 2 and continue the dialog as necessary

NRC Review

NRR/ESGB

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-17

Status: Open

Next Action: NRC

non-seismic building structures (4.3.2.3, 4.7.2.10)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-6) "Revision 0 of Section 4.3.2.3 of Chapter 1 of the Passive Requirements Document requires that non-seismic (NS) building structures be designed to the Zone 2A specification in the Uniform Building Code (UBC) with an importance factor of 1.25 assigned to the structures. In its letter dated April 24, 1991, the staff questioned the use of Zone 2A, which according to the UBC seismic zone map, is lower than the specifications for many regions in the United States. In its July 2, 1991, response, EPRI noted that the UBC Zone 2A specification is intended solely to provide a high degree of investment protection for NS items. However, since many regions in the United States are designated as UBC seismic Zone 2B or higher, the use of the Zone 2A specification in these zones is not acceptable for the design of any NS items. The staff will require the use of Zone 2B of the UBC in an application for final design approval and design certification."	The requirements of Zone 2A are compatible with areas in which plants are expected to be sited. Zone 2B is too conservative. Furthermore, design of non-safety structures need not be NRC controlled.	(DSER) See Abstract	NRC to reconsider position

NRC Review

NRR/ESGB

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-18

Status: Closed(Cert) **Next Action:** none

structural design and construction codes (4.4, 4.4.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-7) "Section 4.4 of Chapter 1 of the Passive Requirements Document provides EPRI-proposed requirements relative to the applicability of major design and construction codes, industry standards, and regulatory positions to the ALWR passive plant design. Tables 1.4-2 and 1.4-3 in Chapter 1 list industry technical standards and major structural design and construction codes, respectively, that are applicable to the ALWR. Several of these standards have not been endorsed by the staff and should not be used as the basis for plant design and construction. In its letter dated August 1, 1991, EPRI agreed to revise Section 4.4 to state that the use of applicable structural design and construction codes and industry standards that conflict with NRC positions will be resolved by the plant designer with the NRC and the resolution will be fully documented. The intent of this requirement was to ensure that the staff's reviews of applications for final design approval and design certification for passive plants will be conducted using acceptance criteria that include those codes and standards most recently approved by the NRC. In Revision 2 to Section 4.4 of Chapter 1, this requirement was revised in accordance with the staff's request. The staff concludes that this commitment is acceptable. The staff will evaluate this matter during its review of an individual FDA/DC or COL application."	Agree	(DSER) See Abstract	

NRC Review

NRR/ESGB

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ALWR/NRC OPEN ISSUES

P.1.V-19	I.M	Status: Open	Next Action: ALWR

elimination of operating-basis earthquake from design (4.4.3, 4.7.3, Appendix B)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-10) "In SECY-90-016 and the draft policy paper on passive plants dated February 27, 1992, the staff stated that it agrees that the OBE should not control the design of safety-related systems. As a result, the staff is currently involved in the rulemaking p ocess for Appendix A to 10 CFR Part 100 to decouple the OBE from the SSE in siting considerations. The staff is also evaluating the possibility of redefining the OBE in order to satisfy the OBE's function without explicitly analyzing responses. This change would diminish the role of the OBE in design by establishing a level that, if exceeded, would require that the plant be shut down for inspection activities. The staff agrees in principle with EPRI regarding the deletion of the OBE from plant design. However, certain issues related to the treatment of earthquake cycles for piping and equipment fatigue evaluations, seismic anchor motion effects, postulated pipe break location criteria, and concrete structure design need to be adequately resolved as a direct consequence of eliminating the OBE from design. The elimination of the OBE from design would require all current OBE design-related checks to be performed for the SSE. The staff is developing various alternatives with the industry to revise the codes and standards when design-related checks are based on the OBE. Resolution of these issues may result in staff recommendations for changes in applicable ASME Code, Section III rules. Therefore, the staff concludes that the elimination of the OBE from design is acceptable. However, the details of how current OBE-related design checks will be performed using the SSE will be resolved between industry and the staff through the appropriate code-related activities or supplemental regulatory guidance. The supplemental regulatory guidance could be in the form of revised SRP sections or the ITAAC (inspections, tests, analyses, and acceptance criteria). The elimination of the OBE from design would require an exemption from the current regulations until the final to pertaining to Appendix A to 10 CFR Part 100 is approved. In the interim, the specification of the OBE ground motion remains an option for ALWR final design approval and design certification applicants. The staff will evaluate application of those applicants that opt to eliminate the OBE from design in accordance with the forthcoming supplemental regulatory guidance mentioned above."	See Policy Issue I.M	(DSER) See Abstract	See Policy Issue I.M

NRC Review

NRR/ESGB

Last Updated: 7/14/92

ALWR/NRC OPEN ISSUES

P.1.V-20 | **Status:** Open | **Next Action:** NRC

definition of support group (4.4.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-11) "Section 4.4.3.3.4 of Chapter 1 states that the plant designer may use approved independent support motion response spectrum analyses techniques as a basis for seismic design and identifies this use as an exception to SRP Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment." The staff's position regarding a definition of "approved techniques" is that this method is only acceptable when used in accordance with the information and recommendations in Sections 2.3 and 2.4 of NUREG-1061, "Report of the U.S. NRC Piping Review Committee," Volume 4. As a part of this position, a support group is defined by supports that have the same time history input. This usually means all supports located on the same floor (or portions of a floor) of a structure. The staff concludes that Sections 4.4.3.3.4 and 4.7.3.4 of Chapter 1 should be revised to provide this commitment. In the interim, the staff will review individual applications for final design approval and design certification in accordance with the above position."	Agree. A URD change will be made.	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/EMEB J. Brammer

Last Updated: 8/6/92

Paragraph No.	Requirement	Rationale	Rev.
4.4.3	**Regulatory Positions (Continued)**	**Regulatory Positions (Continued)**	0
4.4.3.3.1	The evaluation of the dynamic effects of pipe ruptures postulated in SRP 3.6.2 will not be performed for those systems or analyzable portions of systems successfully evaluated according to leak-before-break criteria. See Section 4.5.5.2 for specific criteria.	The ALWR approach is consistent with NRC's "Broad Scope" change to GDC 4.	2
4.4.3.3.2	The Plant Designer shall not combine seismic and pipe rupture events for systems in which LBB is demonstrated. This is in exception to SRP 3.9.3.	LBB demonstrates that pipe rupture will not occur due to seismic loads.	0
4.4.3.3.3	The Plant Designer shall use approved realistic damping criteria in the analyses of buildings, structures and equipment.	More realistic SSE damping values for piping systems will lead to significant improvements in equipment support design and spacing. Regulatory Guide 1.84 permits the use of damping values in accordance with ASME Code Case N-411.	1
2 4.4.3.3.4	The Plant Designer may use approved independent support motion response spectrum analyses techniques as a basis for seismic design. This is in exception to SRP 3.9.2.	Envelope response spectrum analyses have been proven to be excessively conservative in many cases. (see 4.7.3.4)	~~0~~ 1
4.4.3.3.5	The Plant Designer may use spectral shifting analyses as an alternative to spectrum broadening. This is in exception to SRP 3.9.2 and Regulatory Guide 1.122.	Broadened response spectra artificially increase the total energy input for analytical models. NRC has conditionally accepted ASME Code Case N-397 specified for use on a case-by-case basis in Regulatory Guide 1.84. Code Case N-397 has been annulled and provisions inserted into ASME Section III, Subsection NCA, Appendix N.	2
4.4.3.3.6	In the analysis of vibratory loads with significant high frequency input, the Plant Designer may combine high frequency modal results by algebraic combination. Non-linear analysis may be used to account for gaps between pipes and supports for such loadings. This is in exception to Regulatory Guides 1.92 and SRP 3.9.2.	This is consistent with the recommendation of NRC Piping Review Committee in NUREG-1061; however, the NRC staff position is that acceptance of algebraic combination would be on a case-by-case basis. It is anticipated that on-going research will refine the definition of the transition between low and high frequency ranges.	2

Paragraph No.	Requirement	Rationale	Rev.
4.7.3	**Systems and Equipment (Continued)**	**Systems and Equipment (Continued)**	0
4.7.3.3	The spectral peak shifting procedures of ASME Section III, Subsection NCA, Appendix N may be utilized for design of systems with closely spaced structural modes and artificially broadened spectrum input. Modal and spatial components of response shall be combined in accordance with ASME III, Appendix N, except that algebraic combination shall be used for the responses of modes above the ZPA cutoff frequency for loadings having significant high frequency content.	Spectral peak broadening can create overly conservative seismic energy input to the system when two or more structural modes fall within the broadened peak. While the spectral peak broadening analyses may be appropriate for standard engineering designs, more sophisticated analyses are justified for "as-built" analyses. Enveloped peak responses, rather than combined responses, more accurately reflect the response of the system since only one mode at a time will be excited at the peak of the artificially broadened range. Exceptions to Regulatory Guides 1.92 and 1.122 will be required to implement these techniques; refer to Sections 4.4.3.3.5 and 4.4.3.3.6.	2
4.7.3.4	Independent support motion response spectrum analyses, including appropriate seismic anchor motion (SAM), may be used in lieu of a single envelope of input at all supports in order to minimize excess conservatism when piping systems cross several building floor levels. When this methodology is used, the recommendations in NUREG-1061, Volume 2, Section 2.4, shall be followed.	Alternative procedures for evaluating multiple supported systems are identified in NUREG/CR-3811. Grouping of spectrum input by attachment points mitigates the excess conservatisms present when an isolated higher elevation input dominates the analysis of the entire system. (Reference NUREG-1061, Vol. 4, Section 2.); a support group is defined by supports that have the same time history input.	2
4.7.3.5	Stiffness and frequency requirements on piping support structures shall be specified to justify the use of uncoupled dynamic analysis of piping. If physical limitations are such that these requirements make a design impractical, analytical models of the piping support structures may be included in the piping analysis.	ASCE *Standard for Seismic Analysis of Safety-Related Nuclear Structures* provides guidance concerning when integrated analyses are appropriate. It should be noted that PVRC does not recommend use of support stiffnesses in piping models as a standard technique for all analyses.	0
4.7.3.6	Equipment nozzle stiffnesses shall be appropriately considered.	Equipment nozzle flexibilities should be explicitly modeled in piping analyses to reduce the calculated loads and to improve the moments at the nozzle. Care must be taken to avoid iterative analyses due to changes in erection and layout.	0

ALWR/NRC OPEN ISSUES

P.1.V-21

Status: Closed(Cert) **Next Action:** none

use of Appendix N of ASME Code, Section III (4.4.3, 4.7.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-40) "ection 4.7.3.1 of Chapter 1 states that dynamic analysis techniques for safety class components will be in accordance with Appendix N of ASME Code, Section III. Appendix N is a nonmandatory appendix that is still evolving and does not currently agree with some staff positions. Therefore, it has not been endorsed by the staff, and the staff has no immediate plans to review this document. In its letter dated May 17, 1991, the staff requested EPRI to delete the reference to Appendix N and to reference applicable regulatory guides, Standard Review Plan sections, or staff-approved ASME Code Cases in the requirement portion of Section 4.7.3.1. In its letter dated August 1, 1991, EPRI stated that only the rationale portion of Section 4.7.3.1 would be changed and that this change would only address the use of Code Case N-397. The issue of Code Case N-397 is discussed in Section 4.4.3 of this report. Code Case N-397 is only one of several issues that are either currently in Appendix N or are being proposed for future addenda to this document and that have not been endorsed by the staff. Some of these issues are damping values, use of the load coefficient method, use of the independent support motion response spectrum method of analysis, and the nonexceedance probability level in Subsection N-1725 of Appendix N. EPRI's response is not acceptable. Therefore, the staff will evaluate this issue during its review of an individual application for final design approval and design certification in accordance with applicable Standard Review Plan sections in lieu of Appendix N to ASME Code, Section III."	Agree. Appendix N defines precisely the seismic analysis methodology. We understand that the NRC will analyze Appendix N within the FDA review, if referenced in a vendor's submittal.	(DSER) See Abstract	

NRC Review

NRR/EMEB J. Brammer

Last Updated: 7/14/92

P.1.V-22 | Status: Closed(Cert) | Next Action: none

analysis of vibratory loads with significant high-frequency input (4.4.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-12) "In Sections 4.4.3.3.6 and 4.7.3.3 of Chapter 1 and Section 2.1.1 of Appendix B to Chapter 1, EPRI states in various ways that in the analysis of vibratory loads (other than seismic) with significant high-frequency input (i.e., 33 to 100 Hz), the plant designer may combine high-frequency modal results by algebraic combination. This is a deviation from RG 1.92, "Combining Modal Responses and Special Components in Seismic Response Analysis," that the staff currently evaluates on a case-by-case basis. In Revision 2 of the Passive Requirements Document, EPRI added a qualification to the rationale portion of Sections 4.4.3.3.6 and 4.7.3.3 of Chapter 1, and to Section 2.1.1 of Appendix B to Chapter 1, that indicates that in analyses of vibratory loads with high-frequency input, if high-frequency modal results are combined by algebraic combination, the staff will review the methodology on a case-by-case basis. However, the staff does not agree that the plant designer will necessarily treat the rationale as a requirement. Therefore, this same qualification should be added to the requirement portion of Sections 4.4.3.3.6 and 4.7.3.3 in addition to Section 2.1.1 of Appendix B to Chapter 1. In the interim, the staff will review individual applications for final design approval and design certification in accordance with the above position."	• Agree it will be done on a case by case basis. • Modification of the URD is not needed because it's already written in Appendix B	(DSER) See Abstract	

NRC Review

NRR/EMEB J. Brammer

Last Updated: 7/14/92

ALWR/NRC OPEN ISSUES

P.1.V-23		Status: Closed(Cert)	Next Action:

use of nonlinear analysis to account for gaps between pipes and piping supports (4.4.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4.12) "IIn its letter dated May 17, 1991, the staff requested EPRI to revise the requirement portion of Sections 4.4.3.3.6 and 4.7.3.12 of Chapter 1 and Section 2.1.1.2 of Appendix B to Chapter 1 to require that if nonlinear analyses are used to account for gaps between pipes and piping supports subjected to vibratory loads with high-frequency input, such analyses must be submitted to the staff for review and approval before they are used. In its response to this RAI, EPRI stated that since this procedure was identified as an exception to SRP Section 3.9.2 in the requirement portion of the above sections, no further changes were required. The staff does not agree that merely identifying this procedure as an exception to the SRP is sufficient for a requirement. The staff position applies to the requirement portion of Sections 4.4.3.3.6 and 4.7.3.12 and to Section 2.1.1.2 of Appendix B to Chapter 1. Therefore, the staff will review individual applications for final design approval and design certification in accordance with the above position."	Agree it will be done on a case by case basis	(DSER) See Abstract	

NRC Review

NRR/EMEB J. Brammer

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-24 | | **Status:** Closed(Cert) | **Next Action:** none

probabilistic approach for changing existing loads and/or loading combinations (4.5.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4.13) "Section 4.5.1 of Chapter 1 of the Passive Requirements Document provides general requirements for loads and conditions including natural phenomena, site proximity man-made hazards, plant operating loads, and in-plant hazards. Section 4.5.1.2 of Chapter 1 states that, on a case-by-case basis, the plant designer may, with the approval of the NRC, develo, quantitative mechanistic design loads and combinations directly from design-basis events, using probabilistic methodology. The staff concludes that this is acceptable. However, it is not currently accepting a probabilistic approach as a basis for changing existing loads and/or loading combinations, and the loading combinations recommended in SRP Sections 3.7, 3.8, and 3.9 remain valid. The staff will address this issue during its review of an individual application for final design approval and design certification."	Agree	(DSER) See Abstract	

NRC Review

NRR/EMEB J. Brammer

Last Updated: 7/14/92

ALWR/NRC OPEN ISSUES

P.1.V-25

Status: Open

Next Action: NRC

recurrence interval for wind loadings (4.5.2.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-14) "The use of importance factor 1.11 for adjusting the recurrence interval from 50 to 100 years is suitable for the design of safety related structures because the application of an importance factor of 1.11 to calculate the wind speed for a 100-year recurrence interval is equivalent to the guidance in SRP Section 3.3.1, "Wind Loadings." For non safety related struc tures, the use of a 1.0 importance factor implies t at a 50 year recurrence interval is suit able. Howev he staff's interpretation of ANSI A58.1 1982 is that an extreme wind associated with a 50 year recurrence interval is suitable to calculate the wind speed only for Category I and IV structures. Since non-safety-related structures in an ALWR plant are more important than Category IV structures, the staff concludes that both safety- and non-safety-related structures should be designed for an extreme w d associated with a 100 year recurrence interval. The importance factor of 1.0 is not acceptable for non-safety-related structures that are important to safety (e.g., turbine build ing). EPRI has not provided adequate justification for its position. Therefore, the staff will address this item during its review of an individual application for final design approval and design certification."	Consistent with the relative importance of the safety related and non-safety related structures, different importance factors are proposed for the two classes of structures. It is noted that the 110 mph extreme wind speed specified in Chapter 1, Table 1.2-6, is extremely conservative for most of the potential sites. The specification of 110 mph as the design wind speed with the appropriate importance factors provides a conservaqtive design basis for both the safety related aa well as the non-safety related structures.	(DSER) See Abstract	NRC to review this response

NRC Review

NRR/PRPB J. Lee

Last Updated: 7/14/92

P.1.V-26

Status: Open

Next Action: ALWR

maximum ground water level (4.5.2.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-15) ""Maximum ground water level The Passive Requirements Document requires the maximum ground water level to be 2 foot below grade. This requirement is not acceptable; the maximum ground water level should be at grade. EPRI has not provided adequate justification for its position. Therefore, the staff will address this item during its review of an individual application for final design approval and design certification."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/ESGB

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-27		Status: Open	Next Action: ALWR

precipitation for roof design (4.5.2.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-15) "Precipitation (for roof design) - The Passive Requirements Document specified a maximum rainfall rate of 10 in./hr and a maximum snow load of 50 pounds per square foot (psf). In the DSER dated September 1987 on Chapter 1 of the Requirements Document for evolutionary plants, the staff raised a concern that the rainfall rate of 10 in./hr was much too low, citing that the probable maximum precipitation (PMP) in a five-minute interval over one square mile is 6.3 inches at the Great Lakes area and 6.2 inches along the Gulf Coast. The revised Table 1.2-6 specifies a higher 1-square mile, 1-hour PMP of 19.4 inches, together with a 1-square-mile, 5-minute PMP of 6.2 inches. The 5-minute PMP value appears reasonable; however, that might exclude a number of Great Lakes area sites. EPRI should use the SRP guidelines and relevant RGs for developing an adequate structural and flood-prevention design basis for the PMP. EPRI has not provided adequate justification for its position. Therefore, the staff will address this item during its review of an individual application for final design approval and design certification."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/PRPB J. Lee

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P.1.V-28

Status: Open

Next Action: ALWR

snow loading (4.5.2.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-16) "In the DSER dated September 1987 on Chapter 1 of the Requirements Document for evolutionary plants, the staff stated that the 50-psf snow load might limit sites to below 38° north latitude in some regions, citing, for example, the 69-psf snow load and 72-psf snow plus ice load that were used in the design of the Beaver Valley plant. However, in Table 1.2-6 of Chapter 1 of the Passive Requirements Document, the 50 psf snow load was retained. The staff concludes that this snow load is unacceptable. EPRI has not provided adequate justification for its position. Therefore, the staff will address this item during its review of an individual application for final design approval and design certification."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/PRPB J. Lee

Last Updated: 8/18/92

P.1.V-29

Status: Closed(COL)   Next Action: none

detailed quantification of soil parameters (4.5.2.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-16) "In its letter dated April 24, 1991, the staff indicated that Table 1.2 6 should give a range of soil properties to provide consistent guidance to the vendors of the standard plants and potential utilities. In its response dated July 2, 1991, EPRI stated that the level of effort needed to quantify more specific soil parameters was beyond the scope of the Passive Requirements Document, and that the ALWR objectives will be satisfied as long as the standard plant design is suitable for a large range of foundation siting conditions that fall within the envelope of parameters of Table 1.2 6. The staff will address this issue during its review of an application for a combined operating license (COL)."	Agree. The criteria will be checked later during siting.	(DSER) See Abstract	

NRC Review

NRR/ESGB

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-30		Status: Closed(COL)	Next Action: none

minimum margin against liquefaction (4.5.2.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-17) "In its letter dated April 24, 1991, the staff requested that EPRI develop evaluation guidelines regarding the minimum margin against liquefaction. In its letter dated July 2, 1992 (sic 1991? JDT), EPRI stated that the specific guidelines had not been developed for the Requirements Document, and a site-specific evaluation must be performed when a plant is to be founded on a soil site. Consistent with the scope and level of technical details included in the Requirements Document, the staff concludes that the guidelines for minimum margin against liquefaction potential may be addressed by the applicant for a COL as a site-specific issue if the plant is to be founded on a soil site, or if any structures are to be founded on soil having a liquefaction potential at sites with multiple soil conditions. Such guidelines should include a detailed evaluation of the liquefaction potential (as described in SRP Section 2.5.4, "Stability of Subsurface Materials and Foundations"), and consequences of liquefaction, of all subsurface soils, including the settlement of foundations. These evaluations will be based on soil properties obtained by state-of-the-art laboratory and field tests and involve application of both deterministic and probabilistic procedures."	Agree. The criteria will be checked later during siting.	(DSER) See Abstract	

NRC Review

NRR/ESGB

Last Updated: 8/6/92

P.1.V-31

Status: Closed( )

Next Action: none

external hazards evaluation (4.5.2.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-17) "In its letter dated April 24, 191, the staff requested EPRI to define the analyses or evaluation methods that will be used to evaluate hazards such as active faults, man-induced hazards, and soil stability. In its response dated July 2, 1991, EPRI noted that these issues were not applicable in the design of standard plants and should be considered in site-specific assessments, and that it anticipated that NRC-approved state-of-the-art analyses and evaluation methods will be used at that time. The staff will address this issue during its review of an application for a COL."	Agree. The criteria will be checked later during siting.	(DSER) See Abstract	

NRC Review

NRR/ESGB

Last Updated: 8/6/92

P.1.V-32 | I.M | **Status:** Open | **Next Action:** ALWR

number of full-stress cycles (4.5.2.4, 4.8.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-18) "In Section 4.5.2.4 of Chapter 1, the OBE was deleted for consideration in the design process. As discussed in Section 4.4.3 and Appendix B of Chapter 1 of this report, the staff is evaluating the effect of this change on current staff positions. This evaluation will address the requirement in Sections 4.5.2.4.4.1 and 4.8.1.1 of Chapter 1, which reduces the number of full-stress cycles for 1/2 SSE from 50 to 20. The results of this evaluation will be included in the supplemental regulatory guidance discussed in Section 4.4.3 of this DSER chapter. The staff will review an individual application for final design approval and design certification in accordance with the supplemental guidance."	The concern is generic and should be resolved in the context of the Requirements See Issue "I.M"	(DSER) See Abstract	See Issue "I.M"

NRC Review

NRR/ESGB P. Sobel

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-33		**Status:** Closed(Site)	**Next Action:** none

site-specific safe-shutdown earthquake (SSE) (4.5.2.4)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-19) "Although the design-basis SSE of an RG 1.60 spectrum with a zero period acceleration of 0.30g is sufficient for most potential sites in the United States, it may not envelop the ground motion for sites near seismically active areas in the Eastern and Central United States or sites in the Western United States, in addition to those along the California coast. The staff has observed that recordings of earthquakes in the Eastern United States possess more high-frequency (greater than 5 Hz) ground motion than those earthquakes whose records were used to develop the RG 1.60 response spectrum. This could limit the sites at which designs using 0.3g zero period RG 1.60 response spectrum could be located. The staff will review the site specific SSE with respect to the design basis at the time of siting."	Agree	(DSER) See Abstract	

NRC Review

NRR/ESGB P. Sobel

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-34

Status: Closed(Cert) **Next Action:** none

power spectrum density function of the time history (4.5.2.4)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-19) "In Table 1.2-6, the criterion for the SSE ground motion [illegible]e history (time history) is that the response spectra obtained from the time history envelop the design response spectra. Compared to SRP Section 3.7.1, the staff's position is that this criterion should also include the requirement that the power spectrum density (PSD) function of the time history envelop an approved target PSD function if a single time history is used. In addition, SRP Section 3.7.1 specifies a different acceptance criterion if multiple time histories are used. The staff requires that the time history comply fully with the SRP. The staff will address this during its review of an individual application for final design approval and design certification."	Agree	(DSER) See Abstract	

NRC Review

NRR/SRXB

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ALWR/NRC OPEN ISSUES

P.1.V-35		Status: Closed(Cert)	Next Action: NRC

design temperature (4.5.2.7)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-22) "Design temperatures are not included or discussed in Section 4.5.2 of Chapter 1. In Table 1.2 6 of Chapter 1 of the Requirements Document, the ambient temperature is expressed in terms of the maximum and minimum temperatures for both 1 percent exceedance probability and 0 percent exceedance probability. The staff is not certain how the ambient temperature values would be used when they are derived from a probabilistic method and are associated with certain probabilities of exceedance. Table 1.2 6 of the Requirements Document also provides a criterion requiring the site be such as to permit atmospheric heat rejection of cooling water system heat loads or to provide cooling water at the flow rates and temperatures to be specified by the plant designer to achieve certain probability-based cooling performance limits. To review the safety related water supply, the staff typically uses deterministic values based on worst 1 hour, 24 hour, and 30 day values of record. Therefore, the staff will use the deterministic approach to review an individual application for final design approval and design certification."	The method proposed by EPRI/ALWR is currently used and is based on the statistics done on the two hours (0 percent exceedence) or about three days (1 percent exceedence). The method proposed by the staff is the same (for 1 hour, 1 day) and no more deterministic. This does not appear to be an issue.	(DSER) See Abstract	NRC to consider removaing as an issue.

NRC Review

NRR/PRPB J. Lee

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-36 | **Status:** Open | **Next Action:** ALWR

protective against surface vehicle bombs (4.5.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-23) "EPRI stated that the wall capacity (thickness and reinforcement) will likely be dictated by seismic shear load rather than tornado loading, because of the 0.3g SSE requirement. The staff does not agree with this position. If minimum static capacities are less than those assumed in NUREG/CR-2462, the staff may need to revise its guidance on protection against surface vehicle bombs. The staff will address this issue during its review of an application for a COL."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/RSGB R. Dube

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-37

Status: Open

Next Action: NRC

design against internal-missile generation (4.5.5)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-26) "In Table B.1-2 of Appendix B to Chapter 1 of the Requirements Document, EPRI commits to comply with the staff review guidance in SRP Section 3.5.1.1, "Internally Generated Missiles (Outside Containment);" SRP Section 3.5.1.2, "Internally Generated Missiles (Inside Containment);" and SRP Section 3.5.1.4, "Missiles Generated by Natural Phenomena." The staff concludes that this commitment is acceptable. However, Section 4.5.5.4.1 of Chapter 1 states that ANSI/ANS 58.1, "Plant Design Against Missiles," will be used for guidance in meeting the requirements for internal-missile generation. The staff has not endorsed ANSI/ANS 58.1. Therefore, the staff concludes that where differences exist between the above SRP sections and ANSI/ANS 58.1, the guidance of the SRP sections should be used. If a plant designer identifies and provides justification for the differences, the staff will review the justification on a case-by-case basis and address the issue during its review of an individual application for final design approval and design certification."	We would like the NRC to specify which parts of ANSI/ANS 58.1 are in dispute.	(DSER) See Abstract	NRC to indicate disputed parts of standards.

NRC Review

NRR/EMCB

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ALWR/NRC OPEN ISSUES

P.1.V-38 | **Status:** Closed(Cert) | **Next Action:** none

design of concrete containment (4.6.1.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-27) "Section 4.6.1.1 of Chapter 1 requires that the design of concrete containments satisfy the load combinations shown in ASME Code, Section III, Division 2, Section CC, and the design of steel containments follows SRP Section 3.8.2, "Steel Containment." The staff concludes that the specified load combinations for the steel containment design conform with SRP Section 3.8.2 and are acceptable. Because SRP Section 3.8.1, "Concrete Containment," and RG 1.136, "Materials, Construction, and Testing of Concrete Containment," provide additional guidance on the use of ASME Code, Section III, Division 2, Section CC, for design of concrete containments but are not referenced in the Passive Requirements Document, the staff requested EPRI to confirm its position regarding compliance. In its letter dated July 2, 1991, EPRI indicated that while it is impractical to list all applicable RGs and SRP sections in individual paragraphs of the Passive Requirements Document, the requirement of compliance for the ALWRs is shown in Table B.1.1 of Appendix B to Chapter 1 of the Requirements Document. To ensure that the plant designer will use proper additional regulatory guidance for the load combinations, the staff's position is that concrete containment design will follow the guidelines of SRP Section 3.8.1 and RG 1.136. The staff will evaluate this issue during its review of an individual application for final design approval and design certification."	Agree	(DSER) See Abstract	

NRC Review

NRR/ESGB

Last Updated: 7/14/92

P.1.V-39

Status: Closed(Cert) Next Action: none

load combinations for Category I buildings and structures (4.6.1.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-28) "Section 4.6.1.2 of Chapter 1 requires the design of other seismic Category I reinforced concrete and steel structures to satisfy the load combinations specified in American National Standards Institute/American Concrete Society (ANSI/ACI) 349 and ANSI/American Institute of Steel Construction (AISC) N690, respectively. In its letter dated April 24, 1991, the staff noted that RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants," and SRP Sections 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments," and 3.8.4, "Other Seismic Category I Structures," provide additional guidance on the use of ANSI/ACI 349 for concrete Category I buildings and that the NRC has not approved the use of ANSI/AISC N690 for steel Category I structures. In its response of July 2, 1992, EPRI indicated that while it is impractical to list all applicable RGs and SRP sections in individual paragraphs of the Passive Requirements Document, the requirement of compliance for the ALWRs is shown in Table B.1 1 of Appendix B to Chapter 1 of the Passive Requirements Document. To ensure that the plant designer will use proper additional regulatory guidance for load combinations, the staff position is to require adherence to SRP Sections 3.8.3 and 3.8.4 and RG 1.142. The staff will evaluate compliance during its review of an individual application for final design approval and design certification."	Agree	(DSER) See Abstract	

NRC Review

NRR/ESGB

Last Updated: 7/14/92

ALWR/NRC OPEN ISSUES

P.1.V-40		Status: Open	Next Action: NRC

design of Category I steel structures (4.6. .2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-28) "EPRI proposes to use ANSI/AISC N690 for the design of Category I steel structures. The acceptability of using this code is uncertain because this code has not been reviewed and approved by the staff. Therefore, the staff will evaluate this issue during its review of an individual application for final design approval and design certification."	ANSI/AISC N690 defines precisely the design of Category 1 steel structures. NRC should review it and identify specific areas of disagreement.	(DSER) See Abstract	NRC to identify specifics.

NRC Review

NRR/ESGB

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ALWR/NRC OPEN ISSUES

P.1.V-41		Status: Open	... Action: ALWR

combination of pipe rupture loads with seismic loads for seismic Category I structures (4.6.1.3, 4.6.1.4)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-32) "In summary, the staff concludes that eliminating the combination of pipe rupture loads (global effects) with seismic loads for the containment and other seismic Category I structures is not acceptable. Furthermore, the proposal to decouple LOCA and SSE loads for equipment and systems is not acceptable at this time because of the inconsistency that would be created in the SRP and the insufficient technical bases to extend the decoupling to structures. The staff will evaluate this issue during its review of an individual application for final design approval and design certification."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/EMEB J. Brammer

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P.1.V-42

Status: Open

Next Action: ALWR

combination of loss-of-coolant-accident and SSE loads (4.6.1.7)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-34) "This position allows, in principle, the elimination of load combinations (except an SSE plus LOCA) with a probability of occurrence less than 1.0E-6 per reactor-year. However, such an elimination will be approved by the staff only when a specific design applicant justifies its contention with specific examples of how and where the design is governed and a comparison of structural margins with and without the load combination in question. The staff will evaluate this issue during its review of an individual application for final design approval and design certification."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/EMEB J. Brammer

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ALWR/NRC OPEN ISSUES

P.1.V-43		Status: Open	Next Action: ALWR

load combinations for safety-related portions of the plant (4.6.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-34) "Sections 4.6.2.3 and 4.6.2.5 and Tables 1.4-5 and 1.4-7 of Chapter 1 have eliminated the loading combination of SSE and LOCA on the basis of recommendations given in NUREG-1061, Volume 4. NUREG-1061 recommends the elimination of this loading combination only for piping systems in the majority of PWRs. The staff has not endorsed this recommendation. The current staff position, as stated in Appendix A to SRP Section 3.9.3, is that SSE and LOCA loads should be combined and included in the Service Level D loads. This position is applicable to both BWRs and PWRs. In its letter dated May 17, 1991, the staff requested that the rationale portion of Sections 4.6.2.3 and 4.6.2.5 and load combinations in Tables 1.4-5 and 1.4-7 be revised to agree with the above staff position. In addition, a note should be added to Tables 1.4-5 and 1.4-7 to state that the method of combination of dynamic responses to loads is in accordance with NUREG-0484, "Methodology for Combining Dynamic Responses," Revision 1, dated May 1980. In its letter dated August 1, 1991, EPRI stated that the SSE and LOCA loads are not combined because each event is of very low probability, and they are unrelated. The staff agrees that EPRI's position could be applicable only to piping systems in the majority of PWR plants. However, the staff's position is based on the requirements of GDC 2 of Appendix A to 10 CFR Part 50, which states that all structures, systems and components be designed to withstand the effects of appropriate combinations of normal and accident conditions with natural phenomena. Historically, the staff has interpreted GDC 2 as requiring that the effects of the SSE and LOCA be combined for the design of all safety-related portions of the plant. Any change in this interpretation requires either an exemption from or a revision of GDC 2. Therefore, the staff position remains as stated above and as reflected in SRP Section 3.9.3. As a result, the staff will review an individual application for final design approval and design certification in accordance with this position."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/EMEB J. Brammer

Last Updated: 8/19/92

ALWR/NRC OPEN ISSUES

P.1.V-44

Status: Open

Next Action: NRC

dynamic analysis techniques (4.7.2.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4 37) "Section 4.7.2.3 of Chapter 1 requires that dynamic analysis techniques comply with ASCE 4 86, as well as other applicable codes and standards, and be qualified and proven. In its letter dated April 24, 1991, the staff commented that the NRC has not accepted all analysis techniques in ASCE 4 86. In its response of July 2, 1991, EPRI stated that ASCE 4 86 is intended to supplement the overall criteria and methodology specified in RGs and SRP sections, and that Table B.1 1 of Appendix B to Chapter 1 of the Passive Requirements Document confirms EPRI's commitment to comply with the regulatory positions, except for those analysis techniques associated with the optimization subjects. This response and the requirements in this section of Chapter 1 are not acceptable because ASCE 4-86 has not been reviewed and approved by the staff. Therefore, the staff concludes that the SRP and RG guidelines should be used for the plant analysis and design of future ALWRs. Plant designers proposing to use ASCE 4-86 should submit a request for the staff review and approval on a case-by-case basis. The staff will evaluate this issue during its review of an individual application for final design approval and design certification."	• ASCE 4-86 is intended to supplement the overall criteria and methodology specified in Reg Guides and the SRP • NRC should review it and define specific areas of disagreement	(DSER) See Abstract	NRC review ASCE 4-86 and continue dialog as needed

NRC Review

NRR/ESGB

Last Updated: 8/6/92

P.1.V-45

Status: Closed(Cert)

Next Action:

methodology for generating of design spectra or time histories (4.7.2.5)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-37) "Section 4.7.2.5 of Chapter 1 requires that the generation of design response spectra or time histories be based on methods that minimize unnecessary conservatism. Since the Passive Requirements Document did not indicate the need for NRC approval of methods such as the spectrum-to- spectrum generation procedure, the staff requested a complete discussion of the limitations and verifications for such procedures. In its response of July 2, 1991, EPRI did not address the NRC concern sufficiently. The staff position is that all analysis methods used for licensing basis and safety margin basis must be approved by the NRC staff. The staff will evaluate this issue during its review of an individual application for final design approval and design certification."	Agree	(DSER) See Abstract	

NRC Review

NRR/ESGB

Last Updated: 7/14/92

P.1.V-46

Status: Closed(Cert) **Next Action:** None

structural damping values (4.7.2.6)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-38) "Section 4.7.2.6 of Chapter 1 requires that structural damping values be based on confirmed test results, rather than conservative assumptions, whenever such data are available. In its letter dated April 24, 1991, the staff requested the basis for using less conserva tive damping values. In its response of July 2, 1991, EPRI cited cable trays and hangers as an example for which extensive tests have been done by some utilities to generate more realistic damping values than those provided in RG 1.61. The staff position of using structural damping for the design of structures is described in RG 1.61. However, the staff will not exclude the use of structural damping based on test results. Therefore, the staff will evaluate the structural damping values during its review of an individual application for final design approval and design certification."	Agree	(DSER) See Abstract	

NRC Review

NRR/ESGB

Last Updated: 7/14/92

P.1.V-47

Status: Open

Next Action: ALWR

masonry walls in Category I buildings (4.7.2.7)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-38) "Section 4.7.2.7 of Chapter 1 requires that masonry walls used as temporary or permanent partitions in Category I buildings be engineered as a substructure of the building, with consideration given to the effect they could have on safety-related items, and that they be designed according to applicable requirements of the Uniform Building Code (UBC). The use of the UBC for the design of masonry walls in Category I structures and the use of masonry welds in Category I building deviate from SRP guidelines. The staff concludes that EPRI has not provided sufficient justification for this deviation; therefore, the SRP guidelines should be used for the design of masonry walls located in the Category I structures of the future ALWRs. In addition, the staff strongly discourages the use of masonry walls in Category I structures as load-carrying members or non-load-carrying members. The staff will evaluate this issue during its review of an individual application for final design approval and design certification."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/ESGB

Last Updated: 8/18/92

P.1.V-48

Status: Open

Next Action: NRC

use of expansion anchor bolts - compliance with Office of Inspection and Enforcement Bulletin 79-02 (4.7.2.8, 4.7.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-38) "Section 4.7.2.8 of Chapter 1 specifies the use of the direct bearing or undercut type of anchor bolts to ensure the ductile behavior of the bolt when high capacity is needed and the use of wedge and sleeve anchors for small loads. In its letter dated April 24, 1991, the staff questioned the use of expansion bolts for all safety significant applications and encouraged qualification testing under field conditions. Where the expansion anchors are used, the NRC requires the use of the conservative safety factors of Inspection and Enforcement Bulletin (IEB) 79 02 to account for uncertainty in field installation. In its response of July 2, 1991, EPRI acknowledged that it is intended to use expansion bolts only when necessary and that the expansion bolts will be of the undercut type (e.g., Maxibolts) in lieu of friction type. EPRI also noted that the conservative safety factors of IEB 79 02 are intended for friction type expansion anchors and may not apply to Maxibolts. The response is not acceptable because the issue of uncertainty in field installation was not addressed and there is no assurance that the IEB 79-02 safety factors are not applicable to Maxibolts. Therefore, plant designers should submit to the NRC staff the safety factors they propose to use for the capacity of the Maxibolts. The staff will evaluate this issue during its review of an individual application for final design approval and design certification."	We agree that the 79-02 inspection methods and safety factors are applicable to wedge or sleeve type anchors, which are permitted only for small loads. For applications requiring substantial pullout capacity Maxibolts would be used only if embedded anchorage was not preengineered. The Maxibolt installation procedure includes a tensioning of each bolt installed which serves as a proof test of the pullout capacity. For this reason, additional testing of Maxibolts is not required.	(DSER) See Abstract	NRC to review this response

NRC Review

NRR/EMZB J. Brammer

Last Updated: 8/6/92

P.1.V-49

Status: Open

Next Action: ALWR

stability of shell-type structures under compression (4.7.2.9)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-39) "Section 4.7.2.9 of Chapter 1 requires that the potential for global and local shell buckling be considered for shell type structures under compression. In addition, it requires that, after appropriate consideration of the various uncertainties in materials, erection tolerances, and load description, a minimum factor of safety be maintained for all load combinations. In Revision 2 to the Passive Requirements Document, the minimum factor of safety was based on ASME Code, Section III, Subsection NE, and supplemented by Code Case N 284. Because Code Case N 284 provides lower safety factors against shell buckling than Section NE, the staff requested that EPRI alert the plant designer regarding the application of Code Case N 284 to asymmetric containments with large openings or provide specific conditions under which N 284 can be used. In its response of July 2, 1991, EPRI stated that experienced professional containment vessel designers, working under careful review of the utility owners and regulators, will not apply Code Case N 284 where provisions of the code case do not apply. EPRI also stated that it was beyond the scope of the Passive Requirements Document to explain the limitations of the specific code case. This is not acceptable to the staff. The staff concludes that NE requirements should be used for the evaluation of shell-type structures. As for Code Case N-284, the staff will evaluate its applicability during its review of an individual application for final design approval and design certification."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/ESGB

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-50

Status: Open

Next Action: NRC

use of ASME Code Cases N-411 and N-420 in same analysis (4.7.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-41) "Sections 4.7.3.2 and 4.7.3.11 of Chapter 1 of the Requirements Document allow the plant designer to use ASME Code, Section III Code Cases N-411 and N-420 unconditionally. In its letter dated May 17, 1991, the staff requested that the following sentence be added to this requirement: "ASME Code Cases N-411 and N-420 may only be used in separate analyses and as further conditioned in RG 1.84." In its letter dated August 1, 1991, EPRI stated that since its intent to comply with RG 1.84 is indicated in Appendix B of Chapter 1 of the Passive Requirements Document, this additional sentence is unnecessary. The staff's understanding of the Passive Requirements Document is that a requirement in any section could override such a commitment in Appendix B. Therefore, this response is unacceptable and the staff's position remains as stated in its letter of May 17, 1991. The staff will evaluate this issue during its review of an individual application for final design approval and design certification in acc rdance with the above position."	There is no intention that committments in Appendix B could be overridden by other parts of the URD. Any and all contradictions that are brought to our attention will be eliminated and if the NRC reviewer is aware of any such contradictions, he should tell us. Otherwise, the review can assume Appendix B committments are bona fide commmittments.	(DSER) See Abstract	NRC review this response.

NRC Review

NRR/EMEB J. Brammer

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-51

Status: Open

Next Action: NRC

use of ASME Code Case N-411 (4.7.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-41) "In its letter dated May 17, 1991, the staff requested that Section 4.7.3.8 be revised to clarify the use of a single damping value for both the OBE and the SSE. In its letter dated August 1, 1991, EPRI stated that the ALWR program has deleted the OBE in the design process and that damping values in RG 1.61 for the SSE will be applicable to structures and systems except for piping, for which ASME Code Case N-411 is applicable. The elimination of the OBE is discussed in Section 4.4.3 of this report. The resolution of this issue may affect Section 4.7.3.8 of Chapter 1. The use of Code Case N-411, as stated in EPRI's response of August 1, 1991, is not completely acceptable. The staff requested that the requirement portion of Section 4.7.3.8 be revised to include a requirement that Code Case N-411 can be used only as conditioned by RG 1.84. In its response, EPRI stated that since Table B.1-2 of Appendix B to Chapter 1 indicates a commitment to comply with RG 1.84, this revision was unnecessary. The staff's understanding of the Passive Requirements Document is that a requirement in any section could override such a commitment. Therefore, this portion of the response is unacceptable and the staff position remains as stated in its letter of May 17, 1991. The staff will evaluate this issue during its review of an individual application for final design approval and design certification, assuming that the requirement in Section 4.7.3.8 does not override the commitment to RG 1.84 in Appendix B to Chapter 1."	There is no intention that committments in Appendix B could be overridden by other parts of the URD. Any and all contradictions that are brought to our attention will be eliminated and if the NRC reviewer is aware of any such contradictions, he should tell us. Otherwise, the review can assume Appendix B committments are bona fide commmittments.	(DSER) See Abstract	NRC to review this response

NRC Review

NRR/EMEB J. Brammer

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-52		Status: Ope	Next Action: ALWR

construction of core support structures (4.7.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-43) "Section 4.7.3.23 of Chapter 1 states that core support structures will be designed to the criteria specified in ASME Code, Section III, Subsection NG. In its letter dated May 17, 1991, the staff requested that this requirement be revised to read: "Core support structures will be constructed to the criteria specified in ASME Code, Section III, Subsection NG, where "construction" is as defined in ASME Code, Section III, NB/NC/ND-1100(a)." In its letter dated August 1, 1991, EPRI agreed with the staff's request, except that the requirement still contains the words "designed to" rather than "constructed to," which is not completely acceptable. During its reviews of individual applications for final design approval and design certification, the staff will require that core support structures be constructed to the rules of ASME Code, Section III, Subsection NG, where "construction" is either as defined above or as defined in ASME Code, Section III, Subsection NG-1110."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/EMEB J. Brammer

Last Updated: 8/18/92

P.1.V-53

Status: Open

Next Action: ALWR

design fatigue curves (4.7.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.4-44) "Section 3.3 of Chapter 1 of the Passive Requirements Document states that the plant design life for the ALWR will be 60 years. This proposed design life raises questions relative to the margins available in the current ASME fatigue design curves. These margins were established almost 30 years ago and were obtained from best-fit curves of fatigue test data by applying a factor of either 2 on stress or 20 on cycles, whichever was more conservative at each point. These factors were originally intended to cover such effects as environment, size effect, and scatter of data. However, on the basis of limited data currently available, the staff concludes that these margins may not be sufficient to account for variations in the original fatigue test data as a result of various environmental effects. In its letter dated May 17, 1991, the staff requested a commitment in Section 4.7.3 of Chapter 1 to consider such effects in the designs of applicable ASME Code, Class 1 systems, components, and equipment. In its letter dated August 1, 1991, EPRI stated that if additional data or research results yield findings requiring changes to the current ASME fatigue design curves, the Code consensus process will provide the proper vehicle to affect such findings. The staff does not agree with all of the discussions in this response and concludes that the above commitment is not completely acceptable. The ASME Code curves may not be revised for many years. Therefore, the staff's position is that until these curves are revised, all ALWRs and all plants applying for license renewal should propose appropriate design fatigue curves that will be reviewed by the staff. For the Requirements Document, a commitment to this position would be sufficient. Pending such a commitment, the staff will evaluate this issue during its review of an individual application for final design approval and design certification."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/EMEB J. Brammer

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-54		Status: Closed(Cert)	Next Action: none

use of zinc to reduce radiation fields (5.2.7)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.5-5) "To reduce general radiation fields resulting from the presence of cobalt-60 in the oxide layer of the RCS piping, zinc additions may be made to the coolant in BWRs in limited, controlled amounts. Zinc injection reduces the radiation fields by replacing the cobalt with zinc in the piping oxide layer. One of the side effects of zinc injection is the creation of zinc-65, which increases piping dose rates and requires special consideration during radioactive disposal. EPRI is investigating a way to solve this problem by using a zinc isotope depleted in zinc-64. The staff will review this issue again at the vendor application stage to determine what advances have been made in this area."	Agree	(DSER) See Abstract	

NRC Review

NRR/PRPB C. Hinson

Last Updated: 7/17/92

ALWR/NRC OPEN ISSUES

P.1.V-55		Status: Open	Next Action: ALWR

grinding controls for PWRs (5.3.1.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.5-9; "The staff considers these requirements acceptable. The implementation of these requirements will ensure that wrought austenitic stainless steel will perform in service as designed. However, the staff requires that the grinding controls also be applied to PWR applications. The staff will evaluate this issue during its review of an application for a COL."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/EMCB G. Georgiev

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-56

Status: Open

Next Action: ALWR

use of Alloy 600 (5.3.1.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.5-11) "The staff considers these requirements acceptable. The implementation of these requirements will ensure that Ni-Cr-Fe alloys will perform in service as designed. However, the staff will require that the applicant for any standard design application identify the use of Alloy 600 and provide described information concerning its use. Those applications will be reviewed and approved by the staff on a case-by-case basis. In addition, the use of other Ni-Cr-Fe alloys such as Alloy 690 or 800 should be considered in applications for which primary water stress corrosion cracking (PWSCC) is a concern. These applications also will be reviewed on a case-by-case basis."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/EMCB G. Georgiev

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-57

Status: Open

Next Action: NRC

affect(sic) of fabrication processes on intergranular stress corrosion cracking (5.3.1.8)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.5-17) "The staff considers these requirements acceptable. The implementation of these requirements will ensure that austenitic stainless steel will be resistant to IGSCC in service. The staff considers it important that adequate field and shop fabrication processes be used to minimize the sensitization of materials to IGSCC and will review specific controls as a part of the COL process."	The NRC staff should identify the "specific controls" that will be reviewed as part of the COL process. Review criteria should be identified as well.	(DSER) See Abstract	NRC review this response.

NRC Review

NRR/EMCB G. Georgiev

Last Updated: 8/6/92

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ALWR/NRC OPEN ISSUES

P.1.V-58		Status: Open	Next Action: NRC

selection of seal, gaskets, and protective coatings (5.3.5)

Abstract	Industry Position	NRC Position	Action Description
(DSER p 1.5-20) "Section 5.3.5 of Chapter 1 of the Passive Requirements Document requires that the plant designer conduct a program for evaluating the effects of intended products on other ALWR components under normal and postaccident conditions. For each product evaluated, the designer will provide recommendations and limits for its use in the steam supply systems and other applications in the ALWR. Information from component vendors will be relied upon only when substantiated by operational experience. The staff considers these requirements acceptable. The implementation of these requirements will ensure that seal, gaskets, and protective coatings are selected based on operational experience. However, the staff will require that the applicant specify the specific ANSI standard covering this subject. The staff will evaluate these components during its review of an individual application for final design approval and design certification."	The URD will be revised to require the designer to specify the ANSI standard (if any) covering , the particular product and the applicable process.	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/EMCB G. Georgiev

Last Updated: 8/6/92

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Paragraph No.	Requirement	Rationale	Rev.
5.3.5	**Seals, Gaskets, Packings, Sealants, Paints and Protective Coatings, Lubricants and Hydraulic Fluids, Cleaning, Packaging and Storage Materials**	**Seals, Gaskets, Packings, Sealants, Paints and Protective Coatings, Lubricants and Hydraulic Fluids, Cleaning, Packaging and Storage Materials**	0
	The Plant Designer shall conduct a program for evaluating the effects of intended products on other ALWR components under normal and post- accident conditions. ~~For each product evaluated, the Designer shall provide recommendations and limits for its use in the steam supply systems and other applications in the ALWR.~~ Information from component vendors should be relied upon only when substantiated by operational experience. The program should encompass the material categories of Table 1.5-1.	Experience has shown thorough evaluation to be essential in selecting the proper product from the great variety of materials available in the marketplace, many of which have contributed to down-time in operating reactors. Specific considerations include: deterioration with time (including shelf life), compatibility with materials in contact, leaching or outgassing of impurities, and resistance to environment.	0
	The Designer shall provide recommendations and limits, including references to specific ANSI Standards, if any, for each product evaluated for use in the Nuclear Steam Supply Systems and other applications in ALWR.	In addition to the variety of possible non-metallic materials used during design, there are many applications throughout ALWR fabrication, delivery, installation, operation, maintenance and testing which should be considered. This is further complicated by technology advancements for each product category. The plant designer's study of available product experience and qualifications, as well as placing similar requirements on component vendors and erectors, will help simplify the selection process and preclude use of undesirable products.	0

ALWR/NRC OPEN ISSUES

P.1.V-59	

Status: Open

Next Action: ALWR

aging of cable insulations and other electrical materials (5.3.6)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.5-20) "Section 5.3.6 of Chapter 1 of the Passive Requirements Document requires that materials used in the plant's electrical systems, particularly those used in safety-related applications, be reviewed for functional reliability during normal, abnormal, plant operation, and accident conditions. The fire-retardant characteristics of the materials used in the electrical systems will be addressed in order to minimize the probability of fire and the subsequent consequences should a fire occur.		(DSER) See Abstract	(ALWR) Respond to DSER

The staff concludes that this requirement acceptable. However, it is not clear that this requirement is sufficient to address such issues as aging of cable insulations and other electrical materials over the design life and full range of environmental conditions. The staff will evaluate this issue during its review of an individual application for final design approval and design certification.
"

NRC Review

NRR/SPLB G. Hubbard

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-60

Status: Open **Next Action:** ALWR

use of hydrogen water chemistry for the advanced BWR design (5.5.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.5-25) "Section 5.5.2.1 of Chapter 1 requires that the water chemistry design basis for BWR plant systems be in accordance with EPRI NP-4947-SR, "BWR Hydrogen Water Chemistry (HWC) Guidelines," 1987 Revision, and its subsequent revisions, and as supplemented by the guidelines provided in Table 1.5-1 of this report. The HWC specific control values given in EPRI's guidelines relative to recirculating piping (e.g., 230 mV) will apply to nozzles, components, and other non-replaceable components in the reactor vessel lower plenum. EPRI addresses the use of HWC for the advanced BWR design. The use of HWC at plants such as Hatch, Brunswick, and Duane Arnold has resulted in unexpectedly high operational and post-shutdown radiation levels in RCS piping. EPRI has acknowledged the potential drawbacks of using HWC and has stated that investigations are under way to identify a solution to some of the problems resulting from the use of HWC. A special evaluation will be made when considering carbon and low alloy material for reactor coolant service with less than 10 ppm oxygen as the result of HWC. The evaluation will include erosion/corrosion, radiation buildup, and pitting at shutdown. The staff will evaluate the issue of HWC use during its review of an individual application for final design approval and design certification to determine what advances have been made in this area."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/EMCB G. Georgiev

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-61		Status: Open	Next Action: NRC

plant-specific reliability assurance program (6.5)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.6-8) "The staff concludes that, with the exceptions noted above, the reliability and availability requirements established in the Passive Requirements Document are consistent with accepted industry practices and principles and do not conflict with current regulatory requirements and guidelines. They are, therefore, acceptable. However, by themselves, the reliability and availability requirements do not provide sufficient information for the staff to determine if the ALWR design referencing the EPRI requirements will adequately incorporate RAP considerations in a manner that will ensure plant safety and reliability. Therefore, applicants referencing the Passive Requirements Document will be required to provide sufficient information to demonstrate that their RAP will result in a plant that is designed and will perform in a manner that will ensure plant safety and reliability."	The RAP has been modified (Rev 3. of Ch 1, Section 6). We consider that the considerations in it ensure plant safety.	(DSER) See Abstract	NRC review Rev 3.

NRC Review

NRR/LPEB R. Correia

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-62		Status: Closed(COL)	Next Action: none

inspection of construction activities (7, 11.13)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.7-1) "The NRC has the statutory responsibility, regardless of construction schedule, to verify that the plant is constructed in accordance with the design documents tendered with the application for an operating license. The owners/builder must ensure that construction activities permit verification of the acceptability of the plant configuration in accordance with the requisite NRC Inspection Manual chapters. The staff will evaluate this during its review of a specific application for a COL."	Agree	(DSER) See Abstract	

NRC Review

NRR/PDST

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-63		Status: Closed(COL)	Next Action: none

installed operating-phase security system (7)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.7-2) "In Section 7.9.6 of Chapter 1 of the Passive Requirements Document, EPRI requires the utility to establish security boundaries as part of the startup testing program. In a letter dated May 13, 1991, EPRI agreed that the detailed construction and startup schedule will have to address NRC review and approval of the installed security system for the operating phase before first fuel loading, but that this milestone is beyond the scope of the ALWR Requirements Document. This will be addressed by the staff during its review of a specific application for a COL.	Agree	(DSER) See Abstract	

The staff expects that at least sixty days prior to loading fuel, a licensee for a COL will have confirmed that security systems and programs described in its physical security plan, safeguards contingency plan, and guard qualification and training plan have achieved operational status and are available for NRC inspection. Operational status means that the security systems and programs are functioning in entirety as they would when the reactor is operating and will remain so. The COL licensee's determination that operational status has been achieved must be based on tests conducted under realistic operating conditions of sufficient duration that demonstrated that the equipment is properly operating and capable of long-term, reliable operation, that procedures have been developed, approved, and implemented, and that personnel responsible for security operations and maintenance have been appropriately trained and have demonstrated their capability of performing their assigned duties and responsibilities."

NRC Review

NRR/RSGB R. Dube

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-64		Status: Closed(COL)	Next Action: none

reliability of modular construction (7)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.7-3) "Specific licensing criteria addressing modular construction have not been developed for nuclear power plant construction. Structures, systems, and components that are assembled using modular construction techniques must possess, as a minimum, the same degree of structural strength and reliability of such items provided in currently licensed plants that were constructed using current onsite construction techniques. Items to be considered include segmented rebar cage connections and in-containment steel/concrete sandwich-type shear walls for which there are no modular construction design criteria and for which test information is limited. Other areas of concern include the integrity of joints (including strength and ductility), seismic damping values and stiffness degradation in structural modules, quality assurance and quality control requirements for transportation and installation of modules, and the scope of the verification testing after the modules are installed. This will be addressed by the staff during its review of a specific application for a COL, should the applicant propose use of these techniques."	Agree	(DSER) See Abstract	

NRC Review

NRR/ESGB

Last Updated: 8/6/92

P.1.V-65

Status: C'oseJ(COL) | ıext Action: none

inspection and verification of security locks robotically (8.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.8-3) "The plant designer will perform an analysis to determine the effectiveness of using robotic applications in the ALWR. Inspection/surveillance functions will include reading of instruments and gauges, performing radiation surveys and measuring radiation levels, and taking ... surveys. EPRI states that the maintenance functions ... include steam generator inspection and maintenance, control rod drive removal, radwaste drum handling, spent fuel consolidation, equipment decontamination, and ... surveillance and maintenance tasks. The Requirements Document specifies that the ALWR will include design features such as wider doors and aisles, ramps, and modular construction of equipment and systems (for ease of equipment removal and replacement) to facilitate the use of robotic devices. Table 1.8-4 includes "verify security locks" as one of several functions to be evaluated by the plant designer as a candidate for robotic inspection and surveillance. However, in its letter of May 17, 1991, EPRI stated that details of the security functions to be performed and the consideration of replacing a security officer are outside the scope of the Requirements Document. This will be addressed by the staff during its review of a specific application for a COL."	Agree	(DSER) See Abstract	

NRC Review

NRR/RSGB R. Dube

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-66

Status: Open

Next Action: ALWR

compliance of design certification applications with Commission's [illegible] guidance (10)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.10-1) "Section 10.2 of Chapter 1 states that the ALWR will be designed to comply with the NRC regulatory requirements and guidance in effect on January 1, 1990, consistent with the commitments in Section 1 of Appendix B to Chapter 1 of the Passive Requirements Document. EPRI states that these requirements and guidance include applicable Commission regulations specified in Title 10 of the Code of Federal Regulations, general design criteria, NRC policy statements, regulatory guides, the Standard Review Plan, and other documentation that resolves unresolved and generic safety issues. Although the staff understands EPRI's need to "freeze" the requirements it addresses TO those in effect on January 1, 1990, the staff expects that the design certification applications will be in compliance with the Commission's regulations and guidance that are applicable and in effect at the time the certification is issued. The staff will evaluate this compliance during its review of an individual application for final design approval and design certification."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/PDST

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-67

Status: Open

Next Action: ALWR

issue resolution for final design approval and design certification reviews (10)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1 10-1) "In addition, issue resolutions that are different from those arrived at during the staff's review of the Passive Requirements Document may be developed as the staff completes its reviews of the detailed design information provided in the final design approval and design certification applications, and as these designs are litigated in the design certification hearings. Therefore, the staff expects that the ALWR plant designers will comply with issue resolutions adopted by the NRC staff during its reviews of the final design approval and design certification applications in accordance with the requirements of 10 CFR Part 52. The staff will evaluate this compliance during its review of an individual application for final design approval and design certification."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/PDST

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-68	II.L	Status: Closed(Cert)	Next Action: none

inspections, tests, analyses, and acceptance criteria (10)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.10-4) "The staff is developing additional guidelines for the scope and content of ITAAC, and is evaluating pilot ITAAC submittals based on the GE ABWR design. As described in SECY-91-178, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses," dated June 12, 1991, Tier 1 ITAAC will be at a level of detail corresponding to the Tier 1 design information of the certified design rule. The staff expects that the Tier 1 verification requirements will be high level in nature and will address the design at a system functional performance level of detail. Numerical acceptance criteria values will only be specified when failure to meet the stated acceptance criteria would clearly indicate a failure to properly implement the design. While including appropriate guidance on scope and content for ITAAC submittals in the Requirements Document would ensure that each design submittal provide a complete and adequate ITAAC package for staff review, ITAAC is clearly the responsibility of the plant designer during the design certification phase. The staff will evaluate the proposed ITAAC during its review of an individual application for final design approval and design certification and an application for a combined operating license."	See issue II.L	(DSER) See Abstract	

NRC Review

NRR/PDST

Last Updated: 8/6/92

P.1.V-69

Status: Closed(Cert) **Next Action:** none

implementation of simplification objective (11.4)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.11-2) "Section 11.4 of Chapter 1 of the Passive Requirements Document states that ALWR design should be simpler than that of current operating plants because a minimum number of mechanical components (valves, pumps, heat exchangers, snubbers) and a minimum of instrumentation and controls will be used. EPRI specifies that the plant will be designed to simplify operations during all modes of operation, including operator actions to diagnose and manage abnormal and accident conditions. The staff agrees with the overall objective to simplify systems and operations. The staff will ensure that implementation of this objective is in accordance with the Commission's regulations and guidance during its review of an individual application for final design approval and design certification."	agree	(DSER) See Abstract	

NRC Review

NRR/PDST

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P.1.V-70

Status: Closed(Cert) **Next Action:** none

implementation of standardization objective (11.5)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.11-2) "Section 11.5 of Chapter 1 of the Passive Requirements Document states that the ALWR design will be developed as a standard plant design, including, as a minimum, a standard design basis, standard site envelope, standard equipment, and standard technical documentation. The staff supports the concept of standardization, as can be seen in the promulgation of 10 CFR Part 52, the issues of the Commission's Standardization Policy, and the staff's efforts in the review of the EPRI Requirements Document. The staff will ensure that implementation of this concept is in accordance with the Commission's regulations and guidance during its review of an individual application for final design approval and design certification."	agree	(DSER) See Abstract	

NRC Review

NRR/PDST

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ALWR/NRC OPEN ISSUES

P.1.V-71

Status: Open

Next Action: NRC

inservice testing requirements for the essential non-safety-related components (12.2.1, 12.2.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.12-3) "The staff concludes that all passive safety-related equipment, including non-Code safety-related pumps and valves, must be tested in accordance with ASME Code, Section XI. The staff may not require the essential non-safety-related components to meet all of the safety-grade criteria. However, the staff concludes that there are uncertainties concerning the lack of a proven operational performance history. These uncertainties make the essential non-safety-related systems and components more important in providing the defense-in-depth to prevent and mitigate accidents and core damage. The staff is still evaluating this issue for the passive plant designs. The specific staff positions on the inservice testing requirements for the essential non-safety-related components will be determined when the staff completes its review of the issue of regulatory treatment of non-safety-grade systems. Therefore, the staff will evaluate this issue during its review of individual application for final design approval and design certification."	Chapter 1, section 12 of the URD has been substantially revised. For essential valves and pumps, testing is required in accordance with ASME Code Section XI. This should be sufficient to close this issue.	(DSER) See Abstract	NRC review this response

NRC Review

NRR/EMEB

Last Updated: 8/6/92

ALWR/NRC OPEN ISSUES

P.1.V-72 | **Status:** Closed(cert) | **Next Action:** none

quarterly testing of pumps and valves (12.2.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.12-4) "In its letter dated May 17, 1991, the staff requested that the passive ALWR systems be designed to accommodate the applicable ASME Code, Section XI requirements for quarterly testing of pumps and valves, rather than allow designs that only accommodate testing during cold shutdowns or refueling outages. In its letter dated August 1, 1991, EPRI responded to this request by stating that the frequency of testing should be determined by such factors as component design, application, PRA insights, and design alternatives. EPRI further stated that the ALWR program does not propose to change the manner in which the code has been applied to recently licensed plants. For the reasons discussed in Section 12.2.1 of this report, the staff concludes that its position as stated in RAI 210.39(b) of the May 17, 1991, letter will improve the component reliability for passive ALWRs. Therefore, the staff concludes that EPRI's response is not acceptable. The staff will evaluate this during its review of individual applications for final design approval and design certification in accordance with the above position."	All essential (safety and non-safety but important to investment protection) valves of Passive ALWR systems will accomodate the requirements of ASME Code Section XI. This includes the possibility to defer the testing in particular cases as permitted by the Code. For special systems (eg: IRWST injection for the PWR) special and additional requirements are shown in the relevant Chapters of Volume III.	See DSER	none

NRC Review

NRR/EMEB

Last Updated: 7/17/92

ALWR/NRC OPEN ISSUES

P.1.V-73		Status: Open	Next Action: ALWR

check valve testing methods (12.2.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.12-5) "The staff disagrees with EPRI's position that a commitment to check valve testing methods requires a detailed design analysis. The staff concludes that a commitment to check valve testing methods should be part of the Passive Requirements Document. In addition, a requirement should be added to the list of guidelines from EPRI Report NP-5479, "Application Guidelines for Check Valves in Nuclear Power Plants," in the requirement portion of Section 12.2.6.1 of Chapter 1 to state that, in the selection and the application of valves, the plant designer should also consider parts clearance, disc stability, and wear relative to actual operational flow conditions. Pending such changes the staff will evaluate this during its review of individual applications for final design approval and design certification in accordance with the above positions."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR / EMEB

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-74		Status: Closed(Cert)	Next Action: none

full-flow testing of check valves (12.2.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.12-6) "In its letter dated May 17, 1991, the staff requested EPRI to revise Section 12.2.7.2 in Chapter 1 of the Passive Requirements Document to reflect the staff's position on full-flow testing of check valves as described in the letter. In its August 1, 1991, response, EPRI referred to its position as provided in its responses to RAI 210.39(b) and (f). For reasons similar to those discussed above, EPRI's response is not acceptable. The staff maintains that testing method and testability are important to reliability assurance and that a commitment to the staff's position on full-flow testing of check valves should be part of the Passive Requirements Document. Pending such a commitment, the staff will review individual applications for final design approval and design certification in accordance with the above position."	The URD endorses the Staff position requiring system design that will permit full-flow testing of installed safety-related check valves to demonstrate operability of the valves under operating conditions.	(DSER) See Abstract	

NRC Review

NRR/EMEB

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ALWR/NRC OPEN ISSUES

P.1.V-75

Status: Open

Next Action: ALWR

provisions to test hydraulically and pneumatically operated valves under design-basis differential pressure and flow (12.2.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.12-6) "In its letter dated May 17, 1991, the staff requested EPRI to require the plant designer to commit to test hydraulically and pneumatically operated valves in accordance with the forthcoming ASME/ANSI OM Part 18, "Performance Testing of Hydraulic Operated Valve Assemblies in LWR Plants," and Part 19, "Performance Testing of Pneumatically Operated Valve Assemblies in LWR Plants." In its letter dated August 1, 1991, EPRI responded to this request by stating that it is committed to available and applicable codes and standards. At the time that equipment is designed, the plant designer will identify applicable revisions of each document. This response is not entirely acceptable. The staff position is that designs should incorporate provisions to test hydraulically and pneumatically operated valves under design-basis differential pressure and flow. The design-basis capability of these types of valves will be expected to be verified before installation, before startup and periodically through a program similar to that recommended for motor-operated valves in Generic Letter 89-10, dated June 28, 1989. Pending such a commitment, the staff will evaluate this during its review of individual applications for final design approval and design certification in accordance with the above positions."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/EMEB

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-76

Status: Open

Next Action: ALWR

qualification testing of active and non-active motor-operated valves (MOVs) (12.2.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.12-7) "In its May 17, 1991 letter, the staff requested additional information on qualification testing of active and nonactive MOVs. In its August 1, 1991, response, EPRI stated that nonactive MOVs should be designed for potential mispositioning, but qualification testing is not required. This statement does not completely agree with the current staff guidelines on valve mispositioning. The staff concludes that nonactive MOVs in a safety-related system either should be designed to prevent mispositioning or should required to be subjected to qualification testing to demonstrate capability to recover from mispositioning. Mispositioning may occur through actions taken at any time locally (manual or electrical), at a motor control center or in the control room and includes deliberate changes of valve position for performing surveillance testing. Therefore, the staff will evaluate this during its review of individual applications for final design approval and design certification in accordance with the above position."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/EMEB

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-77

Status: Open

Next Action: ALWR

technical concerns regarding MOVs (12.2.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.12-8) "Recent industry experience and the results of NRC inspections of MOV programs have indicated several areas requiring attention in the EPRI document. Speci-fically, in addition to the technical information to be provided with each valve described in Section 12.2.2.5 of Chapter 1 of the Passive Requirements Document, operator loads as a function of fluid temperature (subcooling) and seismic/dynamic effects, as well as precise internal dimensions of the valve, should be provided. In addition to consideration of stem leakage in establishing the proper globe valve orientation described in Section 12.2.2.6.2 of Chapter 1 of the Passive Requirements Document, any reliance on a globe valve to isolate flow or the use of the valve for throttling flow should also be considered in establishing proper orientation. In addition to ensuring that the valve bonnet and disc will be designed to prevent pressurization due to heatup of fluid trapped in the bonnet described in Section 12.2.2.8.2 of Chapter 1, the bonnet should be designed to prevent its internal pressurization greater than both the upstream and downstream piping, or the motor operator should be designed to overcome such pressurization. EPRI should revise Section 12.3.2.3.3 of Chapter 1 to require that provisions be made for the measurement of both stem thrust and actuator torque because of the importance of information regarding the conversion of torque to thrust (i.e., stem factor). As a clarification of Section 6.2.2.1.4 of Chapter 5 regarding the capability of isolation valves to close against conditions that may exist during events requiring containment isolation, the isolation valves should be designed and test-qualified to be able to isolate flow resulting from a pipe break at the worst-case differential pressure (e.g., a condition resulting from a failure to scram the reactor in a timely manner), because the potential for a break in a line from the reactor vessel would likely be greatest when the reactor pressure was abnormally high. Pending modification of these sections, the staff will evaluate this during its review of individual applications for final design approval and design certification in accordance with the above positions."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/EMEB

Last Updated: 8/18/92

ALWR/NRC OPEN ISSUES

P.1.V-78		Status: Open	Next Action: ALWR

leak rate testing for individual containment isolation valve (12.2.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.12-8) "In its letter dated August 1, 1991, EPRI responded to the staff's request by explaining that the requirement described in Section 6.2.2.2 of Chapter 5 of the Passive Requirements Document is intended to require the plant designer to minimize the number of valves that will be subjected to Type C testing in accordance with Appendix J to 10 CFR Part 50 rather than to set down the type of testing required for CIVs. EPRI's response also referred to Section 12.2.7.1 of Chapter 1 of the Passive Requirements Document, which requires the plant designer to provide for testing of essential valves in accordance with ASME/ANSI OM Part 10. Furthermore, EPRI stated that the designation of specific inservice testing requirements is beyond the scope of the Passive Requirements Document and properly belongs in design certification documentation. The staff disagrees with EPRI's position that the designation of specific inservice testing requirements is beyond the scope of the Passive Requirements Document. For the reasons discussed above, the staff has also determined that EPRI's response will not result in individual CIV leakage rate testing and is, therefore, not acceptable. The staff will evaluate this during its review of individual applications for final design approval and design certification in accordance with the above staff position."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/EMEB
NRR/SPLB

Last Updated: 6/18/92

ALWR/NRC OPEN ISSUES

P.1.V-79		Status: Open	Next Action: NRC

frequency and extent of disassembly and inspection of safety-related pumps (12.2.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1.12-10) "In its letter dated May 17, 1991, the staff requested EPRI to provide a requirement to periodically disassemble and inspect all safety-related pumps. The staff requires, as a minimum, a commitment to develop a program that will establish the frequency and the extent of disassembly and inspection of safety-related pumps, including the basis for the frequency and the extent of each disassembly. In its letter dated August 1, 1991, EPRI responded to this request by referring to its position that was provided in the responses to RAIs 210.39(c) and (d). The staff's evaluation of these responses is discussed above. For similar reasons to those discussed above, the staff will evaluate this issue during its review of individual applications for final design approval and design certification."	The Passive ALWRs will not have safety-related pumps. Essential pumps will be inspected and tested according to ASME Code Section XI. A program to establish the frequency and extent of disassembly is not required for non safety-related pumps.	(DSER) See Abstract	NRC review this response

NRC Review

NRR/EMEB

Last Updated: 8/19/92

ALWR/NRC OPEN ISSUES

P.1A.O-1		Status: Open	Next Action: NRC

reporting of core-damage-frequency results as mean values (1.7)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.1-9) "The staff requires that the EPRI guidance be revised to require reporting of mean value core-damage-frequency results, wherever possible."	The URD will be changed to require "mean value core damage frequency wherever possible"	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/PRAB

Last Updated: 8/6/92

VOLUME III, CHAPTER 1, APPENDIX A: PRA KEY ASSUMPTIONS AND GROUNDRULES

Paragraph No.	Assumption/Groundrule	Rationale	Rev
1.2.3	**Core Damage Frequency Requirement**	**Core Damage Frequency Requirement**	0
	The plant design shall be such that a realistic assessment of the core damage frequency will produce ~~an estimate~~ a mean value no higher than 1×10^{-5} events/reactor year (including both internal and external events).	This requirement minimizes the financial risk to the utility from loss of the large capital investment in the generating station. ~~The point estimate as defined in Section 1.3 has been chosen for this comparison.~~ The mean value is specified for this purpose, as detailed in Section 1.3	3
1.3	**POINT ESTIMATE QUANTIFICATION**	**POINT ESTIMATE QUANTIFICATION**	0
	~~For each primary event input into the PRA model, a point estimate will be derived to represent that event in calculating the frequency of event sequences. The mean value shall be the point estimate used for this purpose. These mean values shall be propagated through the PRA models, and point-estimate frequencies shall be obtained for core damage sequences and radionuclide release categories of interest.~~ Insert (1)	~~PRA results, in the form of realistic best estimates for the frequency of core damage and the frequency of serious radionuclide release, will be used to compare against the ALWR Requirements Document values given in Chapter 1, Section 1.4.1. The use of mean input values for quantification and comparison to the ALWR Top-Level Requirements has been specified for several reasons. First, the use of mean values is reasonable, since in the absence of formal propagation of uncertainty, their use in the quantification process provides the most meaningful point estimate. As a practical matter, for the passive plant this point estimate should be very close to the true mean value, since it is unlikely that combinations of identical independent events whose uncertainty distributions would be correlated will be important to the results. Furthermore, the mean value is typically influenced by the extreme values in the uncertainty distribution. For example, for an event whose probability of failure is represented by a lognormal distribution with an error factor of 3 (a typical distribution for a basic event in a PRA model), the mean value is at about the 65th percentile of the distribution. Thus, the use of point estimates calculated from mean input values for comparison against the ALWR criteria provides added assurance that the design is robust, even accounting for random variability in equipment or human performance, or lack of precise knowledge of failure rates.~~	0

ALWR/NRC OPEN ISSUES

P.1A.O-2 | | **Status:** Open | **Next Action:** NRC

point-estimate quantification (1.8)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.1-9) "As noted in Sections 1.9 and 6 of this DSER appendix, a point-estimate quantification by itself is not adequate and must be supplemented by an uncertainty analysis with uncertainties propagated from basic event uncertainties, including uncertainties on phenomenological issues."	We feel that the most appropriate approach to deal with the problems of uncertainty is to perform an extensive set of sensitivity studies. However, the URD will be modified to require a propogation of uncertainties for dominant sequences in the Level 1 PRA.	(DSER) See Abstract	NRC review pen & ink change

NRC Review

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The mean value is specified for this purpose, as detailed in Section 1.3

VOLUME III, CHAPTER 1, APPENDIX A: PRA KEY ASSUMPTIONS AND GROUNDRULES

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
1.2.3	**Core Damage Frequency Requirement**	**Core Damage Frequency Requirement**	0
	The plant design shall be such that a realistic assessment of the core-damage frequency will produce a mean value ~~an estimate~~ no higher than 1×10^{-5} events/reactor year (including both internal and external events).	This requirement minimizes the financial risk to the utility from loss of the large capital investment in the generating station. ~~The point estimate as defined in Section 1.3 has been chosen for this comparison.~~	3
1.3	**POINT ESTIMATE QUANTIFICATION**	**POINT ESTIMATE QUANTIFICATION**	0
	For each primary event input into the PRA model, a point estimate will be derived to represent that event in calculating the frequency of event sequences. The mean value shall be the point estimate used for this purpose. These mean values shall be propagated through the PRA models, and point-estimate frequencies shall be obtained for core damage sequences and radionuclide release categories of interest. Insert (1)	PRA results, in the form of realistic best estimates for the frequency of core damage and the frequency of serious radionuclide release, will be used to compare against the ALWR Requirements Document values given in Chapter 1, Section 1.4.1. The use of mean input values for quantification and comparison to the ALWR Top-Level Requirements has been specified for several reasons. First, the use of mean values is reasonable, since in the absence of formal propagation of uncertainty, their use in the quantification process provides the most meaningful point estimate. As a practical matter, for the passive plant this point estimate should be very close to the true mean value, since it is unlikely that combinations of identical independent events whose uncertainty distributions would be correlated will be important to the results. Furthermore, the mean value is typically influenced by the extreme values in the uncertainty distribution. For example, for an event whose probability of failure is represented by a lognormal distribution with an error factor of 3 (a typical distribution for a basic event in a PRA model), the mean value is at about the 65th percentile of the distribution. Thus, the use of point estimates calculated from mean input values for comparison against the ALWR criteria provides added assurance that the design is robust, even accounting for random variability in equipment or human performance, or lack of precise knowledge of failure rates.	0

Insert 1 (A/G)

1.3 QUANTIFICATION

Point estimates of the risk measures of interest shall be obtained. To the extent practicable, these point estimates shall be mean values. Where formal propagation of uncertainty is not performed, the point estimates shall be obtained by propagating mean values for primary events.

Insert 1 (rationale)

QUANTIFICATION

To the extent possible, these point estimates should be mean values, or values that are consistent with mean values. Where point estimates are used to characterize risk results, the mean value is the representation that is most commonly used. In cases for which the meaningful propagation of probability distributions to calculate an actual mean value is not possible or not practical, use of mean values for the input parameters provides the most consistent and meaningful results.

VOLUME III, CHAPTER 1, APPENDIX A: PRA KEY ASSUMPTIONS AND GROUNDRULES

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
1.3	POINT ESTIMATE QUANTIFICATION (CONTINUED)	POINT ESTIMATE QUANTIFICATION (CONTINUED)	0
	In the event that the PRA analysts choose to propagate the uncertainty distributions for primary events through the sequence quantification, the resulting mean frequencies shall be used for comparison to the quantitative risk criteria presented in Chapter 1, Section 1.4.1.	Some PRA analysts may choose to perform a more rigorous propagation of the sources of uncertainty that can be easily represented in a quantitative manner. In such cases, the calculated mean values should be compared to the risk criteria, rather than some other parameter of the distribution (such as the median or 95th-percentile value). The qualitative evaluation should still be performed for this case, however, for the reasons outlined in Paragraph 1.4.	0
1.4	UNCERTAINTY TREATMENT	UNCERTAINTY TREATMENT	0
	A careful assessment of the potential impact on risk due to uncertainties shall be made, as outlined below.	A thorough understanding of important sources of uncertainty is essential to a proper perspective on the risk results and insights. Although point-estimate values will be used for comparison to the quantitative objectives, it is important that their context be clearly established. Insights gained from an assessment of uncertainties may appropriately result in additional, or different, risk-based decisions regarding particular design features.	3
~~1.4.1~~ 2	~~A~~ qualitative uncertainty analysis ~~shall be performed as part of the PRA.~~ This analysis shall, as a minimum, involve the identification and description of the potentially important sources of uncertainty, and an assessment of the significance of these uncertainties with respect to the results and conclusions of the PRA.	Many of the most important sources of uncertainty do not readily lend themselves to meaningful quantitative treatment. It is important that the analysts give careful, systematic consideration to the sources of uncertainty that could be important and to the impact each of these sources might have on the results. Section 12.2 of NUREG/CR-2300 (Ref. 2) describes methods for such analysis, and Section 12.3.2 of NSAC-60 (Ref. 4) provides an application of a qualitative uncertainty analysis.	3

insert
2

The quantitative uncertainty assessments shall be supplemented with a . . .

Insert 2 (A/G)

1.4.1

Quantitative assessments of uncertainty shall be performed, to the extent that they are practical and meaningful. The nature of the quantitative uncertainty assessments for each element of the PRA is defined in Section 6.

Insert 2 (rationale)

Quantitative assessments of uncertainties can supplement point estimates of risk by providing important perspectives on the results. The nature of these quantitative assessments should be commensurate with the ability to characterize uncertainties and the meaningfulness of the results obtained, as described in Section 6.

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
1.4	**UNCERTAINTY TREATMENT (CONTINUED)**	**UNCERTAINTY TREATMENT (CONTINUED)**	0
1.4.2	The ~~qualitative~~ uncertainty analysis shall be supplemented by a series of quantitative sensitivity studies that investigate the potential impact of particularly important uncertainties. Further specification of potential sensitivity studies is provided throughout the remaining guidelines presented in this document, and is summarized in Section 6.	Quantitative sensitivity studies provide a further important perspective with respect to the potential effects of variability in assumptions, parameter values, etc. on the risk measures of interest. Investigating important issues that lend themselves to quantitative treatment is a necessary element in risk-based decision-making concerning the designs.	3
1.5	**DOCUMENTATION**	**DOCUMENTATION**	3
	The PRA shall be thoroughly documented, as outlined below.		3
1.5.1	The models, data and assumptions for each portion of the PRA shall be formally documented to a sufficient level of detail such that an independent group could recreate the results with a minimum level of interaction with the original analysts. This documentation shall include at least the following: • The reviews made of industry experience and of the plant design to arrive at a comprehensive set of initiating events. • The system interactions and success criteria that form the bases for the core damage event trees. • The system fault-tree (or equivalent) models, including assumptions regarding design details not yet available, types of failure modes included and excluded, the treatment of dependent failures, reviews made for human interactions, and coordination with the reliability data base.	Documentation of the PRA at this level is important for several reasons, including establishing credibility with the NRC and other reviewers; ensuring that the PRA is suitable for use as a "living model" of the plant, as called for in Section 11.6.3 of Chapter 1; providing the information needed to support the development of the Reliability Assurance Program; and supporting the plant reliability and availability analyses. More extensive guidance on documentation for PRA can be found in NUREG/CR-2300 (Ref. 2), NUREG/CR-2815 (Ref. 3), and in Documentation Design for PRA, EPRI Report NP-3470 (Ref. 43).	3
	• The details of the human reliability analysis, as described later in Section 2.9.		3

VOLUME III, CHAPTER 1, APPENDIX A: PRA KEY ASSUMPTIONS AND GROUNDRULES

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
6	**UNCERTAINTY AND SENSITIVITY ANALYSES**	**UNCERTAINTY AND SENSITIVITY ANALYSES**	0
	insert A ~~Sensitivity studies shall be performed for those issues or parameters that are judged to have relatively large associated uncertainty or that are particularly important to the PRA results. These sensitivity studies shall include important aspects from each of the areas outlined below. The sensitivity studies may be qualitative or quantitative, depending on the nature of the issue being addressed.~~	A comprehensive set of uncertainty and sensitivity studies is needed to provide adequate perspective with respect to uncertainty in the PRA results and the significance of potential contributors to risk. As discussed in Paragraph 1.4, such studies are considered to provide more meaningful input to the plant designer and to decision-makers regarding the areas that are most important with respect to risk. This section describes many of the areas in which uncertainty and sensitivity should be considered. ~~In most cases, this list represents a compilation of the studies suggested in individual sections of this document.~~	0
6.1	**ANALYSIS OF SYSTEMS AND SEQUENCES**	**ANALYSIS OF SYSTEMS AND SEQUENCES**	0
	insert X Areas that may be of particular importance with respect to the estimated frequency of core damage include the following: this paragraph as rev. in $\boxed{4}$	~~These are the primary areas that may be most uncertain or are likely to be important to the frequency of core damage. They also include some areas that are of interest for other reasons, as outlined below.~~	0
6.1.1	Frequencies of rare initiating events that are important contributors to risk, and any initiating events whose frequencies are assessed to be low relative to similar events for other nuclear power plants, or that are unique to the passive plant designs.	Initiating events with very long recurrence intervals inherently have the potential for large uncertainty. For some initiating events, features unique to the passive plant may warrant lower best-estimate frequencies; these should be explored so that the rationale for and effects of the lower frequencies are adequately understood and communicated.	0

INSERT A

Uncertainties in the results of the PRA shall be investigated through appropriate quantitative and qualitative assessments. Propagation of uncertainties shall be performed where it is feasible and provides useful results. (it is only asked for level 1 PRA). Quantitative and qualitative sensitivity studies shall be performed where propagation is not practical, or where the uncertain issues do not readily lend themselves to quantitative treatment.

INSERT X

For the frequency of core damage, probability distributions shall be developed for each of the primary events in the plant models. These distributions shall be propagated to provide a quantitative characterization of both the mean core-damage frequency and uncertainty associated with that frequency.

This propagation shall be supplemented with well-defined sensitivity studies to investigate sources of uncertainty that do not readily lend themselves to quantification.

Areas that may be of particular importance...

Propagation of probability distributions for primary events in the level 1 portion of a PRA is a standard task in most PRAs, and is relatively straightforward to accomplish. In addition to potentially providing further insight into the features of the plant design important to risk, the propagation of uncertainties permits the mean core-damage frequency to be calculated. This allows accounting for correlations among the failure data that could cause the propagation only of mean values to result in a result different from the mean core-damage frequency. Sensitivity studies are a more effective means to investigate "modeling" uncertainties, and to consider alternative views of some reliability data.

ALWR/NRC OPEN ISSUES

P.1A.O-3

Status: Open

Next Action: ALWR

quantitative treatment of uncertainties (1.9)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.1-11) "The Passive Requirements Document should provide guidance and a framework for systematically conducting and interpreting sensitivity analyses and for identifying those issues that require further consideration in the context of a quantitative uncertainty analysis. Essential elements of this framework that should be specified in the Passive Requirements Document include: (1) initial screening of issues for applicability to the passive design, (2) sensitivity analyses to further delineate issues of potential risk significance, and (3) systematic analysis of issue uncertainty as part of a broader assessment of uncertainty in the overall risk measures. It should be noted that the treatment of uncertainties for Level 2 issues need not be as extensive as that of NUREG-1150, but must be such that the staff has reasonable assurance that the PRA reflects the significance of key actions, events, and phenomena for the plant design and the effectiveness of the accident-mitigation systems."	See P.1A.O-2	(DSER) See Abstract	NRC review the response to P.1A.O-2

NRC Review

NRR/PRAB

Last Updated: 7/15/92

ALWR/NRC OPEN ISSUES

P.1A.O-4 | Status: Open | Next Action: NRC

guidance on presenting results of PRA (1.10)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A,1-12) "In addition to the expanded guidance in Section 1.5, the Passive Requirements Document should require reporting of the following: ° the frequency of challenging passive decay heat removal (passive residual heat removal or isolation condenser), and the leading contributions to this frequency (the combinations of failures most likely to cause this event) ° the frequency of challenging the passive inventory makeup systems, and the leading contributions to this frequency ° the frequency of challenging the depressurization function, and the leading contributions to this frequency (for some designs, this will be the same as challenging passive inventory makeup systems; for others, it will not) ° the conditional probability that the depressurization function will fail to reduce the reactor coolant system pressure to the point at which gravity injection can function as designed, and the leading contributions to this event ° the conditional probability that the staging of the depressurization will fail in such a way as to affect the fuel adversely (e.g., excessive blowdown caused by opening too many valves or the wrong valves first), and the resulting occupational exposure to the workforce ° the frequency with which depressurization will actuate spuriously, and the leading contributions to this event ° the method of truncation used in the quantification process In addition, the Passive Requirements Document should provide PRA guidance on providing a thorough assessment of the conditional probability that the sources of gravity injection (core makeup tanks, incontainment refueling water storage tank, GDCS, etc.) will, for some reason, be unable to perform their functions as required (e.g., lack of inventory, improper chemistry, human error, leaks)."	The staff asks to specifically present some results.The URD will be modified to take that into account. For the last additional remark about the conditional probability of failures of the different sources of gravity injection, it has no reason to be written in the parr aph "form of results" and should be put in another place. Moreover some sensitivity studies have already been asked for that purpose in section 6.1.2 and 6.1.3.	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/PRAB

Last Updated: 8/18/92

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
1.5	**DOCUMENTATION (CONTINUED)**	**DOCUMENTATION (CONTINUED)**	3
	• The data base used for initiating event frequencies, component failure rates, common-cause failure rates, and maintenance unavailabilities, including an assessment of the applicability of available data for particularly important types of events or components.		3
	• The sequence quantification process, including how any truncation was applied and how sequence-specific aspects such as recovery events were handled.		3
	• The development of the containment event tree and the rationale for events included and excluded.		3
	• The bases for the selection of best-estimate values for the events in the containment event tree and a discussion of the potential ranges of these values.		3
	• The bases for the source-term characterizations.		3
1.5.2	The results of the PRA shall be compiled and presented in such a manner that they clearly convey the quantitative risk measures, the aspects of plant design and operation that are important contributors to those risk measures as well as those responsible for limiting risk, and the effects of important sources of uncertainty.	Clear explanations of the key results is crucial both to properly characterizing the comparisons of the assessed risk measures to the overall safety criteria for the plant design, as well as to understanding the significance of the results in a qualitative manner. The discussions of results should be augmented by clear tabular and graphical representations. Specific forms of presentation are discussed further in Chapter 13 of the PRA Procedures Guide (Ref. 2).	3
1.5.3	The formal documentation for the PRA shall include a summary of the manner in which the PRA effort was integrated into the design process. Specifically, it shall discuss any significant design changes or decisions made based at least in part on use of the PRA models and data.	Documentation of the use of the PRA in the design process is required by Section 11.6.3 of Chapter 1. It is important to demonstrate that the design process appropriately benefited from the insights available from the PRA.	3

Insert here

Insert

Add to the Rationale of Volume III, Chapter 1A, Section 1.5.2

In order to understand the significance of the results with respect to passive plant concepts, the docum[illegible]tation should address and describe the quantification of specific events such as: 1) the frequency of challenging passive decay heat removal and important contributions to this frequency, 2) the frequency of challenging the depressurization function and the passive invento[illegible] makeup systems, 3) the frequency of spurious depressurization, and 4) the conditional probability that depressurizaton will fail or that the staging of depressurization will fail (e.g., excessive blowdown caused by opening too many valves or the wrong valves).

ALWR/NRC OPEN ISSUES

P.1A.O-5 | **Status:** Open | **Next Action:** NRC

guidance on modeling detail required to represent passive system behavior (2.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.2-2) "EPRI should provide guidance that states that the language in the Passive Requirements Document regarding modeling is the minimum level of sophistication required, and that more careful and detailed modeling of some systems (e.g., consideration of a continuous or more finely discretized spectrum of selected physical process variables) may be necessary to adequately represent system behavior."	The industry asumes that the functional operation of passive systems have to be demonstrated on a deterministic basis (testing programs for example.) The probabilistic tool can not be used to make assumptions regarding the design performance characteristics of passive systems. Probabilistic analysis is based on the probabilities of failure of the different components which lead to physical consequences based on testing or deterministic studies. The PRA will use the testing information and studies generated for designing the systems to the extent possible.	(DSER) See Abstract	NRC review this response

NRC Review

NRR/PRAB

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ALWR/NRC OPEN ISSUES

P.1A.O-6 | | **Status:** Open | **Next Action:** NRC

guidance on modeling interactions between passive and active systems (2.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.2-2) "EPRI should provide guidance on modeling potential systems interactions between passive and active systems, since the philosophy of passive plants appears to be to use non-safety-grade active systems as the first line of defense in the event of a transient."	Industry agrees that the potential systems interactions have to be taken into account. It will be integrated into the requirements.	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/PRAB

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Paragraph No.	Assumption/Groundrule	Rationale	Rev.
2.6	**MODELING OF DEPENDENCIES**	**MODELING OF DEPENDENCIES**	0
	The potential for dependent failures shall be considered in a comprehensive manner and shall be treated quantitatively using the best available methods. The types of dependencies that shall be treated explicitly are outlined in the following paragraphs.	Dependencies have the potential to defeat redundancy in the design, and they deserve careful attention in PRA. This is particularly true for the ALWR since the greater degree of redundancy called for in the design requirements would tend to make dependencies relatively more important. It is particularly important to understand the potential effects of such dependencies on an integrated level for the plant.	0
2.6.1	**Sequence Functional Dependencies**	**Sequence Functional Dependencies**	0
	Sequence functional dependencies shall be incorporated into the sequence event trees or equivalent sequence logic. These functional dependencies indicate the effects of the status of one system or safety function on the success or failure of another, or of the same system in different configurations and/or performing different safety functions. Functional dependencies between systems or functions responsible for core cooling and containment systems shall be modeled explicitly. The interaction between safety and non safety systems shall be taken into account.	This is required for proper modeling of the sequences. Both success and failure of a system can affect the performance of another system and the same system in a different configuration or role. In some cases passive and active systems may have interactions because some non safety systems are used as first line of defence in the event of a transient.	0
2.6.2	**Inter-system Dependencies**	**Inter-system Dependencies**	0
	Inter-system dependencies, including both hard-wired dependencies (e.g., through electric power, cooling water, interlocks, permissives, etc.) and functional dependencies (e.g., ambient cooling, adequate net-positive suction head, etc.) shall be included explicitly in the system fault trees or other models.	Shared support systems or other inter-system dependencies may result in bypassing intended redundancy or diversity in the systems designed to prevent core damage.	0

ALWR/NRC OPEN ISSUES

P.1A.O-7		Status: Open	Next Action: NRC

guidance for developing the success criteria for passive systems (2.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER,p 1A.2.2) "The staff believes that the fundamental differences between active and passive design concepts will necessitate a different approach to defining the success criteria, and that the Passive Requirements Document should provide additional guidance on this aspect of the PRA."	It is not the role of probabilistic studies to define the success criteria, but rather the role of deterministic studies (thermal-hydraulic studies for example.) Specific analysis will have to be done to determine the success criteria of the plant taking into account active systems, passive systems and their possible interactions. It will be a large number of studies, but the approach is not different from the current process.	(DSER) See Abstract	NRC review this response

NRC Review

NRR / SRXB

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ALWR/NRC OPEN ISSUES

P.1A.O-8

Status: Open

Next Action: ALWR

determination of an appropriate mission time (2.10)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.2-9) "the staff concludes that the Passive Requirements Document should be revised to require that (1) the scope of the PRAs performed for passive plant designs be expanded to include treatment of the plant evolutions and system functions (active and passive) necessary to bring the reactor to (a) cold shutdown and (b) a static shutdown condition of 400°F, and to maintain either of these conditions for the long term, and (2) mission times be determined and justified by the plant designer consistent with this expanded scope."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

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ALWR/NRC OPEN ISSUES

P.1A.O-9 | **Status:** Open | **Next Action:** ALWR

requirements to address those important passive design-specific areas of uncertainty (6.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.6-2) "The staff requires EPRI to provide guidance on how a full uncertainty analysis will be performed for the Level 1 portion of the PRA, with uncertainties propagated from basic events, including initiating event frequencies, data, common cause/mode failure, success criteria, and human error."	See P.1A.O-2	(DSER) See Abstract	See P.1A.O-2

NRC Review

NRR/PRAB

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ALWR/NRC OPEN ISSUES

P.1A.O-10

Status: Open

Next Action: NRC

failure rate for the main stepup transformers (8.2)

Abstract	Industry Position	NRC Position	Action Description
(CAN NOT FIND THE REFERENCED SECTION "8.2" HENCE CAN NOT DETERMINE THE ISSUE -JDT 4/29/92)	NRC should clarify this "issue"	(DSER) See Abstract	NRC clarify this issue

NRC Review

NRR/SELB J. Lazevnick

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P.1A.V-1

Status: Closed(Cert) **Next Action:** none

long-term decay heat removal in the PRA (1.6)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.1-7) "In response to staff questions in this area, EPRI indicated that the Passive Requirements Document will be revised to include (1) an addition to Section 2.3.3.7 of Chapter 1 to require that "all operating conditions (including shutdown ones) shall be taken into account" in the PRA analysis and (2) additional requirements in Appendix A to Chapter 1 to encompass shutdown conditions within the PRA. The Appendix A modifications will require a Level 1 analysis and a simplified evaluation of release frequencies and magnitudes for shutdown conditions. EPRI also proposed an additional section in the Passive Requirements Document providing guidance for the conduct of a simplified PRA for conditions other than full power. The staff has not yet completed its review of the simplified method and is evaluating the need for consideration of external events during shutdown.	Agree but no change to the URD is needed.	(DSER) See Abstract	

The modifications proposed by EPRI are noteworthy and go a long way toward addressing staff concerns regarding the functional assessment of plant response and the systematic treatment of accidents initiated in modes other than full-power operation. However, the Passive Requirements Document does not provide sufficient guidance regarding treatment of long-term DHR in the PRA. Based on the above considerations and the evaluations described in Chapter 5, the staff concludes that the scope of the PRA for passive plant designs must include treatment of all plant evolutions and system functions (active and passive) necessary to bring the reactor to cold shutdown and to maintain this condition for the long term. The PRA must model (1) all equipment that the NRC requires to be available in passive plants to prevent or mitigate severe accidents and (2) all equipment that an operator is likely to attempt to use to prevent or mitigate any initiating event that may lead to core damage. This should include consideration of accident-management measures or strategies for establishing and maintaining long-term cooling or containment heat removal. Mission times for associated plant equipment must be justified and consistent with this objective as discussed in Section 2.10 of this appendix. The staff will evaluate these aspects of the PRA during its review of an individual application for final design approval and design certification."

NRC Review

NRR/PRAB
NRR/SRXB

Last Updated: 8/6/92

P.1A.V-2

Status: Open

Next Action: ALWR

justification of mission times and success criteria (2.10)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.2-9) "Although the risk significance of long-term core cooling may not be as great for passive plant designs in absolute terms, the results of the French studies suggest that greater attention should be paid in the PRA to events outside the 24-hour time window that has been traditionally assumed in the PRA. This may be particularly important for passive plant designs in that they use a combination of passive and non-safety active systems to perform long-term residual heat removal, and that the reliability of these systems may be less than for current plant designs. In view of the potential risk contribution from plant operations later than 24 hours, the staff considers the guidance on mission time in the Passive Requirements Document to be inadequate. On the basis of the considerations discussed above, as well as in Section 1 of this DSER Appendix, the staff concludes that the Passive Requirements Document should be revised to require that (1) the scope of the PRAs performed for passive plant designs be expanded to include treatment of the plant evolutions and system functions (active and passive) necessary to bring the reactor to (a) cold shutdown and (b) a static shutdown condition of 400°F, and to maintain either of these conditions for the long term, and (2) mission times be determined and justified by the plant designer consistent with this expanded scope. This will necessitate consideration of mission times considerably longer than 24 hours, as well as explicit treatment of actions by on-site operating staff and offsite support organizations that may need to be accomplished within this timeframe. Determination of an appropriate mission time should be based on consideration of (1) passive and active system performance and reliability late in an event, (2) availability and reliability of available backup systems or components, and (3) actions required to be taken by operating staff and offsite response organizations, and the provisions that would be in place to ensure such actions could be taken in a timely manner. Mission times may be different for going to cold shutdown and static shutdown. The different types of equipment and operator actions needed to go to these modes must be reflected explicitly in the system and function success criteria (see Section 2.3). Mission times and success criteria will have to be justified by the plant designer and will be reviewed by the staff as part of the final design approval process for each passive design PRA."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/PRAB

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P.1A.V-3

Status: Open

Next Action: ALWR

reliability data (2.11)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.2-11) "Reliability data for non-safety-related (but normally safety-related in tradi-tional PWRs and BWRs) components must be justified on the basis of the quality of the equipment purchased, test intervals, capability to perform its intended function in an adverse environment, experimental data, and applicable technical specifications. This is part of the overall concern regarding the regulatory treatment of non-safety-related systems for the passive plant designs. This is an open issue that must be satisfactorily resolved before the staff can complete its review of Appendix A to Chapter 1."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/PRAB

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ALWR/NRC OPEN ISSUES

P.1A.V-4

Status: Closed(Cert) **Next Action:** none

review of core-damage-sequence binning (4.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.4-1) "In Section 4.1 of Appendix A to Chapter 1, EPRI states that core-damage sequences are expected to be binned (grouped). If core-damage bins are used, they must be defined so that all sequences within a particular bin lead to similar effects with respect to containment sequence and source term phenomena. EPRI requires that the definition of bins provide a means to ensure that the delineation of core-damage sequences is discriminated sufficiently to afford the proper level of coordination with the containment analysis. This is intended to provide (1) a means of managing the number of accident sequences and (2) an additional means of gaining information needed for the in-plant analysis. The binning of sequences is an acceptable procedure to limit the number of containment analyses performed. As EPRI states, it is necessary that all sequences within a bin lead to similar effects with respect to containment and source term phenomena. The staff will review core-damage-sequence binning when the design-specific PRA is submitted."	Agree	(DSER) See Abstract	

NRC Review

NRR/PRAB

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ALWR/NRC OPEN ISSUES

P.1A.V-5		Status: Closed(Cert)	Next Action: none

review of actual groupings of the accident sequences into plant damage states (4.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.4-1) "To reduce the number of deterministic analysis runs necessary to develop the containment event tree branch point probabilities, EPRI made a simplifying assumption. The frequency dominant accident sequence for each plant damage state will be used to define in-plant phenomenological analysis parameters for use in determining containment performance source terms. EPRI cautions that the plant damage states must be sufficiently and uniquely defined to ensure that they adequately reflect the characteristics important to the containment response and release magnitudes in order to avoid introducing uncertainties that could otherwise be avoided. This procedure is a generally accepted part of PRA methodology. The staff will review the actual groupings of the accident sequences into plant damage states when the design-specific PRA is submitted."	Agree	(DSER) See Abstract	

NRC Review

NRR/PRAB

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ALWR/NRC OPEN ISSUES

P.1A.V-6

Status: Open

Next Action: ALWR

review of the evaluation of containment leakage paths (4.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.4-3) "Section 4.4 of Appendix A to Chapter 1 requires that containment bypass sequences be assessed and include all connections to the reactor coolant system. EPRI states that containment bypass sequences can result in significant releases from the containment and have the potential to be important risk contributors. Examples of bypass sequences that have been identified as important in past PRAs include °steam generator tube rupture (PWR only) °residual heat removal isolation failure °high-pressure coolant injection (BWR only) °core spray (BWR only) °feedwater and main steam (BWR only) In the NUREG-1150 studies for Surry and Sequoyah, bypass sequences dominate early fatality risk. The staff agrees that containment bypass sequences are important and, therefore, agrees with the EPRI requirement to assess such sequences. The staff notes, however, that the requirement to assess all connections to the reactor coolant system may be unrealistic; otherwise, even instrument lines would need to be considered. The staff suggests that EPRI modify this section to incorporate some screening criteria below which connected piping would not need to be addressed in detail."		(DSER) See Abstract	(ALWR) Respond to DSER

NRC Review

NRR/SPLB
NRR/PRAB

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ALWR/NRC OPEN ISSUES

P.1A.V-7		**Status:** Closed(Cert)	**Next Action:** none

computer codes for in-plant sequence assessment (4.4)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.4-3) "ection 4.5.2 of Appendix A to Chapter 1 states that the MAAP code will be the primary tool used to assess the thermal-hydraulic and other physical processes involved in accident progression. While recognizing the value of MAAP as an integrated code for severe-accident analysis, the staff concludes that this requirement places an undue reliance on MAAP and fails to establish a requirement for dealing with deficiencies of the this code. More specifically, the staff is concerned that for best-estimate calculations, the MAAP code (or other codes for that matter) will not adequately treat certain physically important severe-accident phenomena, and that the models and assumptions made in the MAAP code will not be generally accepted by the experts in the field. A particular concern for passive plants is that the adequacy of the code for predicting natural circulation flows has not been demonstrated by EPRI and assessed by the staff.	Agree	(DSER) See Abstract	

In its response to a staff request for additional information, EPRI proposed a modification to the Passive Requirements Document to further c‥‥fy, by example, the situations in which use of codes other than MAAP might be more appropriate. These examples were "to develop realistic success criteria for core cooling, or to investigate specific phenomena that are not addressed by MAAP or that may be subject to large uncertainties." This modification adequately addresses the staff's concern regarding the adequacy of the MAAP code for determining success criteria. The staff intends to review applications for final design approval and design certification against the codes most recently endorsed and approved by the NRC at that time, and will evaluate the acceptability of the plant designer's analyses on a case-by-case basis."

NRC Review

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ALWR/NRC OPEN ISSUES

P.1A.V-8

Status: Open

Next Action: NRC

verify that the reference site parameters identified in Annex B are consistent with the revised 10 CFR Part 100 (5.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.5-2) "As part of design certification for each passive ALWR, the staff will require ALWR vendors to provide an assessment of additional risk measures (such as person-rem, and early and latent fatalities) to support the vendor's assessment of Severe Accident Mitigation Design Alternatives (SAMDAs) for the ALWR design. Meteorological data alone is insufficient to calculate these additional risk measures, and will need to be supplemented with bounding population data, such as that provided in Regulatory Guide 4.7."	• Annex B has been totally modified to be consistent with 10CFR100 • The paragraph of the DSER seems to ask to put data on bounding population in the URD. We need this request clarified.	(DSER) See Abstract	NRC review the modified Annex B(Rev 3) NRC clarify second concern

NRC Review

NRR/PRPB J. Lee

Last Updated: 8/19/92

ALWR/NRC OPEN ISSUES

P.1A.V-9

Status: Closed(Cert) **Next Action:** none

differences in computer codes used for calculating offsite consequences (5.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.5-2) "ection 5.2.2 of Appendix A to Chapter 1 states that either MACCS or CRAC2, or another suitable code, will be used for calculating offsite consequen es. Although the CRAC2 code provides an acceptable characterization of the consequences of severe accidents, the MACCS code represents an improvement over CRAC2 and is preferred by the staff for calculating cancer risk. The present version of MACCS (Version 1.5) uses the results of the BEIR III study in the calculation of health effects. However, the BEIR III study has been superseded by the BEIR V study. The BEIR V results indicate a higher cancer risk from low levels of ionizing radiation than did the earlier BEIR III report. The results of the BEIR V study should be taken into account in the calculation of health effects. The staff is preparing an addendum to NUREG/CR 4214, Revision 1, Part II, to address the modification of models resulting from recent reports on the health effects of ionizing radiation. The risk coefficients for fatal cancers would be approximately doubled or tripled by the model modifications (memorandum dated April 11, 1990, from J. M. Taylor, NRC, to Chairman K. M. Carr, "Evaluation of Recent Reports on Health Effects of Low-Level Ionizing Radiation"). Until these modifications are incorporated into MACCS, the staff concludes that the use of CRAC2 is acceptable, but that the effect of model differences must be taken into account in interpreting risk results. The staff will address this issue during its review of an individual application for final design approval and design certification.	Agree	(DSER) See Abstract	

NRC Review

NRR/PRPB J. Lee

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ALWR/NRC OPEN ISSUES

P.1A.V-10 | | Status: Open | Next Action: NRC

source terms for rep[illegible]tive accident sequences are bounded by the physically-based source term (6.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 1A.6-5) "In a February 7, 1991 response to an NRC question concerning plant certification issues contained in SECY-91-151, EPRI stated that the plant designers must confirm as part of the PRA that the source terms for representative accident sequences for their actual standard plant designs are bounded by the physically-based source term in the Requirements Document. However, no mention of the need for PRA analysts to perform this source term assessment is made in Appendix A to Chapter 1.	Agree. However, for Volume III, the changes to address this issue are being made to Chapter 5, Section 2.4.1.2 instead of to Chapter 1A, Section 4.9.	(DSER) See Abstract	NRC review proposed URD revision (Rev 4)

In response to staff concerns regarding the lack of guidance on this matter in Appendix A, EPRI indicated that "as part of the on-going source term development, additional guidance will be provided to ensure that this check on the physically-based source term relative to those derived in the PRA is made."

By letter dated March 19, 1992, EPRI subsequently provided a new section 4.9 to Chapter 1, Appendix A which re[illegible]uires that designers confirm that the PRA source terms for representative accident sequences for their design are bounded by the physically-based source term used in licensing calculations. The staff considers this acceptable, and will evaluate this analysis as part of the design certification review for each ALWR PRA."

NRC Review

NRR/PRPB J. Lee

Last Updated: 8/19/92

Paragraph No.	Requirement	Rationale	Rev
2.4	**MITIGATION**	**MITIGATION**	0
2.4.1	**Source Term Definition**	**Source Term Definition**	0
2.4.1.1	A physically-based source term shall be used as the passive plant accident mitigation feature design basis as defined for each standard plant design in Appendix B.	A physically-based source term is being required as the design basis for the Passive ALWR in order to factor in the source term experience gained in nearly thirty years since TID 14844 was issued. The physically-based source term also provides coupling of the source term and containment thermal-hydraulics, thereby assuring a more consistent, rational basis for containment design features and mitigation systems as well as a [redacted] Since each standard plant design will be different, the associated physically based source term will be specific to that design.	0
		Appendix B defines the physically-based source term for a passive PWR and a passive BWR meeting the ALWR requirements. The report, *"Estimate of Physically-Based Source Term for Passive Advanced Light Water Reactors,"* provides the basis for the various aspects of the Appendix B source terms.	0
2.4.1.2	~~The Plant Designer shall confirm that, for the core damage event used as the basis for estimating the physically-based source term, the source term for the standardized plant design is bounded by the source term in Appendix B.~~	~~This is necessary to assure that the actual standardized plant design systems and features are consistent with the specified source term.~~	0

See attachment

Replace 2.4.1.2. by the following:

Requirement:

The Plant Designer shall compare the fission product release to the environment for functional sequences types (see Chapter 5 - Section 6.6.2.3) greater than approximately 10^{-7} per year with the fission product release to the environment from the updated design basis source term. Cases ~~of~~ for which the PRA release exceeds the design basis source term release ~~should~~ shall be reported and explained.

Rationale:

The PRA provides a significant amount of information on the integrated plant severe accident performance, and a comparison to assure that there are no probabilistically important sequences with fission product release greater than the design basis provide additional confidence in the updated source term.

ALWR/NRC OPEN ISSUES

P.12.O-1 | Status: Open | Next Action: NRC

source term basis for designing radioactive waste systems and evaluation of offsite effluent radioactive nuclide concentration (2.2.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 12.2-9) "the staff concludes that EPRI's source terms for the design of the radioactive waste processing systems given in Passive Requirements Document Table 12.1.1 are inconsistent with SRP Sections 11.2 and 11.3. The staff considers that the 0.25-percent failed fuel percentage for a PWR, specified in the Passive Requirements Document, and the 100,000 μCi/sec noble gas release rate (30 minutes' decay) for the offgas system for a BWR specified in the Passive Requirements Document are inadequate bases for designing radioactive waste processing systems and evaluating offsite radioactive nuclide concentration in effluents in accordance with the limits specified in 10 CFR Part 20."	The industry disagrees, however, it has been determined that the system designs are not likely to be affected, and thus the URD will be modified to accomodate the NRC's position. We will modify the URD to require use of SRP 11.2 and 11.3 unmodified for regulatory conformance evaluation.	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/SPLB Chandra

Last Updated: 8/6/92

Rev 5/92

VOLUME III, CHAPTER 12: RADIOACTIVE WASTE PROCESSING SYSTEMS

Paragraph No.	*Requirement*	*Rev.*

1.5 POLICY STATEMENTS 0

ALWR program policy statements regarding radioactive waste processing systems (RWPS) follow. They are intended to assist the reader in gaining an understanding of the requirements given in subsequent sections and the approach that was taken in their formulation. 0

1.5.1 Good Neighbor Policy 0

The Good Neighbor policy as stated in Volume I is that the plant be a good neighbor to its surrounding environment and population, and requirements to limit radioactive releases from normal operation shall be defined. The radioactive release limitations will apply to solid waste shipment quantities and radioactive liquid and radioactive gaseous release quantities to the environment. 0

Volume I top tier design requirements which apply to radwaste are the following Plant Characteristics: 0

- The [licensing] design basis for radioactive waste processing shall use a ~~failed~~ fuel [defect] rate consistent with regulatory requirements. For purposes of normal [, expected] operation performance evaluation, the Designer shall utilize ~~either ANSI/ANS 18.1 or~~ .025% ~~failed~~ fuel [defects] for PWRs, and a noble gas release rate of 15,000 uCi/sec at 30 minutes for BWRs ~~shall be utilized~~. ~~(Also see Table 12.1-1.)~~ 2

  , i.e., NUREG 0800, Stand. Review Plans 11.1 and 11.2.

- The ALWR shall be designed and constructed so that the amount of radioactive gaseous, liquid and solid waste released from the plant shall be equal to or better than comparable values for the 10% best plants of the same type (i.e., BWR or PWR) currently operating in the U.S. 0

In order to specify the design requirements to limit radioactive releases of liquids and gases, operating plant data were obtained from the then most recently available (1984 and 1985) annual reports of the radioactive materials released from nuclear power plants in the U.S. (NUREG/CR-2907). The 10% best PWR and BWR plants were determined for both gaseous and liquid releases. This data base does not represent the plants operating in the U.S. at the time of these requirements, however, the data chosen present an adequate goal. 0

This data source gives neither fuel leakage information nor the relative amounts of radioactivity released from gaseous radwaste and from ventilation air sources. It is thus not possible to define the specific basis for the performance of these plants. Nevertheless, on a comparative basis, the 10% best plants were established and are to be used as a reference condition for use in evaluating ALWR designs. 0

Table 12.1-1

FUEL SOURCE TERM FOR RADWASTE SYSTEM DESIGN

FUEL SOURCE TERM	LICENSING EVALUATION RADWASTE PROCESSING 10CFR50 App. I	10CFR20	SYSTEM OPERATIONAL PERFORMANCE EVALUATION
PWR			
% Fuel Defects		0.25	0.025**
ANSI/ANS 18.1*	yes		yes**
BWR			
μCi/sec. @ 30 min. decay		100,000	15,000**
ANSI/ANS 18.1*	yes	Use the ANSI mix scaled to 100,000 μCi/sec. @ 30 min. decay	yes**

* ANSI/ANS 18.1 is periodically updated to reflect cumulative experience and may be revised prior to the finalization of any design. The version of ANSI/ANS 18.1 used for design should be checked for consistency with NUREG-0016 and NUREG-0017 and, if there are any substantive differences, made consistent with NRC evaluations.

** The Designer may use either the source terms in ANSI/ANS 18.1 or the value indicated in this table.

Delete this table

Paragraph No.	*Requirement*	*Rev.*
1.5.1	**Good Neighbor Policy (Continued)**	0
	It should be realized that the amounts of radioactivity released to the environment via the liquid and gaseous pathways are a measure of the degree of fission product input from fuel defects and of radioactivity input into ventilation air via valve and other radioactive equipment seal leakage. The radioactivity removal mechanisms of the liquid and gaseous radioactive waste processing systems serve to limit and control release of radioactivity. To properly design the systems to meet specific goals beyond the regulatory requirements it is necessary to specify the extent of fuel leakage and input of radioactivity into ventilation air. To this end the values of fuel ~~leakage~~ of ~~Table 12.1-1~~ ~~Paragraph~~ for normal, expected operation are to be used. ~~(See Section 1.5.2.)~~	0
	It is expected that design to releases based upon these values will compare favorably with those of the 10% best operating plants. Reasons for this are: The ALWR requirements provide a number of features to eliminate or reduce input of radioactive substances into the liquid and gaseous radwaste systems or into the environment. These include valve designs to minimize leakage into ventilation air and into liquids, a goal of zero fuel leakage and minimal fuel manufacturing defects, materials improvements to limit Cobalt 60 production, and the Requirements of this chapter for the radwaste systems, etc. (See Appendix B.)	0
	For the solid wet and dry radioactive wastes volumes were derived from a study by Sargent and Lundy using EPRI reports NP3370 and NP5526 as a foundation.	0
	The norms of performance based upon the best operating plants are:	0

defects stated above

X

GASEOUS EFFLUENTS — Rev. 0

Policy Basis	BWR	PWR	Rev.
Total radioactivity, excluding tritium, will be equal to or lower than the 1984-85 10% best plants of the same type in the U.S.	2000 Ci./yr.	200 Ci./yr.	0

LIQUID EFFLUENTS — Rev. 0

Same as above	0.05 Ci./yr.	0.05 Ci./yr.	0

VOLUME III, CHAPTER 12:
RADIOACTIVE WASTE PROCESSING SYSTEMS

Paragraph No.	*Requirement*	*Rev.*
1.5.1	**Good Neighbor Policy (Continued)**	0
	SOLID WASTES	0
	Low level dry and wet waste will be equal to or lower than the 10% best plants of the same type in the U.S. — 3500 Cu.Ft./yr. — 1750 Cu.Ft./yr.	0
	Dry waste volumes are based on the as-compacted form for the compactible fraction of dry wastes and on the as-shipped form for non-compactible dry wastes. Wet waste volumes are based on the dewatered volume not the as-shipped volume.	0
	The ability to achieve the solid waste goals not only requires an appropriate design, but is dependent on (1) how the plant is operated and (2) national and local regulations established by agencies regulating form and concentration of wastes to be shipped and disposed. Requirements for shipping and disposal have been established by such agencies independently of how a plant is designed and operated.	0
	The Volume I good neighbor policy also includes requirements to limit non-radioactive, hazardous, and toxic chemical releases. As stated in 1.2 above, implementation of this aspect of the good neighbor policy is not in the scope of Chapter 12.	0
1.5.2	**Fuel Source Term Parameters for RWPS Design and Evaluation**	0
	The fuel defect source terms for RWPS design and evaluation have been selected to provide the bases for (1) evaluation of annual average off-site dose in accordance with 10CFR50, Appendix I, (2) evaluation of 24-hour off-site radionuclide concentrations in effluents in accordance with the limits of 10CFR20, and (3) evaluation of the normal, expected RWPS performance for comparison with good neighbor policy goals.	2
	Table 12.1-1 shows the values to be used for the above purposes. For the PWR, the values are given in percent failed fuel. For the BWR, the release rates are measured in the GRWPS in terms of μCi/sec at 30-minute decay.	2
1.5.3	**Base Line Design and Options**	0
	A top tier design requirement (Volume I, page v) is that the ALWR "must be acceptable for most available sites in the U.S." From the viewpoint of the LRWPS, this means that the LRWPS is not designed for "zero liquid release."	0

Paragraph No.	Requirement	Rationale	Rev.
2.2	**KEY PERFORMANCE REQUIREMENTS**	**KEY PERFORMANCE REQUIREMENTS**	0
2.2.1	**Goals of Radioactive Releases and Waste Reduction**	**Goals of Radioactive Releases and Waste Reduction**	0
2.2.1.1	The radioactive waste processing systems shall (in concert with other features specified in the Requirements Document) enable the ALWR to meet the goals of reducing radioactive releases and of reducing solid low level waste volume from the ALWR plant.	This is an ALWR program general requirement and is required to meet the "good neighbor" policy of Section 1.5.1.	0
2.2.1.2	The Plant Designer shall demonstrate for the given design that the expected inputs to the radwaste processing systems and process methods result in outputs that meet Chapter 12 and overall ALWR objectives and policies.	This will show that the ALWR objectives and requirements relative to reducing radwaste inputs and outputs are met and that effective processing methods are employed. The Plant Designer should consider the information provided in ANS/ANSI Standards 55.1, 55.4, 55.6, and 40.35 as a design base, as modified by the changes due to the ALWR.	0
2.2.2	**Source and Input Terms**	**Source and Input Terms**	0
X 2.2.2.1	Licensing d Design basis fission product radioactivity concentrations in reactor coolant and associated release rates shall be based upon the fuel leakages given in ~~Table 12.1-1. The several values given in Table 12.1-1~~ Paragraph 1.5.1 and shall be used for the purposes stated in the policy statement of 1.5.2. The Plant Designer shall make the evaluation necessary to show that the licensing requirements and evaluations against "good neighbor" policy goals are met. Activation product source terms in reactor coolant shall be consistent with those given in NUREG-0016 for the BWR and in NUREG-0017 for the PWR ~~and~~/in ANSI/ANS 18.1 for both. (for ~~Standard~~ for	The fission product and activation product bases for design are provided, along with requirements for evaluation of the design.	2

ALWR/NRC OPEN ISSUES

P.12.O-2		Status: Open	Next Action: NRC

basis for 2-minute delay requirement for BWR turbine gland seal system exhaust (3.3.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 12.3-5) "since EPRI requires only the use of essentially non-radioactive steam for the BWR turbine gland seal system, it is not clear why EPRI has identified a 2-minute delay line as a requirement for the offgases from the BWR turbine gland seal system exhaust under certain circumstances."	Two minute delay for gland exhaust effluent was revised in Revision 2 in Figure 12.3-1 and Section 3.2.4.1.2. Two minute delay is no longer required, rather designer analysis.	(DSER) See Abstract	NRC review this response

NRC Review

NRR/SPLB Chandra

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ALWR/NRC OPEN ISSUES

P.12.O-3

Status: Open

Next Action: NRC

production sources for "essentially nonradioactive steam" (3.3.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 12.3-5) "EPRI has not explained why essentially nonradioactive steam is preferred over totally clean steam for the BWR turbine gland seals and how such steam is to be produced."	Totally clean steam will unnecessarily generate additional liquid radwaste because of the need to evaporate demineralized water at all times. Steam that has had sufficient time to decay N-16 is sufficient for this application. The normal source of such steam can be from anywhere in the turbine/ feedwater heating cycle where it can be demonstrated that N-16 is not a problem.	(DSER) See Abstract	NRC review this response

NRC Review

NRR/SPLB Chandra

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ALWR/NRC OPEN ISSUES

P.12.O-4

Status: Open

Next Action: NRC

discrepancy between Figure 12.3-1 and requirement in Chapter 13 (3.3.1)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 12.3-5) "Figure 12.3-1 does not reflact the requirement in Chapter 13"	Agree. Chapter 13, Section 3.2.4.4 was changed in Rev. 3 to only require monitoring.	(DSER) See Abstract	NRC review Rev. 3 of Chapter 13, Section 3.2.4.4.

NRC Review

NRR/SPLB Chandra

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ALWR/NRC OPEN ISSUES

P.12.O-5

Status: Open

Next Action: NRC

use of post-filter downstream of charcoal adsorber in ventilation exhaust systems (3.3.3)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 12.3-7) "the staff concludes that it will be inappropriate to assume the regulatory guide efficiencies for the removal of elemental and organic iodine from the effluent stream if there is no HEPA filter downstream of the charcoal adsorber for collecting carbon fines."	The ALWR position is that designers should meet regulatory dose criteria without charcoal. If this can be done, charcoal need not be installed and the question of downstream filters is moot. On the other hand, if charcoal is required, the ALWR program believes the most recent (1989) standards (N509/N510) should be used. Those design standards call for "bag filters" downstream of charcoal, not HEPAs. This change is made possible by the advances in charcoal technology and quality. The NRC should continue the dialog with the ALWR and not remain with old and outdated requirements.	(DSER) See Abstract	NRC review need for charcoal and, if needed, newer industry standards.

NRC Review

NRR/SPLB Chandra

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ALWR/NRC OPEN ISSUES

P.12.O-6		Status: Open	Next Action: NRC

guidance regarding direct piping from radioactive plant systems to sumps or waste collection tanks (BWR) (4.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 12.4-4) "EPRI [illegible] not provided adequate guidance or requirements regarding direct piping from radioactive plant systems to sumps or radioactive waste collection tanks to eliminate potential sources of airborne radioactivity for BWRs."	Agree - As stated, it is misleading in that it implies applicability to PWRs only, which was not the intent. We will modify requirement 4.2.2.2 to make it clear that direct piping of drains applies to BWRs as well as PWRs..	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/SPLB Chandra

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Paragraph No.	*Requirement*	*Rationale*	*Rev.*
4.2.2.2	**Segregation Within Subsystems**	**Segregation Within Subsystems**	0
4.2.2.2.1	In the PWR, reactor coolant which normally contains hydrogen and may contain fission gases shall be collected in covered drain tanks or routed directly via CVCS to the borated waste processing subsystem.	Hydrogen containing wastes must be kept from entry of oxygen by exclusion or by nitrogen blanketing. A covered tank is one wherein air is excluded by use of diaphragms or inert gas blanketing in a non-vented tank.	0
4.2.2.2.2	Liquid radioactive wastes from non-hydrogen bearing (PWR) radioactive or potentially radioactive plant systems shall be directly piped to sumps or tanks in the various buildings, or to radwaste collection tanks.	Wastes are directly piped; open funnels or routing via trenches is not permitted. The wastes are thereby contained and are not a source of airborne radioactivity.	0
BWR radioactive or potentially radioactive plant systems and from	• At least one floor drain sump in LRWPS service in each building shall be a building low point.	• Floor drain sumps collect building drainage. Equipment drain sumps (or tanks) need only be below equipment drain elevations.	0
	• Wastes routed to sumps shall flow by gravity.	• In most cases, there is no driving force for flow other than gravity.	0
	• The leakage from pump shaft seals and water from pump casing drains shall be collected and routed to the appropriate LRWPS subsystem, as defined in Section 4.2.2.1.	• Pump drainage needs to be routed to the LRWPS subsystem provided for the type of water which is present.	0
	• Pump baseplate drains shall be routed to the floor drain subsystem because these may be contaminated with oil. However, every effort shall be made to minimize oil leakage and to keep it out of LRWPS. See Section 4.2.2.3.3.	• Baseplate drains can be different from pump seal water. The use of mechanical seals also minimizes the leakage from seals.	0

ALWR/NRC OPEN ISSUES

P.12.O-7	

Status: Open

Next Action: NRC

requirements for LRWPS filter housing and components (4.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 12.4-4) "EPRI has not provided adequate guidance or requirements for filter housing and components. This is an outstanding issue that must be resolved before the staff can complete its review of Chapter 12."	This is addressed directly in Chapter 1, Section 12.9 generally and Sections 12.9.1.1 and 12.9.2 specifically; and indirectly in Chapter 12, Section 2.2.2.2 by reference to ANS/ANSI 55.6. To clarify this, we will modify Chapter 12 to refer to Chapter 1, Section 12.9	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/SPLB Chandra

Last Updated: 8/6/92

Paragraph No.	Requirement	Rationale	Rev.
4.5.8	**Covered Drain Tanks (PWR)**	**Covered Drain Tanks (PWR)**	0
	The reactor coolant drain tank and the auxiliary building drain tank and/or its equivalent in the LRWPS shall have adequate provisions to prevent release of radioactive gases and for oxygen exclusion to prevent hydrogen explosions. Typical industry methods include the use of a diaphragm cover and/or an inert cover gas.	Fluid contained in these tanks will not have been degassified so personnel radiation exposure due to releases is an important concern for meeting ALARA requirements. Hydrogen must be controlled to preclude the possibility of explosion.	0
X 4.5.9	FILTERS and **Ion Exchangers**	FILTERS and **Ion Exchangers** X	0
	4.5.9.1 ION EXCHANGER REQUIREMENTS - LRWPS		
4.5.9.1.1	**Resin Type**	**Resin Type**	0
	The resins used shall be strong acid styrene-divinylbenzene cation and strong base quarternary amine resins except where special ion selective resins are used.	These resin types show the best removal characteristics for the variety of ions requiring removal.	0
4.5.9.~~2~~1.2	**Resin Form**	**Resin Form**	0
	The resin form, H+ for cation resin and OH- for anion resin, shall be selected, except in Boron service.	H+ and OH- resins provide the best removal capacity.	0
4.5.9.~~3~~1.3	**Special Resins**	**Special Resins**	0
	When treated wastes are to be discharged to the environment rather than recycled for reuse, the addition of a bed of sodium-aluminum silicate type ion exchanger, e.g., large-port Mordenite, shall be considered and qualified for use upstream of the mixed bed.	This zeolite material has shown a large capacity for radiocesium and thus shows potential for more economical processing and reduced solid radioactive waste; however, its use should first be qualified by testing. (See EPRI NP-5099.)	0

X Chapter 1 Section 12.9 provides equipment requirements for filters and ion exchangers for all water treating systems including LRWPS.

ALWR/NRC OPEN ISSUES

P.12.O-8

Status: Open

Next Action: NRC

requirements for LRWPS filters (4.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 12.4-4) "EPRI has not provided adequate guidance or requirements for various types of filters in the LRWPS including the capability to disassemble, reassemble, and replace internal components."	This is addressed directly in Chapter 1, Section 12.9 generally and Sections 12.9.1.2 and 12.9.2.7 specifically; and indirectly in Chapter 12, Section 2.2.2.2 by reference to ANS/ANSI 55.6. To clarify this we will modify Chapter 12 to refer to Chapter 1, Section 12.9	(DSER) See Abstract	NRC review pen & ink change

NRC Review

NRR/SPLB Chandra

Last Updated: 8/6/92

Paragraph No.	Requirement	Rationale	Rev.
4.5.8	**Covered Drain Tanks (PWR)**	**Covered Drain Tanks (PWR)**	0
	The reactor coolant drain tank and the auxiliary building drain tank and/or its equivalent in the LRWPS shall have adequate provisions to prevent release of radioactive gases and for oxygen exclusion to prevent hydrogen explosions. Typical industry methods include the use of a diaphragm cover and/or an inert cover gas.	Fluid contained in these tanks will not have been degassified so personnel radiation exposure due to releases is an important concern for meeting ALARA requirements. Hydrogen must be controlled to preclude the possibility of explosion.	0
X FILTERS and 4.5.9	**Ion Exchangers** 4.5.9.1 ION EXCHANGER REQUIREMENTS - LRWPS	FILTERS and **Ion Exchangers** X	0
4.5.9.1, 1	**Resin Type**	**Resin Type**	0
	The resins used shall be strong acid styrene-divinylbenzene cation and strong base quarternary amine resins except where special ion selective resins are used.	These resin types show the best removal characteristics for the variety of ions requiring removal.	0
1.2 4.5.9.2	**Resin Form**	**Resin Form**	0
	The resin form, H+ for cation resin and OH- for anion resin, shall be selected, except in Boron service.	H+ and OH- resins provide the best removal capacity.	0
1.3 4.5.9.3	**Special Resins**	**Special Resins**	0
	When treated wastes are to be discharged to the environment rather than recycled for reuse, the addition of a bed of sodium-aluminum silicate type ion exchanger, e.g., large-port Mordenite, shall be considered and qualified for use upstream of the mixed bed.	This zeolite material has shown a large capacity for radio-cesium and thus shows potential for more economical processing and reduced solid radioactive waste; however, its use should first be qualified by testing. (See EPRI NP-5099.)	0

X Chapter 1 Section 12.9 provides equipment requirements for filters and ion exchangers for all water treating systems including LRWPS.

ALWR/NRC OPEN ISSUES

P.12.O-9

Status: Open

Next Action: NRC

requirements for LRWPS ion exchangers. (4.2)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 12.4-4) "EPRI has not specified design considerations and operational requirements for ion exchangers (e.g., addition and removal; retention; strainers; underdrains; and disassembly, assembly, and replacement of internal components)."	This is addressed directly in Chapter 1, Section 12.9 and 12.9.3 generally and Sections 12.9.1.1, 12.9.1.2, 12.9.3.13 through 12.9.3.14 specifically; and indirectly in Chapter 12, Section 2.2.2.2 by reference to ANS/ANSI 55.6. To clarify this, we will modify Chapter 12 to refer to Chapter 1, Section 12.9	(DSER) See Abstract	NRC review pen & ink change

NRC Review

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Paragraph No.	Requirement	Rationale	Rev.
4.5.8	**Covered Drain Tanks (PWR)**	**Covered Drain Tanks (PWR)**	0
	The reactor coolant drain tank and the auxiliary building drain tank and/or its equivalent in the LRWPS shall have adequate provisions to prevent release of radioactive gases and for oxygen exclusion to prevent hydrogen explosions. Typical industry methods include the use of a diaphragm cover and/or an inert cover gas.	Fluid contained in these tanks will not have been degassified so personnel radiation exposure due to releases is an important concern for meeting ALARA requirements. Hydrogen must be controlled to preclude the possibility of explosion.	0
X	4.5.9.1 ION EXCHANGER REQUIREMENTS - LRWPS	X	
4.5.9	FILTERS and **Ion Exchangers**	FILTERS and **Ion Exchangers**	0
4.5.9.1, 1	**Resin Type**	**Resin Type**	0
	The resins used shall be strong acid styrene-divinylbenzene cation and strong base quarternary amine resins except where special ion selective resins are used.	These resin types show the best removal characteristics for the variety of ions requiring removal.	0
4.5.9.~~2~~ 1.2	**Resin Form**	**Resin Form**	0
	The resin form, H+ for cation resin and OH- for anion resin, shall be selected, except in Boron service.	H+ and OH- resins provide the best removal capacity.	0
4.5.9.~~3~~ 1.3	**Special Resins**	**Special Resins**	0
	When treated wastes are to be discharged to the environment rather than recycled for reuse, the addition of a bed of sodium-aluminum silicate type ion exchanger, e.g., large-port Mordenite, shall be considered and qualified for use upstream of the mixed bed.	This zeolite material has shown a large capacity for radiocesium and thus shows potential for more economical processing and reduced solid radioactive waste; however, its use should first be qualified by testing. (See EPRI NP-5099.)	0

X Chapter 1 Section 12.9 provides equipment requirements for filters and ion exchangers for all water treating systems including LRWPS.

P.12.V-1

Status: Closed(Cert) **Next Action:** none

requirements for radioactive waste processing systems and effluent paths (2.2.8)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 12.2-13) "The staff concludes that the Passive Requirements Document does not adequately and explicitly address all associated regulatory requirements for control, monitoring, and sampling of radioactive waste processing systems and effluent paths. Therefore, the staff will review the plant-specific design for the control, monitoring, and sampling of radioactive waste processing systems and effluent paths against the criteria in SRP Section 11.5 during its review of an individual application for final design approval and design certification."	Agree	(DSER) See Abstract	

NRC Review

NRR/SPLB Chandra

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ALWR/NRC OPEN ISSUES

P.12.V-2

Status: Closed(Cert) **Next Action:** none

GRWPS hydrogen control design (3.3.4)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p12.3-8) "Although EPRI committed, in Chapter 1 of the Passive Requirements Document, to comply with SRP Section 11.3, Chapter 12 of the Passive Requirements Document does not address all the criteria in the SRP. The staff expects that applicants referencing the Passive Requirements Document will comply with the SRP, as committed to in Chapter 1"	Agree	(DSER) See Abstract	

NRC Review

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ALWR/NRC OPEN ISSUES

P.12.V-3

Status: Open

Next Action: NRC

design of dry waste shipping containers (5.5)

Abstract	Industry Position	NRC Position	Action Description
(DSER, p 12.5-7) "the staff will review the design of shipping containers both for wet and dry solid wastes on a plant-specific basis against the acceptance criterion in SRP Section 11.4, 'Solid Waste Management Systems' and applicable sections of 10 CFR Part 61."	The NRC's observation is correct and is intended to be that way in the URD. That is, they do not specify shipping container design. However, it is not clear why shipping containers should be considered part of the plant design and license. The NRC should consider deleting shipping container design from the SRP 11.4 insofar as it relates to plant design.	(DSER) See Abstract	NRC review this response.

NRC Review

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Enclosure 2

List of DSER Issues in Chapters 1 and 1A with Responses Outstanding

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P.1A.V-2
P.1A.V-3
P.1A.V-6