ADVANCED LIGHT WATER REACTOR REQUIREMENTS DOCUMENT

APPENDIX A

PRA KEY ASSUMPTIONS AND GROUNDRULES

Prepared For
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APPENDIX A
PRA KEY ASSUMPTIONS AND GROUNDRULES

FOREWORD

The EPRI Advanced Light Water Reactor (ALWR) Utility Requirements Document contains a set of design requirements for the ALWR. As part of the detailed design of a plant to these requirements, a probabilistic risk assessment (PRA) will be required. The primary purposes of the PRA are as follows:

- To provide a mechanism for assuring a balanced design from a risk standpoint.
- To demonstrate that the detailed plant will be capable of meeting the utility investment protection requirement frequency for core damage of $\leq 10^{-5}$ per reactor year.
- To demonstrate that the detailed plant design with the plant located at a representative site will be capable of meeting the public risk requirement of 10^{-6} per reactor year for releases > 25 rem.

In addition, the PRA will be used to accomplish a number of other objectives, including the following:

- To identify the leading core-damage and risk sequences.
- To identify potential vulnerabilities to core damage and containment performance for the ALWR design.
- To satisfy the NRC Severe Accident Policy Statement requirement that a PRA be conducted (Ref. 1).
- To serve as a basis for an accident-management program.

It is anticipated that the PRA will be performed in parallel with the plant detailed design and that it will be completed at the time of licensing certification package completion, thus enabling use of the PRA to support certification. For portions of the plant design which are not fully detailed for certification, interface requirements will have to be defined by the Plant Designer to allow a complete PRA. The PRA will assume that the plant will be built in accordance with the detailed design and any interfacing requirements. In order to obtain a meaningful assessment of the important contributors to core-damage frequency and risk, it is intended that the PRA use best-estimate methods, data, and assumptions, to the extent that they are available and it is practical to do so.

In order to provide guidance to be used in performing the PRA, this PRA Key Assumptions and Groundrules Document has been prepared. The purposes of this document are the following:

- Define the purposes of the PRA as discussed in the above paragraphs.
- Define the scope of the PRA, including sources of risk to be considered, types of events to be analyzed and those to be explicitly excluded, and level of detail of the analysis.

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- Identify previously developed methods to be used. Most of the methods are identified by reference. Examples are NUREG/CR-2300 (Ref. 2), which is referenced extensively for analysis of external events since it has undergone comprehensive peer review, and NUREG/CR-2815 (Ref. 3), which is referenced in areas where it is considered an appropriate supplement. Identify new or improved methods where previously developed methods were determined to be lacking or better methods have recently become available. Examples are in the areas of common-cause failures and human interactions.

- Define procedures to be used in those few cases where existing procedures are incomplete or conflicting. Examples are the definition of severe core damage and treatment of uncertainties.

This PRA Key Assumptions and Groundrules document does not define complete, detailed PRA procedures and methods but, rather, relies primarily on existing procedures and methods by reference, and supplements these where necessary.

The intention of this document is to specify an approach that will result in a comprehensive, high quality, understandable PRA. If the Plant Designer takes exception to any of the requirements [as indicated by the term "shall"] of this document, those exceptions shall be listed in the introduction of the PRA report, and the Plant Designer shall justify the approach taken as being appropriate for the intended purpose.

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List of Acronyms

AFW	Auxiliary Feedwater
ALWR	Advanced light-water reactor
CCDF	Complementary cumulative distribution function
CCF	Common-cause failure
ECCS	Emergency core-cooling system
LOCA	Loss-of-coolant accident
NRC	Nuclear Regulatory Commission
PGA	Peak ground acceleration
PRA	Probabilistic risk assessment
PSH	Probabilistic seismic hazard
RCIC	Reactor-core isolation cooling
SHARP	Systematic human action reliability procedure
SQUG	Seismic Qualification Utility Group
SRSS	Square root of the sum of the squares

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Paragraph No.	Assumption/Groundrule	Rationale	Rev.
1	**OVERALL SCOPE AND METHODS**	**OVERALL SCOPE AND METHODS**	0
1.1	**SCOPE**	**SCOPE**	0
1.1.1	The scope of the PRA performed for use in comparing with the ALWR core-damage frequency requirement and the site boundary-dose requirement shall encompass evaluation of the core damage frequency, assessment of containment response and estimation of release frequencies and magnitudes, and analysis of off-site consequences. In the terminology of the PRA Procedures Guide (Ref. 2), a Level 3 PRA is required.	A Level 3 PRA is required in order to obtain estimates of the risk measures needed to compare against the overall requirements for the ALWR.	0
1.1.2	The scope of the PRA shall include internal and external events **except** sabotage. Sabotage, by either an external armed force or by an internal saboteur or group, shall be explicitly excluded from the PRA.	The inclusion of external events (fire, flood, earthquake, etc.) as well as internal events is done to ensure that the plant design provides balanced protection from all classes of events that can be reasonably envisioned. The sole exception is sabotage. The frequency of acts of sabotage cannot be meaningfully quantified, and thus the core damage frequency from sabotage sequences cannot be estimated. Plant protection from acts of sabotage will continue to be provided by deterministic requirements for physical barriers, security systems, security forces, etc. (The qualitative insights gained by the performance of PRA will be used in determining which deterministic means for sabotage protection are most effective.)	0
1.1.3	The plant shall be assumed to be correctly designed to meet the plant functional requirements, and shall be assumed to be constructed as designed.	The PRA is intended to analyze design capability, as stated in the design documentation, as well as operational aspects, and is not intended to be a primary means of identifying or resolving design errors or construction deficiencies.	0

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Paragraph No.	Assumption/Groundrule	Rationale	Rev.
1.1.4	**Initiating Events - Modes of Operation**	**Initiating Events - Modes of Operation**	0
	The PRA used to test against the requirements stated in Sections 1.2 and 5.1 of this document shall be limited to consideration of initiating events that occur at nominal full-power operation and of the radionuclide inventory of the fuel in the reactor vessel.	Plant-specific analyses performed to date have found that the frequency of core damage and the public health risk of initiating events that occur in states other than power operation and from sources other than the fuel in the vessel are not as significant as events originating from power operation and involving the core inventory. These studies have indicated that such events have been important for specific plants for which procedures, training, and administrative controls were less than optimal. For current plants, design changes were generally unnecessary for the non-power events. For the ALWR, the opportunity has been taken to address many of these non-power events in the design, although procedures, training, and administrative controls will also be necessary. The requirements for the ALWR (refer to Paragraph B.10 of Requirements Document, Chapter 5) have specifically addressed the events that have occurred in current generation plants by eliminating specific failure modes, adding additional shutdown heat removal redundancy, reducing the opportunity for such events to occur, and providing significant emphasis in the design requirements with respect to preventing these events.	0

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Paragraph No.	Assumption/Groundrule	Rationale	Rev.
1.1.5	**Consequence Analysis**	**Consequence Analysis**	0
	Off-site consequences shall be calculated using meteorological and demographic data for a reference site. The reference site shall be bounding for most sites in the United States, and shall be as defined in Section 5.2.	Use of the reference site is desired since the primary purpose of the PRA is to assess the plant design relative to the overall requirements. Estimation of off-site consequences for a reference site that bounds the majority of U.S. sites permits determination of whether the design should be adequate from a risk standpoint, irrespective of the site at which it may be located. Moreover, since the PRA will be performed at the design-certification stage, no specific site will be available for the analysis.	0
1.2	**DEFINITION OF CORE DAMAGE**	**DEFINITION OF CORE DAMAGE**	0
1.2.1	"Core Damage" shall be assumed to have occurred if and only if both of the following have occurred:	A practical definition for core damage that is structured to be useful to the PRA analyst is needed. This definition is intended to represent a condition where there is extensive physical damage to the core such that fuel assemblies would be disfigured either by mechanical fracturing or by melting, and removal of intact fuel assemblies or groups of assemblies could not be accomplished. (It is understood that this definition results in some event sequences where the core is overheated to a lesser extent and there may be clad perforation, deformation, or ballooning of fuel rods that would not be classified as core damage.)	0

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Paragraph No.	Assumption/Groundrule	Rationale	Rev.
1.2.1.1	The collapsed level in the reactor has decreased such that active fuel in the core has been uncovered.	This is a conservative condition for core damage because actual damage is not likely to occur until water level is lower (i.e., nearer the mid-plane of the core.) However, if the core remains covered, then prevention of core damage is assured. This portion of the definition may allow the analyst to eliminate events which do not produce core uncovering from the analysis without having to use a detailed analysis. This condition may be hand-calculated using reactor coolant system volumes, temperatures, decay-heat levels, and heat-removal rates.	0
1.2.1.2	A temperature of 2200°F or higher is reached in any node of the core as defined in a best-estimate thermal-hydraulic calculation.	This second tier of the definition is provided so that if a probabilistically important sequence exceeds the core uncovering criterion stated in 1.2.1.1, the analyst has the option of demonstrating that the fuel temperature is acceptable. The temperature selected considers the following. At an actual temperature of about 2800°F, the rate of zircaloy oxidation increases rapidly, and the exothermic reaction will proceed to rapidly heat the core further. A temperature criterion of 2200°F to avoid excessive zircaloy oxidation has substantial technical basis from emergency core-cooling system (ECCS) research to date, and the practical impact of the difference between 2200°F and 2800°F with respect to the ability to obtain a meaningful estimate of core-damage frequency is expected to be negligible. The MAAP code is the currently available calculational tool that is expected to be used for such calculations.	0

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Paragraph No.	*Assumption/Groundrule*	*Rationale*	*Rev.*
1.2.2	**Core Damage Frequency Requirement**	**Core Damage Frequency Requirement**	0
	The plant design shall be such that a realistic assessment of the mean core-damage frequency will produce a best 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 mean value is the point estimate that has been chosen for this comparison.	0
1.3	**POINT ESTIMATE QUANTIFICATION**	**POINT ESTIMATE QUANTIFICATION**	0
	For each basic 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 or expected value shall be the point estimate used for this purpose. These mean values shall be propagated through the PRA models, and mean frequencies shall be obtained for core damage sequences and radionuclide release categories of interest.	PRA results, in the form of estimated mean frequency of core damage and estimated mean frequency of a 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 values for quantification and comparison to the ALWR Top-Level Requirements has been specified for several reasons. First, the use of mean values is practical, since propagation of mean values through the PRA logic models will yield a time mean value for the result. Second, the mean value is influenced by extreme values in the distribution. For example, for 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 85th percentile of the distribution. Thus, the use of mean values (rather than other point estimates such as median or mode) for comparison against 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

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Paragraph No.	*Assumption/Groundrule*	*Rationale*	*Rev.*
1.4	**UNCERTAINTY TREATMENT**	**UNCERTAINTY TREATMENT**	0
	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.	Although the mean values will be used for comparison to the quantitative objectives, it is important that their context be clearly understood. Quantitative treatment of some aspects of uncertainty in PRA (e.g., completeness of models and human interactions) is considered intractable. Therefore, a qualitative uncertainty analysis is called for to aid in gaining further insights into the important contributors to risk, and into the potential for variations in the quantitative risk estimates. Quantitative sensitivity studies or other similar approaches may be employed to help to determine the significance of specific areas of uncertainty. Section 12.3 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.	0
1.5	**FORM OF THE RESULTS**	**FORM OF THE RESULTS**	0
	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).	0

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Paragraph No.	*Assumption/Groundrule*	*Rationale*	*Rev.*
2	**PLANT MODELING**	**PLANT MODELING**	0
2.1	**MODEL STRUCTURE**	**MODEL STRUCTURE**	0
	The plant shall be modeled in terms of a set of initiating events, event sequences composed of function or system success or failure, and logic models that describe combinations of basic events that define the possible success and failure states. Each end state of each event tree shall be designated either "success" or "core damage." The core-damage sequences, when combined with success or failure of containment systems, shall be categorized and grouped into plant-damage states for downstream modeling of the containment physical processes.	These provisions are consistent with the state of the art in PRA methods and are appropriate to the intended use of the results. It may be necessary in some cases to perform containment-performance analyses to determine if the end state is success or failure.	0
2.2	**INITIATING EVENTS**	**INITIATING EVENTS**	0
	The analyst shall develop a comprehensive list of potential initiating events for consideration in the PRA. The systematic search for initiating events shall include, as a minimum, examination of summaries of operating experience for current-generation plants, PRAs for plants with similar design characteristics, and review of the system designs, including the system failure models for events unique or specific to the ALWR.	An exhaustive search for possible initiating events is one of the key elements in achieving an acceptable level of completeness for the PRA. The intended use of the PRA as a means for testing design adequacy and the potential that new design features may suggest some initiating events that are different from those that have been found to be important for current-generation plants combine to place additional burden on the analyst to be particularly vigilant in accomplishing this task.	0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
2.3	**SUCCESS CRITERIA**	**SUCCESS CRITERIA**	0
	A definition of success and failure for each function or system represented in each event sequence shall be provided based on realistic analysis of plant response. For economy of resources the analysts may choose to use conservative criteria. In this case, the analysts shall identify where conservative assumptions have been used and review the results to ensure that such conservatism does not obscure insights from the results. The analyst shall also exercise caution to ensure that any assumption or criterion considered to be conservative in one context does not introduce a non-conservatism in some other area.	PRA results are intended to be realistic, not conservative. However, conservative criteria may be applied in areas that are not important to risk to avoid the unnecessary expenditure of resources that might be required to perform more detailed realistic calculations. One problem that arises, however, is that very often an assumption that is conservative in one respect may be non-conservative in another. This is particularly true with regard to assumptions that might affect both the assessment of core-damage frequency and the treatment of containment response. Therefore, the analyst must be certain to understand all implications of conservative criteria.	0
2.4	**SEQUENCE LOGICAL IDENTITY**	**SEQUENCE LOGICAL IDENTITY**	0
	The plant model and the solution and quantification techniques employed shall retain the logical identity of the basic events that comprise each sequence.	In order to understand and check the realism of the results, it is necessary to specifically identify which basic-event combinations contribute to the frequency of the dominant event sequences. It is not considered sufficient only to calculate sequence frequencies. The specific equipment conditions must be known to determine whether recovery by the operations staff is possible and to judge how likely such recovery may be.	0

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Paragraph No.	Assumption/Groundrule	Rationale	Rev.
2.5	**QUANTIFICATION**	**QUANTIFICATION**	0
	The sequence models shall be quantified in an integrated fashion. The following additional groundrules apply to the quantification process.	Integrated quantification is necessary to ensure that dependencies are treated properly, and that conditions important to the recovery analysis are explicitly identified.	0
2.5.1	**Truncation of Sequence Frequencies**	**Truncation of Sequence Frequencies**	0
	For each functional sequence (i.e., an initiating event and the safety functions or principal systems whose failures and successes comprise a sequence), analysts shall retain and account for all event combinations that are higher in frequency than 1% of the highest-frequency combination for that sequence. In no case shall a truncation frequency higher than 1×10^{-8} be applied.	In order to solve the plant's models for dominant sequences, it may be necessary to truncate combinations of basic events whose frequency is below that of interest or significance to the results. Setting a sequence-dependent truncation value ensures that each functional sequence is investigated, and provides additional assurance that large numbers of potentially-important contributors are not truncated. Retaining information regarding low-frequency sequences may also be important with respect to identifying those with a relatively higher potential for containment failure, as well as preserving the ability to assess the effects of certain sensitive areas.	0
2.5.2	**Nested Solution Process**	**Nested Solution Process**	0
	A "nested" approach, whereby support-system models are solved first, and then the failure combinations whose probabilities are greater than a truncation value are used to represent the system model in the sequence quantification, is acceptable. In such an approach, the analyst shall use a truncation value that is consistent with the truncation value for the relevant sequence. Treatment of inter-system dependencies is discussed further in Section 2.6.	System interdependencies have the potential to bypass design redundancy and deserve careful attention in the quantification process. It is therefore important that potentially-important failure modes associated with support systems be retained in the quantification process.	0

Paragraph No.	Assumption/Groundrule	Rationale	R
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, which indicate the effects of the status of one system or safety function on the success or failure of another, shall be incorporated into the sequence event trees or equivalent sequence logic.	This is required for proper modeling of the sequences.	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
2.6.3	**Inter-component Dependencies**	**Inter-component Dependencies**	0
	Inter-component dependencies due to shared root causes of failures shall be modeled and quantified using the methods outlines in Section 2.8 below.	The potential for common-cause failure of key components should be recognized and evaluated using the most recent methods and data.	0
2.6.4	**Dependencies Due to Human Actions**	**Dependencies Due to Human Actions**	0
	Dependencies involving human actions shall be considered using the methods referenced in Section 2.10.	Human actions have the potential to result in the unavailability of multiple components and, consequently, merit particular attention in the assessment of human reliability.	0

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Paragraph No.	*Assumption/Groundrule*	*Rationale*	*Rev.*
2.7	**INTERACTION AND MODELING OF THE CONTAINMENT SYSTEMS**	**INTERACTION AND MODELING OF THE CONTAINMENT SYSTEMS**	0
	The delineation of the core-damage sequences shall be coordinated with the assessment of containment response to ensure that any effects of containment systems or of containment phenomena on the availability of the systems needed to prevent core damage are appropriately reflected in the event trees.	The response of containment or containment systems may impact the ability of the systems providing core cooling to continue to operate. For example, if core cooling is provided for a long period by the reactor core isolation cooling (RCIC) system in a BWR with no heat removal from the suppression pool, the result may be loss of the RCIC turbine, due to high backpressure, and consequential loss of cooling.	0
2.8	**COMMON-CAUSE FAILURES**	**COMMON-CAUSE FAILURES**	0
2.8.1	**Definition**	**Definition**	0
	It is assumed that direct component-to-component and system-to-system functional dependencies are addressed explicitly in the plant model. It is further assumed that common cause initiating events are explicitly addressed under external events and specific internal events. Only root-caused events leading directly to multiple component outages from the shared cause are addressed here.	Great care must be exercised not to double count events but to nevertheless achieve coverage of all dependency types by specific means.	0
2.8.2	The methodology described in the joint EPRI/NUREG report (Ref. 5) on common-cause analysis procedures shall be used. The analyst may choose to use the common-cause factors presented in Section 11 of this appendix, which were developed using this methodology.	This methodology is the culmination of research by many organizations worldwide and represents an industry consensus. It emphasizes qualitative analysis, careful event interpretation, screening, and parameter estimation. Although the source data is necessarily generic for common-cause failures (CCFs), this data must be interpreted in a plant-specific sense to determine applicability.	0

Paragraph No.	*Assumption/Groundrule*	*Rationale*	*Rev.*
2.8.2.1	The simplest parametric model (i.e., the β factor) should be used in the treatment of common-cause failures, except for cases in which the analyst desires specifically to investigate the effects of levels of redundancy beyond two-fold, or if the use of a single factor for common-cause failure of more than two components within a group leads to overestimation of an important sequence frequency. In these cases, the α factor or multiple-Greek letter approach will be used.	EPRI NP-5613 (Ref. 5) illustrates a number of parametric models that may be employed in implementing the common-cause methodology, and allows free choice from among them. The development of the failure data base is more important to the results than the choice of the model. Therefore, the simplest model should be used whenever possible, allowing attention to be focused on the data development.	0
2.8.2.2	The following shall be used as primary sources of multiple-failure data: • EPRI NP-3967, *Classification and Analysis of Reactor Operating Experience Involving Dependent Events*, June 1985 (Ref. 6). • EPRI NP-5777, *Defensive Strategies for Reducing Susceptibility to Common Cause Failures, Vol. 1, Defensive Strategies, Vol. 2 Data Analysis* (Ref. 7).	NP-3967 and NP-5777 are the most recent publications in this area. Both sources incorporate a classification scheme which enables one to apply the data to the methodology described in NP-5613. The following additional sources of data are recommended; however, these documents do not contain a consistent classification scheme. Therefore, it is expected that the analyst will wish to refer to the actual event reports in order to fully evaluate the applicability of the data. • NUREG/CR-2770, *Common Cause Fault Rates for Valves*, February 1983 (Ref. 8). • NUREG/CR-3289, *Common Cause Fault Rates for Instrumentation and Control Assemblies*, May 1983 (Ref. 9). • NUREG/CR-2098, *Common Cause Fault Rates for Pumps*, May 1983 (Ref. 10). • NUREG/CR-2099, *Common Cause Fault Rates for Diesel Generators*, June 1982 (Ref. 11).	0
		These sources are less recent sources than NP-3967 and NP-5777.	0

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Paragraph No.	*Assumption/Groundrule*	*Rationale*	*Rev.*
2.9	**HUMAN INTERACTION**	**HUMAN INTERACTION**	0
2.9.1	An approach that shows consistency, traceability, and realism is needed. The EPRI Systematic Human Action Reliability Procedure (SHARP) analysis framework (Ref. 12) shall be used for this purpose. The analysis must deal explicitly with (a) definition of human actions, (b) screening for importance, (c) task breakdown, (d) representation in relation to systems logic models, (e) iteration between human and hardware modeling, (f) quantification, and (g) documentation.	The EPRI report NP-5546, *Benchmark of SHARP* (Ref. 13), contains an evaluation and critique of SHARP, including suggestions for improvements. The SHARP steps, contained in EPRI NP-3583 (Ref. 12), are important for each type of human interaction. If the individual SHARP steps are followed, the result is likely to be understandable.	0
2.9.2	The analysis requires a disposition of each of the following types of human interactions:	The five different types of human interaction require significantly different treatments. They can all be significant to plant risk.	0
	• Type 1: test and maintenance actions;		0
	• Type 2: actions causing initiating events;		0
	• Type 3: procedural actions leading to appropriate plant response;		0
	• Type 4: actions leading to inappropriate plant response;		0
	• Type 5: recovery or use of initially unavailable equipment.		0
2.9.2.1	For Type 1 actions that remain after screening, an acceptable approach is to use a value generated using the technique for human error rate prediction (THERP, Ref. 14) for a current-generation plant that is representative of the plant design being analyzed.		0

Paragraph No	Assumption/Groundrule	Rationale	Rev.
2.9.2.2	Type 2 interactions are usually contained within the initiating event data sources; however, the analyst should be alert for human actions that can cause initial conditions significantly more severe than the initiating events otherwise chosen for analysis.		0
2.9.2.3	Types 3 and 5 have a strong time dependence. An acceptable correlation, based on actual data, for treating these interactions is the HCR probability-time correlation (Ref. 15).		0
2.9.2.4	Type 4 actions will be excluded.	Current symptom-based procedures greatly reduce the opportunities for serious misdiagnosis.	0
2.9.3	The PRA may consider actions to recover failed functions even if non-safety equipment is involved or if there is no written procedure. All recovery actions proposed must be screened to establish feasibility, using applicable reference material (e.g., procedures, etc.), engineering drawings, design specifications, or by comparison with existing designs.	Focus throughout should be on representing realistic options using realistic quantification. Human interaction can dominate risk. Too much conservatism or optimism in human reliability treatment is very likely to lead to wrong insights being drawn from the PRA. It will be necessary to utilize past and current operator experience to make judgments regarding operator interactions for the ALWR.	0
2.9.4	The PRA analysts shall carefully document any assumptions regarding the content of procedures and the relative priorities of actions as established by procedures and training.	Because of the potential importance of operator actions and the dependence of the assessment of these actions on procedures, it will be very important for the provisions the analysts assume will eventually be reflected in the procedures to be very thoroughly documented.	0

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2.10	**MISSION TIME**	**MISSION TIME**	0
	For equipment required to remain running for successful core cooling after an initiating event, and for containment safeguards systems, a mission time of 24 hours will be used. A mission time of less than 24 hours may be used if the actual mission time is less.	If the core has been successfully cooled for 24 hours, then decay-heat levels are significantly lower than at the start of the transient. The time available for recovery actions and repair of subsequent failures is long enough that the probability of core damage from such events is not significant in comparison to core-damage events within the first 24 hours. If the containment is cooled for 24 hours, then long times exist for recovery from subsequent hardware failures.	0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
2.11	**RELIABILITY DATA**	**RELIABILITY DATA**	0
2.11.1	**Introduction**	**Introduction**	0
	Of necessity, the PRAs for the ALWRs will utilize generic data for initiating-event frequencies and component failure rates. These PRAs should use the most current and representative data available. This portion of the document suggests data, based on a combination of assessments of industry wide operating experience, generic data bases, and plant specific data published in a number of PRAs. This data base can be supplemented for unique components by use of additional data sources as necessary.		0
2.11.2	**Initiating-event Frequencies**	**Initiating-event Frequencies**	0
	In estimating the frequencies for initiating events, experience for current-generation plants should be examined and applied to the ALWRs in an appropriate manner that reflects, to the extent possible, differences in the ALWR designs from current plants.	Although it will be necessary to use generic data derived from the operation of current-generation plants as a basis for the initiating-event frequencies for ALWRs, it is possible for the analysts to examine the specific events in the data base with regard to their applicability to ALWRs. An example of such an approach is provided in Annex A, Sections A1 and A2.	0
2.11.2.1	**Nominal Frequencies for Initiating Events**	**Nominal Frequencies for Initiating Events**	0
	For an initial, nominal set of initiating events, the analyst may use the following frequencies:	The derivation of these frequencies is outlined in the first two sections of Annex A. As indicated above, this treatment outlines an approach to assessing the applicability of experience for current-generation plants for the ALWRs. It is expected that the PRA for an actual design will consider a more detailed breakdown of initiating events than is reflected by this set, necessitating further evaluation of their frequencies.	0

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2.11.2.1	**Nominal Frequencies for Initiating Events (Continued)**			0
	Event	**Annual Frequency**		0
	BWR			0
	Turbine trip	2.3		0
	Loss of main condenser	0.49		0
	Loss of feedwater	0.37		0
	Loss of normal off-site power	0.035		0
	Loss of a major ac power bus	1.5×10^{-3}		0
	Large loss-of-coolant accident	5.8×10^{-4}		0
	Small loss-of-coolant accident	5.1×10^{-3}		0
	Event	**Annual Frequency**		0
	PWR			0
	Reactor/turbine trip	2.8		0
	Loss of main feedwater	0.46		0
	Loss of normal off-site power	0.035		0
	Loss of a major ac power bus	1.5×10^{-3}		0
	Steam line break	1.5×10^{-3}		0
	Large loss-of-coolant accident	3.4×10^{-4}		0

Paragra	Assumption/Groundrule	Rationale	Rev.
	N· ninal Frequencies fo· Initiating Events Frequencies (C ntinued)		0
	Event / **Annual Frequency**		0
	PWR (Continued)		0
	Intermediate loss-of-coolant accident — 3.4×10^{-4}		0
	Small loss-of-coolant accident — 3.0×10^{-3}		0
	Steam generator tube rupture — 6.1×10^{-3}		0
2.11.2.2	**Frequency of Loss of Off-site Power**	**Frequency of Loss of Off-site Power**	0
	The frequency estimated for loss of all off-site power shall reflect consideration of the reserve source. A conditional probability that the reserve source of power will be unavailable (given loss of normal off-site power) of 0.22 may be applied to the frequency of loss of normal off-site power. In addition, the frequency of demand for emergency power for the advanced PWR shall account for the potential for the full load rejection capability to function to avert a need for the reserve or emergency power. A conditional probability of 0.23 may be used for the chance that the initial loss of normal off-site power could be of a nature to preclude use of the full-load rejection capability. The unavailability of the full-load rejection capability itself shall also be added to this conditional probability. A nominal unavailability of 0.1 for the full-load rejection feature may be assumed.	Chapter 11 of the Requirements Document spells out specific requir ents for an independent reserve source c off-site power and, for the advanced PWR, specifies incorporation of a full load rejection capability. Section A2 of Annex A describes in detail the assessment of off-site power experience for curre eration plants used to obtain the frequency of loss of r al off-site power and the conditional unavailabilities of r erve power and the full-load rejection capabilities.	0

Paragraph No.	*Assumption/Groundrule*	*Rationale*	*Rev.*
2.11.3	**Component Failure Data**	**Component Failure Data**	0
	The following component failure data are recommended:	The component failure rates were estimated based on a survey of generic data sources and available plant specific experience. The details of the survey itself are provided in Section A3 of Annex A.	0

Component	**Failure Mode**	**Failure Rate**	**Rev.**
Motor operated valve	Fails to operate on demand Transfers closed	4.0×10^{-3}/d 1.3×10^{-7}/hr	0
Air operated valve	Fails to operate on demand Transfers closed	2.0×10^{-3}/d 1.5×10^{-7}/hr	0
Check valve (other than stop)	Fails to operate on demand Transfers closed Reverse leakage (gross)	2.0×10^{-4}/d 2.0×10^{-7}/hr 6.0×10^{-7}/hr	0
Stop check valve	Fails to operate on demand Transfers closed Reverse leakage (gross)	1.0×10^{-3}/d 2.0×10^{-7}/hr 6.0×10^{-7}/hr	0
Check valve	Internal rupture	5.0×10^{-9}/hr	0
Manual valve	Plugs/transfers closed	3.7×10^{-8}/hr	0

Paragraph No.	Assumption/Groundrule			Rationale	Rev.
2.11.3	**Component Failure Data (Continued)**				0
	Component	**Failure Mode**	**Failure Rate**		0
	Pressurized safety valve (PWR)	Fails to open on demand Fails to reclose	1.0×10^{-3}/d 7.0×10^{-3}/d		0
	Safety relief valve (BWR)	Fails to open on demand Fails to reclose	6.0×10^{-3}/d 6.5×10^{-3}/d		0
	Pilot operated relief valve	Fails to open on demand Fails to reclose	7.0×10^{-3}/d 2.5×10^{-2}/d		0
	Motor driven pump (all types)	Fails to start on demand Fails to run	2.0×10^{-3}/d 2.5×10^{-5}/hr		0
	Motor driven pump (LPI/RHR)	Fails to start on demand Fails to run	2.3×10^{-3}/d 1.3×10^{-5}/hr		0
	Motor driven pump (safety injection)	Fails to start on demand Fails to run	1.0×10^{-3}/d 5.0×10^{-5}/hr		0
	Motor driven pump (emerg. feedwater)	Fails to start on demand Fails to run	3.0×10^{-3}/d 1.5×10^{-4}/hr		0
	Motor driven pump (service water)	Fails to start on demand Fails to run	2.4×10^{-3}/d 3.2×10^{-5}/hr		0
	Motor driven pump (comp. cooling)	Fails to start on demand Fails to run	1.3×10^{-3}/d 5.0×10^{-6}/hr		0
	Motor driven pump (BWR CRD)	Fails to start on demand Fails to run	2.4×10^{-3}/d 2.4×10^{-6}/hr		0
	Motor driven pump (cont. spray)	Fails to start on demand Fails to run	5.0×10^{-3}/d 5.0×10^{-5}/hr		0

APPENDIX A: PRA KEY ASSUMPTIONS AND GROUNDRULES

Paragraph No.	Assumption/Groundrule		Rationale	Rev
2.11.3	**Component Failure Data (Continued)**			0
Component	**Failure Mode**	**Failure Rate**		0
Turbine driven pump (AFW)	Fails to start on demand Fails to run	1.5×10^{-2}/d 3.0×10^{-4}/hr		0
Turbine driven pump (RCIC)	Fails to start on demand Fails to run	2.0×10^{-2}/d 4.0×10^{-4}/hr		0
Diesel driven pump	Fails to start on demand Fails to run	2.0×10^{-2}/d 1.0×10^{-4}/hr		0
Motor driven air compressor	Fails to start on demand Fails to run	1.0×10^{-2}/d 1.0×10^{-4}/hr		0
Blower/ventilation fan	Fails to start on demand Fails to run	6.0×10^{-4}/d 1.0×10^{-5}/hr		0
Room chiller unit	Fails to start on demand Fails to run	8.1×10^{-3}/d 5.0×10^{-6}/hr		0
Motor driven strainer	Fails to start on demand Fails to run	2.7×10^{-5}/d 5.0×10^{-6}/hr		0
Filter/strainer	Plugs	2.0×10^{-6}/hr		
Heat exchanger	Fails while operating (leaks, plugs)	1.0×10^{-6}/hr		0
Tank	Fails catastrophically	1.0×10^{-7}/hr		0
Off-site power	Fails following reactor trip	1.2×10^{-3}/d		0
Diesel generator	Fails to start and load Fails to run	1.4×10^{-2}/d 2.4×10^{-3}/hr		0
Gas turbine-generator	Fails to start on demand Fails to run	2.5×10^{-2}/d 2.9×10^{-6}/hr		0

Paragraph No. | *Assumption/Groundrule* | *Rationale* | *Rev.*

2.11.3 Component Failure Data (Continued) — Rev. 0

Component	Failure Mode	Failure Rate	Rev.
Battery	Fails to provide output on demand	5.0×10^{-4}/d	0
Battery charger	Fails to maintain output	7.0×10^{-6}/hr	0
Circuit breaker (4 kv)	Fails to close on demand Opens spuriously	3.0×10^{-4}/d 6.0×10^{-7}/hr	0
Circuit breaker (≤ 600 v)	Fails to close on demand Opens spuriously	4.0×10^{-4}/d 5.0×10^{-7}/hr	0
Transformer (high voltage)	Fails to continue operating	1.2×10^{-6}/hr	0
Transformer (4 kv to 600/480 v)	Fails to continue operating	7.0×10^{-7}/hr	0
Transformer (lower voltage)	Fails to continue operating	8.0×10^{-7}/hr	0
Fuse	Opens spuriously	5.0×10^{-7}/hr	0
Electrical buswork	Fails during operation	2.0×10^{-7}/hr	0
Inverter	Fails during operation	2.0×10^{-5}/hr	0
Relay	Fails to operate on demand Operates spuriously	1.0×10^{-4}/d 6.0×10^{-7}/hr	0
Flow transmitter	Output fails during operation	6.0×10^{-6}/hr	0
Pressure transmitter	Output fails during operation	5.0×10^{-6}/hr	0
Level transmitter	Output fails during operation	5.0×10^{-6}/hr	0
Temperature transmitter	Output fails during operation	1.0×10^{-6}/hr	
Pressure switch	Fails during operation Fails to respond on demand	3.0×10^{-7}/hr 2.0×10^{-4}/d	0

Paragraph No.	Assumption/Groundrule			Rationale	Rev.
2.11.3	**Component Failure Data (Continued)**				0
	Component	**Failure Mode**	**Failure Rate**		0
	Level switch	Fails during operation Fails to respond on demand	3.0×10^{-7}/hr 1.0×10^{-5}/d		0
	Reactor core isolation cooling (BWR)	Unavailable due to maint.	4.0×10^{-3}	Note that the maintenance unavailabilities generally reflect a philosophy of not performing on-line preventive maintenance. These are considered to be the most appropriate values available, but the analysts may need to reconsider them for the specific application in the PRA.	0
	High pressure injection train (BWR)	Unavailable due to maint.	4.0×10^{-3}		
	Low pressure injection train (BWR)	Unavailable due to maint.	2.0×10^{-3}		
	Emergency service water train (BWR)	Unavailable due to maint.	2.0×10^{-3}		0
	Standby liquid control train (BWR)	Unavailable due to maint.	3.0×10^{-3}		0
	Turbine driven AFW train (PWR)	Unavailable due to maint.	5.0×10^{-3}		0
	Motor driven AFW train (PWR)	Unavailable due to maint.	2.0×10^{-3}		0
	Safety injection train (PWR)	Unavailable due to maint.	2.0×10^{-3}		0
	Residual heat removal train (PWR)	Unavailable due to maint.	2.0×10^{-3}		0
	Containment spray train (PWR)	Unavailable due to maint.	2.0×10^{-3}		0
	Diesel generator	Unavailable due to maint.	6.0×10^{-3}		0
	Gas turbine-generator	Unavailable due to maint.	6.8×10^{-2}		0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
2.11.4	**Common Cause Factors**	**Common Cause Factors**	0
	As an alternative to the implementation of the method for the assessment of common cause failure rates outlined in Section 2.8.2, the following nominal values may be used. The values were developed for application using the multiple-Greek letter approach.	The common cause factors were developed using the methods described in EPRI-5613 (NUREG/CR-4730), *Procedures for Treating Common Cause Failures in Safety and Reliability Studies* (Ref. 5). The methods were applied to the base provided in EPRI-3967, *Classification of Dependent Failures*. The analyst may choose to use these values rather than expend the effort to implement the procedures in EPRI-5613, as outlined in Section 2.8.2.	0

Component	Failure Mode	Number of Failures	Common Cause Factor	Rev.
				0
Safety injection pump	Fails to start	2 of 2	1.4×10^{-1}	0
		2 of 4	4.7×10^{-2}	
		3 of 4	7.6×10^{-3}	
		4 of 4	$3.[illegible] \times 10^{-3}$	
	Fails to run	2 of 2	8.0×10^{-3}	
		2 of 4	7.6×10^{-3}	
		3 of 4	1.7×10^{-4}	
		4 of 4	7.4×10^{-6}	
Emergency feedwater pump	Fails to start	2 of 4	3.0×10^{-2}	0
		3 of 4	1.3×10^{-3}	
		4 of 4	4.1×10^{-5}	
	Fails to run	2 of 4	3.0×10^{-3}	
		3 of 4	2.6×10^{-5}	
		4 of 4	7.1×10^{-7}	

Paragraph No.	*Assumption/Groundrule*	*Rationale*	*Rev.*
2.11.4	**Common Cause Factors (Continued)**		0

Component	Failure Mode	Number of Failures	Common Cause Factor	Rev.
Low pressure injection pump	Fails to start	2 of 2	1.4×10^{-1}	0
		2 of 3	5.4×10^{-2}	
		3 of 3	1.4×10^{-2}	
	Fails to run	2 of 2	3.9×10^{-2}	
		2 of 3	1.9×10^{-2}	
		3 of 3	1.6×10^{-3}	
Containment spray pump	Fails to start	2 of 2	1.3×10^{-1}	0
Service water/CCW pump	Fails to start	2 of 3	5.6×10^{-2}	0
		3 of 3	1.7×10^{-2}	
		2 of 4	3.8×10^{-2}	
		3 of 4	4.9×10^{-3}	
		4 of 4	2.2×10^{-3}	
	Fails to run	2 of 3	3.6×10^{-2}	
		3 of 3	3.9×10^{-3}	
		2 of 4	2.2×10^{-2}	
		3 of 4	1.1×10^{-3}	
		4 of 4	1.8×10^{-4}	
Motor operated valve	Fails to operate on demand	2 of 2	6.8×10^{-2}	0
		2 of 3	3.2×10^{-2}	
		3 of 3	4.5×10^{-3}	
		2 of 4	2.1×10^{-2}	
		3 of 4	1.4×10^{-3}	
		4 of 4	2.9×10^{-4}	
	Transfers closed	2 of 4	1.6×10^{-2}	
		3 of 4	8.5×10^{-4}	
		4 of 4	1.4×10^{-4}	

Paragraph No.	*Assumption/Groundrule*	*Rationale*	*Rev.*
2.11.4	**Common Cause Factors (Continued)**		0

Component	Failure Mode	Number of Failures	Common Cause Factor	Rev.
				0
Diesel generator	Fails to start	2 of 2	3.8×10^{-2}	0
		2 of 3	1.9×10^{-2}	
		3 of 3	1.3×10^{-3}	
	Fails to run	2 of 2	6.8×10^{-2}	
		2 of 3	3.2×10^{-2}	
		3 of 3	3.8×10^{-3}	
Dc battery	Fails on demand	2 of 2	7.3×10^{-2}	0
		2 of 3	9.2×10^{-2}	
		3 of 3	1.0×10^{-2}	

Paragraph No.	Assumption/Groundrule		Rationale	Rev.
2.11.5	**Non-recovery Data for Loss of Off-site Power**		**Non-recovery Data for Loss of Off-site Power**	0
	The recommended values for the conditional probability of failure to restore off-site power, as a function of time following plant trip, are as follows:		The non-recovery values are based on an assessment of experience for current-generation plants. This assessment is described in Section A2 of Annex A.	0
	Time (hr)	**Probability of not recovering power**		0
	0.5	0.61		0
	1	0.54		0
	2	0.32		0
	3	0.25		0
	4	0.18		0
	5	0.14		0
	6	0.14		0
	7	0.14		0
	8	0.11		0
	9	0.11		0
	10	0.11		0
	11	0.071		0
	12	0.019		0
	13	0.013		0

Paragraph No.	Assumption/Groundrule		Rationale	Rev.
	Time (hr)	**Probability of not recovering power**		0
	14	9.1×10^{-3}		0
	15	6.1×10^{-3}		0
	16	4.1×10^{-3}		0
	17	2.7×10^{-3}		0
	18	1.8×10^{-3}		0
	19	1.2×10^{-3}		0
	20	7.5×10^{-4}		0
	21	4.8×10^{-4}		0
	22	3.1×10^{-4}		0
	23	1.9×10^{-4}		0
	24	1.3×10^{-4}		0

APPENDIX A: PRA KEY ASSUMPTIONS AND GROUNDRULES

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
3	**EXTERNAL EVENTS**	**EXTERNAL EVENTS**	0
3.1	**INITIATING EVENTS IDENTIFICATION**	**INITIATING EVENTS IDENTIFICATION**	0
3.1.1	The external events, identified in Sections 3.2 and 3.3 and as listed in the PRA Procedures Guide (Ref. 2), will be considered in the performance of a PRA on an ALWR.	This list of potential external initiating events was taken from ANSI/ANS-2.12-1978 (Ref. 16) and is considered to be an exhaustive listing of the external initiating events which should be considered for an ALWR PRA.	0
3.1.2	The methods identified in the PRA Procedure Guide (Ref. 2) will be used for screening of external events except where otherwise specified in this document.	To ensure consistent treatment of external events in the PRAs to be performed during the ALWR design process, a single source for methodology is specified.	0
3.2	**EVENTS THAT MAY BE EXCLUDED BASED ON QUALITATIVE EVALUATION**	**EVENTS THAT MAY BE EXCLUDED BASED ON QUALITATIVE EVALUATION**	0
	The following external events shall be reviewed to ensure that they are precluded as a result of either design, siting, or low frequency of occurrence.	Some of the initiators listed in the PRA Procedures Guide have been shown to be important risk contributors for older plants. Many of these events can be addressed by design improvements or proper siting. The initiating events listed are considered not to be important contributors based on improved design, proper siting, and low probability. The evaluation includes credit for design and siting regulations such as regulatory guides or ANSI/ANS standards.	0
3.2.1	Avalanche, landslide, volcanic activity, soil shrink-swell consolidation.	It is anticipated that the ALWR will not be located at a site which would be vulnerable to these events.	0
3.2.2	Drought, low lake or river level, high summer temperature, river diversion.	The ultimate heat sink will be designed to account for low water level or lack of water, and will be designed for a lengthy period of operation without external makeup.	0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
3.2.3	Tornadoes and extreme winds (including sandstorms), except for loss of off-site power, as noted in Section 3.3.	Tornadoes encompass these initiators. The design of the ALWR will eliminate the concern over this initiator. Building materials, strengths, and missile barrier design will be such that the impact to the plant of tornadoes will, at worst, only generate a loss of off-site power. Losses of off-site power which are caused by tornadoes away from the site (i.e., through grid upsets) are included in the loss of off-site power data. Therefore, only the contribution to core damage attributed to an extended loss of off-site power due to a site strike needs to be addressed. A simplified methodology for this analysis is presented in Section 3.3.	0
3.2.4	Forest fire.	The plant design requires that the site be cleared and that adequate fire-protection provisions to mitigate the effects of a forest fire be provided. The frequency of this event is also included in the frequency loss of off-site power to account for the potential consequential failure of off site sources.	0
3.2.5	Frost.	Snow and ice encompass this initiator.	0
3.2.6	Hail.	Other missiles, such as those resulting from extreme winds, are more serious and govern.	0
3.2.7	Industrial or military facility accidents.	The site shall be in compliance with regulations which require that the site be outside the radius of influence of potential explosions due to existing industrial or military facilities.	0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
3.2.8	Lightning.	The effects of lightning strikes on the plant structures are factored into plant design. The potential for loss of off-site power due to lightning strikes is reflected in the data base used to generate the frequency of loss of off-site power.	0
3.2.9	Hurricanes, except for loss of off-site power, as described in Section 3.3.	Plant design will be such that the winds and water produced from this event will not be significant to risk. In addition, the potential for indirect effects on the plant, such as debris from the hurricane blocking the intake structure, will be precluded. The effects of a hurricane at the plant site on the availability of off-site power should be treated as noted for tornados in Sections 3.2.3 and 3.3.	0
3.2.10	Low winter temperatures.	Thermal stress and embrittlement are covered by applicable design codes and standards. Other effects of low temperature are accommodated by specific design requirements.	0
3.2.11	Pipeline accidents (natural gas, etc.) and toxic gas release.	The site selected for the construction of an ALWR will be outside the radius of influence of potential explosions of leaks due to existing pipelines, or major toxic gas storage areas.	0
3.2.12	Snow and ice cover.	The plant will be designed for higher loadings than those produced by snow. However, the designer should review the design to assure that all necessary ventilation paths are free of snow blockage. The frequency for loss of off-site power includes the effects of ice storms.	0
3.2.13	Turbine-generated missiles.	Proper orientation of the turbine with respect to safety-related equipment will be such that the hazard from turbine-generated missiles will be negligible.	0
3.2.14	Meteorite.	All sites have approximately the same frequency of occurrence. This frequency is sufficiently low that it may be neglected.	0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
3.2.15	Release of chemicals in on-site storage.	It is anticipated that the amounts of chemicals stored on-site will be kept at a level such that it will not impact plant risk. In addition, the chemical form will be such that gaseous releases will be precluded.	0
3.2.16	Transportation accidents (including aircraft, ground transportation, and water transportation).	The location of the plant site with respect to airports and air traffic results in a negligible contribution to core damage. Plant security and other barriers preclude any significant contribution from other transportation accidents. The use of closed cycle cooling systems will eliminate the potential for boat or barge impact.	0
3.2.17	External flooding (including coastal erosion, high tide, high lake level, high river stage, flooding due to intense rainfall or snow melt, flooding due to ice blockage, seiche, storm surge, tsunami and wave action).	The site selection process will eliminate many of these individual sources of external flooding. During the site selection process, the maximum heights of the listed water levels must be deterministically calculated to ensure that the safety structures are located above projected flood level. Proper placement of these structures will eliminate the risk due to these events.	0
3.2.18	Fog	Fog may impact occurrence frequency for transportation accidents. However, these effects are contained in the accident data.	0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
3.2.19	Internal fire	Internal fire is not expected to be a major contributor to core damage frequency due to design improvements. Chapter 6, Section 2.3.3 of the EPRI Requirements Document provides requirements for separation and three-hour fire barriers. The implementation of these requirements is expected to provide a level of fire protection such that, at most, a single safety train will fail. This is an improvement over the plant performance that has been observed in prior plant PRAs. These PRAs have identified the potential for total system failures due to inadequate barriers or separation. Given the low initiating event frequency of internal fire and its expected consequences, transient sequences (such as a loss of an electrical bus) are expected to encompass the impact of internal fire events. Therefore, a detailed probabilistic assessment is not required.	0
3.2.20	Internal flooding	The requirements contained in Chapter 6, Section 2.3.6 provide for significant plant protection from internal flooding. For reasons similar to those discussed for internal fire, a detailed assessment of internal flooding is not required.	0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
3.3	**EVENTS WHICH MAY REQUIRE QUANTITATIVE ASSESSMENT FOR EACH ALWR**	**EVENTS WHICH MAY REQUIRE QUANTITATIVE ASSESSMENT FOR EACH ALWR**	0
	Some of the external initiating events listed in the PRA Procedures Guide (Ref. 2) may not be able to be excluded based on a qualitative evaluation. These events may require a site-specific quantitative evaluation. Past PRAs have shown the following external initiating events to require additional analysis. Therefore, it is very important that the evaluation be well documented as to whether a qualitative or quantitative evaluation is performed.		0
3.3.1	**Tornado Assessment (Site Strike)**	**Tornado Assessment (Site Strike)**	0
	The tornado assessment will be performed using a simplified loss of off-site power model.	Because ALWR structures will be reinforced concrete and careful attention will be paid to physical separation of divisions of safety systems, the frequency of a tornado strike or event involving high wind that could cause sufficient failures to lead to core damage is extremely low. The most serious potential effect is likely to be a loss of off-site power, with restoration of power more difficult than would usually be the case for other causes. Therefore, a simplified model is sufficient, providing that it addresses appropriate combinations of random failures (e.g., of diesel generators) in conjunction with an extended loss of off-site power.	0
3.3.1.1	Independent random failures of equipment can be excluded if the failure rate is less than 10^{-3}.	This probability level will result in simplifying the model. Because the initiator frequency is low, it is not expected that events less than 10^{-3} would impact the result.	0
3.3.1.2	The probability of failure to recover from a loss of off-site power within 24 hours following a tornado site strike will be assumed to be 1.0.	This is a conservative assumption for the analysis. However, to address the significant uncertainty about the ability to restore ac power, this assumption will be made.	0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
3.3.1.3	The plant site area used to determine the frequency of site strike will be .14 square miles.	Tornado effects are typically important for a region equivalent to a square 2000 ft on a side. This distance is typically used to define the plant site when determining the plant site strike frequency. Based on prior PRA analyses, tornado missiles are not important at distances beyond 2000 feet. This value, when multiplied by the tornado frequency, in units of tornadoes/square mile - year, yields the annualized site-strike frequency.	0
3.3.2	**Earthquake**	**Earthquake**	0
	A seismic risk analysis shall be performed as part of the PRA.	The objectives of the seismic risk portion of the PRA are to assure that the standardized plant at the certification stage has a balanced design from a seismic risk standpoint as well as to demonstrate that the ALWR Requirements Document risk requirements can be met. This is consistent with the basic purpose of the overall PRA as expressed in the Foreword to this Appendix A, PRA Key Assumptions and Groundrules. The emphasis of the seismic PRA at the certification stage will be on the system's contributions to seismic risk. It is considered that there is significant value to a disciplined review of seismic risk considering seismic and non-seismic failures.	0

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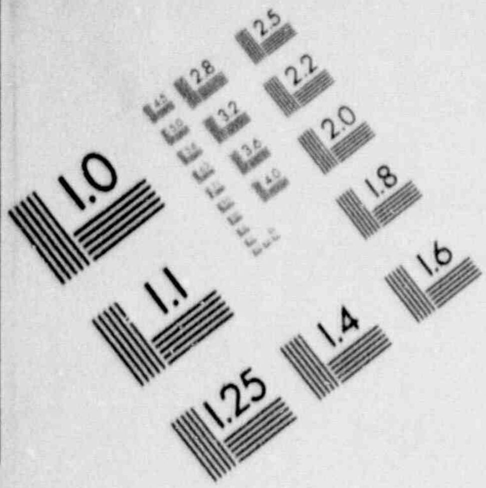
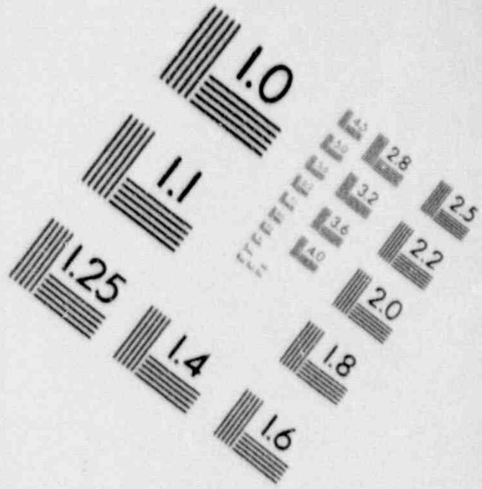
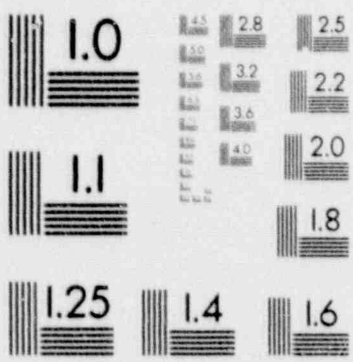
Paragraph No.	Assumption/Groundrule	Rationale	Rev.
3.3.2.1	**Seismic Hazard Analysis**	**Seismic Hazard Analysis**	0
	To be completed by 8/15/89.		0

1

IMAGE EVALUATION TEST TARGET (MT-3)







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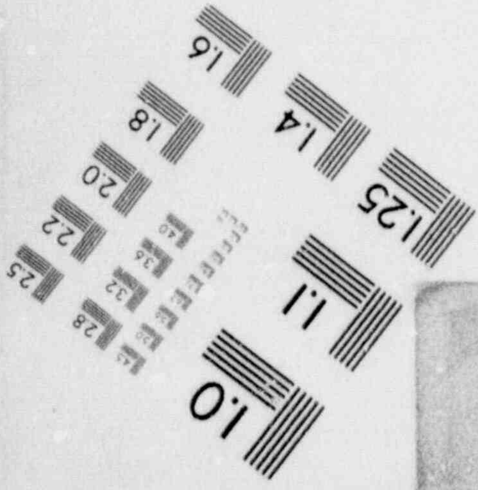
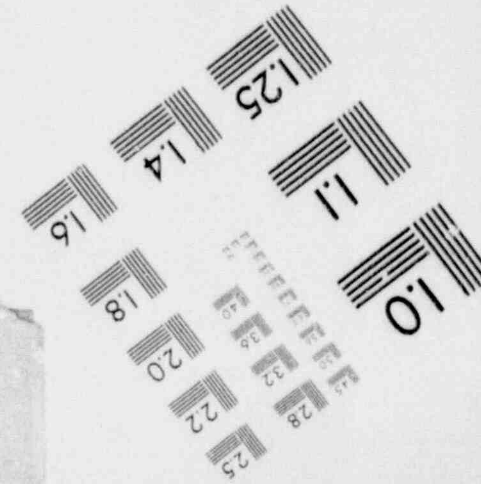




1

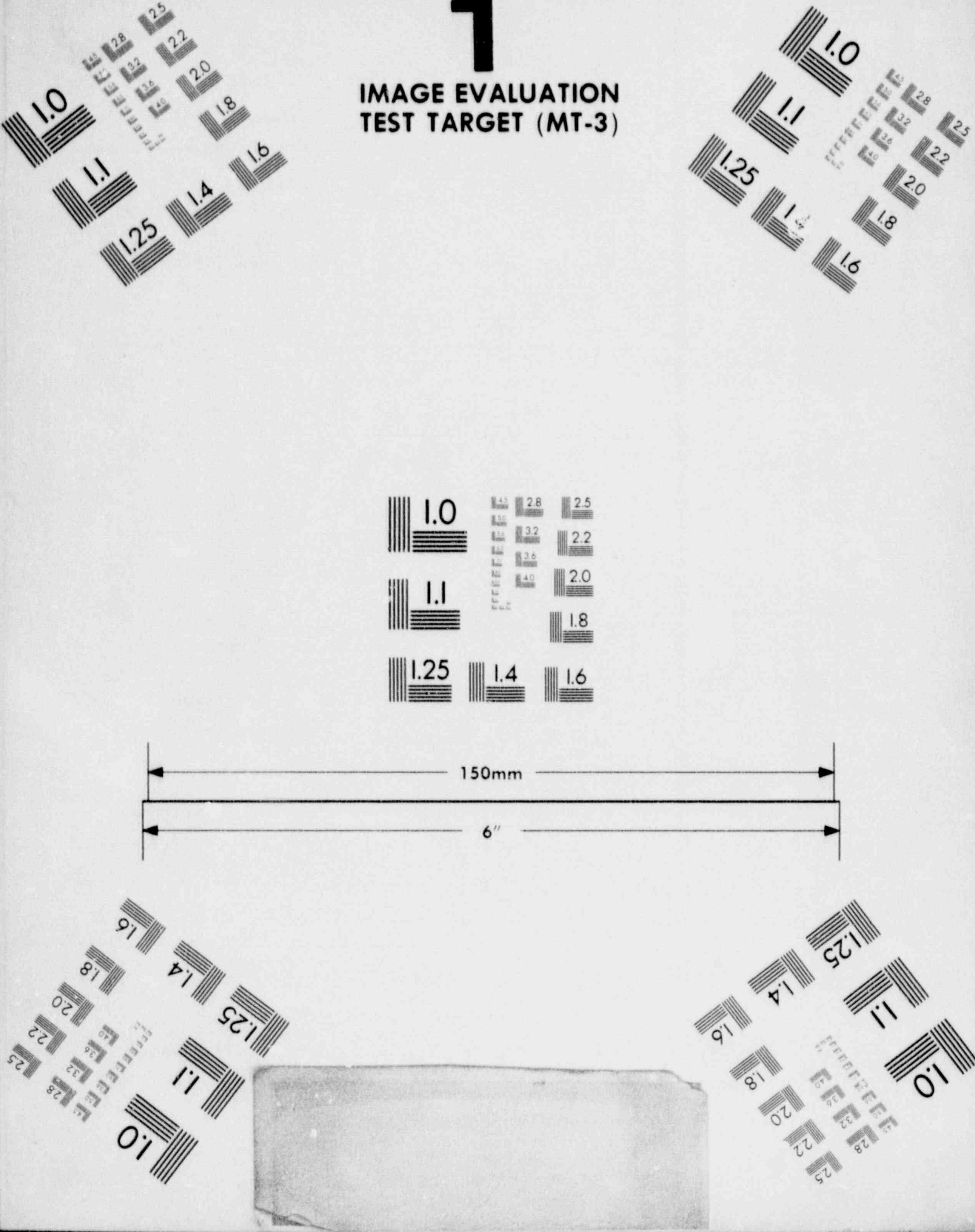
IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)



Paragraph No.	Assumption/Groundrule	Rationale	Rev.
3.3.2.2	**ALWR Seismic Hazard Input**	**ALWR Seismic Hazard Input**	0
	To be completed by 8/15/89.		0

Paragraph No.	*Assumption/Groundrule*	*Rationale*	*Rev.*
3.3.2.3	**Uncertainty Treatment**	**Uncertainty Treatment**	0
	To be completed by 8/15/89.		0

APPENDIX A: PRA KEY ASSUMPTIONS AND GROUNDRULES

Paragraph No.	*Assumption/Groundrule*	*Rationale*	*Rev.*
3.3.2.4	**Ground Response Spectrum**	**Ground Response Spectrum**	0
	To be completed by 8/15/89.		0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
3.3.2.5	**Hazard/Fragility Interface**	**Hazard/Fragility Interface**	0
	To be completed by 8/15/89.		0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
3.3.2.6	**Fragility Analysis**	**Fragility Analysis**	0
	To be completed by 8/15/89.		0

APPENDIX A: PRA KEY ASSUMPTIONS AND GROUNDRULES

Paragraph No.	*Assumption/Groundrule*	*Rationale*	*Rev.*
3.3.2.7	**Systems Analysis**	**Systems Analysis**	0
	To be completed by 8/15/89.		0

APPENDIX A: PRA KEY ASSUMPTIONS AND GROUNDRULES

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
4	**CONTAINMENT ANALYSIS**	**CONTAINMENT ANALYSIS**	0
4.1	**CORE DAMAGE SEQUENCE BINNING**	**CORE DAMAGE SEQUENCE BINNING**	0
	Core damage sequences are expected to be binned (grouped). If core damage bins are used, they shall be defined such that all sequences within a particular bin lead to similar effects with respect to containment sequence and source term phenomena. The definition of the bins shall 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.	Binning of similar sequences provides a means of managing the number of accident sequences. In addition, it provides a means of gaining information needed for the in-plant analysis task.	0
4.2	**CONTAINMENT SYSTEM ANALYSIS**	**CONTAINMENT SYSTEM ANALYSIS**	0
4.2.1	A containment systems analysis shall be developed such that it will explicitly account for any common failures between the core damage prevention systems and the containment systems.	Conditional probability of failure of containment systems must be determined by correctly accounting for dependencies between "upstream" events in the core damage sequence (such as support system failures) and the causes of failure of the containment systems.	0
4.2.2	If binning of accidents, including the status of containment systems, is used prior to the in-plant analysis, the frequency dominant accident sequence for each plant damage state shall be used to define in-plant phenomenological analysis parameters for use in determining containment performance source terms.	This simplifying assumption is made in order to reduce the number of deterministic analysis runs necessary to develop the containment event tree branch point probabilities. It is important to note 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.	0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
4.3	**CONTAINMENT ISOLATION**	**CONTAINMENT ISOLATION**	0
4.3.1	Containment penetrations shall be accounted for in the evaluation of containment leakage paths.	The potential for releases to occur due to failure of some penetrations to be isolated or properly sealed has been found to be important in previous studies. In particular, large leakage paths may be available. For example, equipment hatches that are left open may result in a large leakage path.	0
4.3.2	Containment penetrations can be screened from the analysis if they can meet one of the following criteria:	Not all containment penetrations have the potential to be important pathways for releases from containment. In order to focus the PRA effort on the penetrations that are most likely to be important, screening criteria may be applied.	0
	• Conditional probability of failure is small (i.e., less than about 10^{-3}/event);	• Failure of penetrations at a frequency of less than 1.0×10^{-3} are not expected to significantly contribute to risk and are excluded from the analysis.	0
	• Low consequence (e.g., release that must take place through a line that will remain filled with water throughout the accident);	• The consequences resulting from a release through water are not significant.	0
	• Closed loop;	• Any system which starts and terminates in the containment without any release path to the environment can be excluded from the containment penetration model, provided that its design against external event hazards is adequate.	0
	• Small in size (e.g., instrumentation lines).	• Small lines typically tend to become plugged quickly and are generally not important potential release pathways.	0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
4.4	**CONTAINMENT BYPASS**	**CONTAINMENT BYPASS**	0
	Containment bypass sequences shall be assessed and shall include all connections to the reactor coolant system.	Containment bypass sequences can result in significant releases from containment and have the potential to be important risk contributors. Past PRAs have identified the following bypass sequences as important:	0
		• Steam generator tube rupture (PWR only);	0
		• Residual heat removal isolation failure;	0
		• High-pressure coolant injection (BWR only);	0
		• Core spray (BWR only);	0
		• Feedwater and main steam (BWR only).	0
4.5	**IN-PLANT SEQUENCE ASSESSMENT**	**IN-PLANT SEQUENCE ASSESSMENT**	0
4.5.1	The containment ultimate strength calculation shall be made using the method discussed in Chapter 5, Section 6.6.2.2, of the Requirements Document. Calculation of containment capability shall consider the phenomena identified in Section 6.6.2.3.	The evaluation of containment ultimate strength shall include all features necessary to maintain containment integrity, including the containment shell, hatches, personnel locks, seals, penetrations, and valves. The phenomena to be considered include the potential for bypass of the suppression pool (BWR), effects of direct contact of core debris, and consideration of dynamic loading of the containment during containment-flooding scenarios.	0
4.5.2	The MAAP code shall be the primary tool used to assess thermal-hydraulic and other physical processes and phenomena such as core heat-up, containment loading, release of radionuclides, and combustible gas generation and ignition for use in establishing accident progression. Other computer codes and analysis methods may be used to supplement the MAAP code, or may be used in place of the MAAP code with appropriate justification.	In order to adequately model the processes involved, an integrated model of the core melt and containment is required to address generation, effects of steam inerting, containment geometry, and containment pressurization.	0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
4.6	**CONTAINMENT EVENT ANALYSIS**	**CONTAINMENT EVENT ANALYSIS**	0
4.6.1	A containment event tree shall comprise the important phenomenological issues associated with containment loading and/or source term evolution.	A containment event tree provides an excellent means to identify and quantify important phenomena. Elements which have been addressed in past large, dry containment PWR and in BWR PRAs and which should be considered for the ALWR in the development of the containment event tree include:	0
		• Potential for early and late hydrogen burns;	0
		• Pressure and temperature loadings on the cavity/drywell following reactor vessel failure;	0
		• Containment loadings due to noncondensible gas generation and gas generation during corium-concrete interaction;	0
		• Potential for direct interaction between corium and containment;	0
		• Availability of containment scrubbing, pool scrubbing, and containment/pool heat removal;	0
		• Venting availability;	0
		• Standby gas treatment system operability (BWR);	0
		• Fire suppression system operability (BWR);	0
		• Containment inertability (BWR);	0
		• Ability to flood and replenish the cavity/drywell region of the containment;	0
		• Hydrogen generation rates and core blockage model;	0
		• Adiabatic burn temperature;	0

APPENDIX A: PRA KEY ASSUMPTIONS AND GROUNDRULES

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
		• Debris coolability (amount of water required);	0
		• Location and size of containment break;	0
		• Size and timing of containment failure prior to RPV melt through;	0
		• Hydrogen concentration in secondary building (BWR);	0
		• Suppression pool scrubbing (BWR);	0
		• Operation of the standby gas treatment system (BWR);	0
		• Revaporization and composition of Iodine (CSOH and CSI);	0
		• Variation of iodine compounds.	0
4.6.2	Potentially important phenomena which are not currently addressed in the MAAP code shall also be considered.	Some phenomena that have been found to be important in previous risk analyses are not currently explicitly treated using MAAP. These phenomena include the following:	0
		• Direct containment heating;	0
		• Steam explosions;	0
		• Hydrogen deflagration due to equipment or operator failures;	0
		• Failure of vapor suppression (BWR).	0
4.6.3	The quantification of containment event trees shall be performed using best estimate values.	This is consistent with the guidance provided for the core damage assessment.	0
4.6.4	The basis and supporting information used to determine containment event tree probability shall be thoroughly documented.	In order to ensure traceability of this process, it should be well documented.	0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
4.7	**SOURCE TERM DEFINITION**	**SOURCE TERM DEFINITION**	0
	The most current version of MAPP shall be used for source term calculations. Alternative codes may be used if justification is provided.	In order to adequately model the processes involved, an integrated model of the core melt and containment is required to address generation, effects of steam inerting, containment geometry, and containment pressurization.	0
4.8	**PLANT RELEASE CATEGORIES**	**PLANT RELEASE CATEGORIES**	0
	Similar end points of the containment analysis may be grouped into release categories for use in the ex-plant consequence analysis.	Past PRAs have shown that containment event tree end points may be grouped to simplify the analysis. This reduces the number of ex-plant runs required. Elements to be considered during the grouping process include:	0
		• Time of release;	0
		• Duration of release;	0
		• Energy of release;	0
		• Types and amounts of isotope fractions released.	0

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
5	**OFF-SITE CONSEQUENCES**	**OFF-SITE CONSEQUENCES**	0
5.1	**IMPLEMENTATION OF THE PUBLIC-SAFETY REQUIREMENT**	**IMPLEMENTATION OF THE PUBLIC-SAFETY REQUIREMENT**	0
	A mean complementary cumulative distribution function (CCDF) for whole-body dose shall be developed for a half-mile radius. This shall include all core-damage sequences with a mean frequency greater than 10^{-8} yr^{-1} from both internal and external initiators. The design shall be considered to have met the risk requirement if this CCDF falls outside the region bounded by a lower limit for frequency at 1×10^{-6}/year and by a lower limit for consequences of 25 rem whole-body dose at one-half mile, as shown in Figure A.5-1.	The CCDF is a well accepted method of visually displaying risk curves. A composite CCDF, including the contributions from all release categories, will be developed based on best-estimate source terms and will include all core-damage sequences with frequencies greater than 10^{-8} yr^{-1}. This will provide a visual display which shows that the ALWR meets the off-site consequence risk requirement. The mean curve is the curve used for this demonstration, consistent with the Rationale described in Section 1 above.	0
5.2	**METHOD FOR OFF-SITE CONSEQUENCE ANALYSIS**	**METHOD FOR OFF-SITE CONSEQUENCE ANALYSIS**	0
5.2.1	A "reference site" with the characteristics listed in Annex B shall be used for calculating off-site consequences for the ALWR.	The primary purpose of the PRA is to assess the plant design, and use of a reference site permits determination of whether the design should be adequate, irrespective of the site at which it may be located. Moreover, it is anticipated that this PRA will be performed at the time of design certification in the licensing process. Hence, an actual site will not yet be identified, and a reference site is therefore specified. This "reference site" represents the consequences of most potential sites. Factors which affect consequences include: (1) climatography, (2) demography, (3) topography, and (4) evacuation and sheltering. Characteristics of 91 U.S. reactor sites are tabulated in the NRC document, NUREG/CR-2239 (Ref. 36). Based upon the data presented in NUREG/CR-2239, the "reference site," as modified, is estimated to equal the 80th percentile or above for those characteristics which are correlated to high off-site consequences.	0

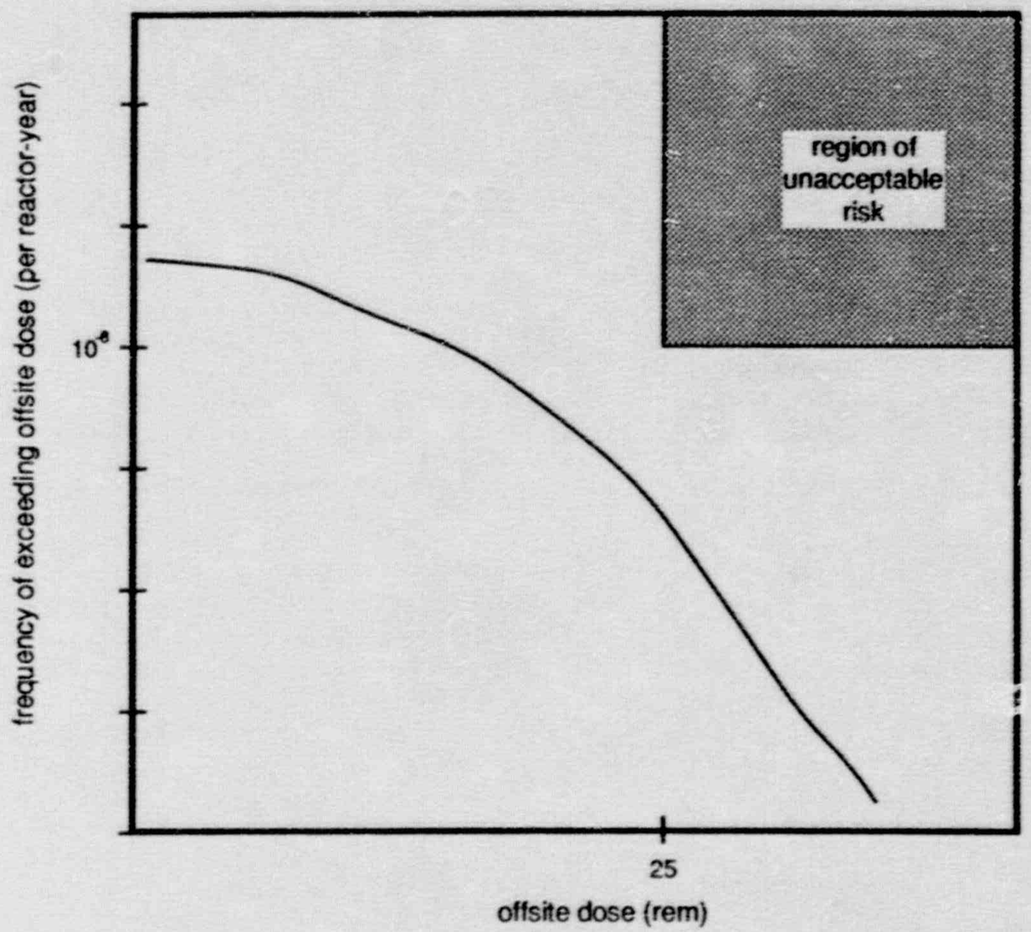


Figure A.5-1. Illustration of Application of Risk Criterion for Offsite Exposure

APPENDIX A: PRA KEY ASSUMPTIONS AND GROUNDRULES

Paragraph No.	Assumption/Groundrule	Rationale	Rev.
5.2.2	The off-site consequences calculations shall be performed using either CRAC2 (Ref. 37) or MACCS (Ref. 38). The CRAC2 input file, ALWR Reference Site, shall be used for this purpose (Reference 40).	The computer code CRAC2 is the best tool presently available for performing off-site consequence calculations. It has been shown through benchmark studies to give acceptable results when compared with other consequence codes. The application of the CRAC2 input file, ALWR Reference Site, provides a basis for consistency among the users of the code.	0
5.2.3	It will be assumed that there will be no evacuation for 24 hours following the release. Cloud and ground shielding factors for normal activity should be used. These assumptions are **only** for the purposes of comparison against the requirement stated in 5.1 above. For estimation of public health risk, realistic estimates for these parameters shall be used.	Calculating 24 hours of exposure with no emergency response provides a check against the requirement stated in Section 5.1, above, independent of future emergency planning requirements.	0

Reference No.	Title	Rev.
6	**REFERENCES**	0

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ANNEX A
RELIABILITY DATA BASE FOR ALWR PRAS

This annex describes the development of the initiating-event frequencies and component reliability data that are summarized in Section 2.11. Section A1 outlines the methods used in obtaining initiating-event frequencies for loss-of-coolant accidents (LOCAs) and for most transient events. The treatment of the frequency and recovery of losses of off-site power is described in somewhat more detail in Section A2. Section A3 summarizes the sources of data used to arrive at the recommended hardware failure rates, maintenance unavailabilities, and common-cause factors.

A1 FREQUENCY OF INITIATING EVENTS

The selection of initiating events to be subjected to detailed analysis is one of the key tasks of the PRA effort. Clearly, it is not possible to obtain a set of initiator frequencies without first establishing the events to be evaluated. The development of frequencies in this annex corresponds to the set of initiating events derived for preliminary PRAs of an advanced BWR and an advanced PWR (Refs. 1 and 2), which were based almost entirely on the Requirements Document, Chapters 1 through 5. It is expected that the actual PRAs will each define a set of initiating events that represents a different and more detialed breakdown than that obtained in these preliminary PRAs. Consequently, the frequencies presented here and in Section 2.11.2 will require revision. However, this assessment provides some guidance with respect to the reasonable frequencies to be used and methods to use in developing them. The events from the preliminary PRAs are listed in Table A1-1 (the designators are provided solely for ease of reference).

Table A1-1
INITIATORS FOR WHICH FREQUENCIES ARE SUGGESTED

BWR		PWR	
Designator	**Event**	**Designator**	**Event**
T_1	Reactor/turbine trip	T_1	Reactor/turbine trip
T_2	Loss of condenser	T_2	Loss of main feedwater
T_3	Loss of feedwater	T_3	Loss of offsite power
T_4	Loss of offsite power	T_4	Steam-line break
T_5	Loss of major ac power bus	T_5	Loss of major ac power bus
A	Large loss-of-coolant accident	A	Large loss-of-coolant accident
S	Small loss-of-coolant accident	S_1	Intermediate loss-of-coolant accident
		S_2	Small loss-of-coolant accident
		R	Steam-generator tube rupture

ANNEX A

RELIABILITY DATA BASE FOR ALWR PRAS

The design requirements for ALWRs incorporate a number of features aimed at reducing the frequencies of plant transients in order to provide further improvements in both safety and plant availability. The overall design requirements include a limit of 1.0/year for the frequency of plant trips, and this level appears to be attainable, based on recent experience for some U.S. and foreign plants. Therefore, although the initiating-event frequencies must be estimated based on the operating experience for current-generation plants, it was deemed appropriate to account in some manner for the improvements required for ALWRs. Two measures were taken to achieve this objective for the more frequently-occurring transient events:

- Only recent operating experience (i.e., 1984 through August 1988) was used, to reflect the increased reliability that many current plants appear to be exhibiting relative to earlier years; and

- The specific events in the base of operating experience were reviewed to determine applicability to the ALWRs. Events that should be precluded for the ALWRs based on the design requirements were deleted.

It should be noted that the second measure requires that the analyst exercise particular caution so that no events that could be representative of initiating failures for the ALWRs are deleted. Furthermore, the potential exists that the designs will introduce the possibility of new initiating events, especially during the early years of operation, that would not have been experienced in current plants. Nevertheless, provided that care is taken, this appears to be an appropriate approach in order to provide the most realistic assessment for ALWRs.

The first step was to map the transient initiators into the categories of events provided in NUREG/CR-3862 (Ref. 3), which is an update of a data base originally developed for EPRI. A number of the requirements are aimed at reducing the potential for some types of transients, and it was therefore judged desirable to eliminate such events from the data base reflecting past experience. The corresponding trip categories deleted for the ALWRs are presented in Table A1-2. For those that are deleted due to design requirements, references to the appropriate requirement are provided in brackets.

ANNEX A

RELIABILITY DATA BASE FOR ALWR PRAS

Table A1-2
INITIATOR CATEGORIES DELETED FROM CONSIDERATION FOR THE ALWR

	EPRI Category	Reason for Deletion
BWR		
6.	Closure of one main-steam isolation valve	Reactor would not necessarily trip on closure of one valve [3.5.4.A.2].
16.	Trip of one recirculation pump	Plant must be designed not to trip for this event [3.5.3.D.4].
21.	Loss of a feedwater heater	Loss of a single train must not cause a trip [2.4.2.A.4].
23.	Trip of a feed or condensate pump	Loss of a single train will not cause a trip [2.4.2.4.4].
25.	Feed increasing flow during startup	New electric-driven feed pumps should eliminate
26.	Feed decreasing flow during startup	New electric-driven feed pumps would likely eliminate this as a trip concern.
28.	Rod withdrawal at startup	Event has limited impact and frequency low enough that there are no occurrences.
36.	Manual scram	This category includes many non-significant transients, such as test of the scram system when lowering power for a scheduled outage.
PWR		
23.	Loss of condensate pumps (one loop)	Plant must be designed not to trip for this event [3.4.2.A.4].
36.	Manual scram	This category includes many non-significant transients, such as test of the scram system when lowering power for a scheduled outage.
41.	Fire within plant	Fires will be considered separately, as external initiating events.

ANNEX A

RELIABILITY DATA BASE FOR ALWR PRAS

The data base developed by the Institute for Nuclear Power Operations (INPO) for reactor trips was then reviewed to determine the number of events that have occurred for the remaining categories. The data base covers the period from 1984 to the present (i.e., through August, 1988), and was judged to provide the most up-to-date and representative summary of current operating experience for the more frequent types of tansients. The frequencies (per reactor-year) are as follows:

	Event	Frequency (/reactor-year)
BWR		
T_1	Turbine trip	2.3
T_2	Loss of main condenser	0.49
T_3	Loss of feedwater	0.37
PWR		
T_1	Reactor/turbine trip	2.8
T_2	Loss of main feedwater	0.46

For other, less frequent initiators, the INPO data base was judged not to cover a sufficient period of operating experience to provide an adequate basis for quantification. For both the advanced PWR and the advanced BWR, the frequency of the loss of normal off-site power is estimated in the next section to be 0.035/yr. For the loss of a major ac power bus (event T_5 for both plants), the frequency is estimated based on extending the hourly failure rate for such a bus over a year (accounting for the capacity factor of 87% for the unit):

$$\phi(T_5) = (2.0 \times 10^{-7}/hr)(8760 \frac{hr}{yr})(0.87)$$

$$= \quad 1.5 \times 10^{-3}/yr$$

For LOCAs and the steam-generator tube rupture, the frequencies were estimated based on available information. Although there have been no pipe ruptures that have constituted LOCAs in either PWRs or BWRs, there have been some operational events that are similar in nature to small LOCAs. The evidence used to characterize the frequencies of these events was as follows:

BWR: no large LOCAs
2 equivalent small LOCAs
390 plant-years of relevant experience

PWR: no large or intermediate LOCAs
2 equivalent small LOCAs
3 steam-generator tube ruptures
1 steamline break
660 plant-years of relevant experience

For the cases of the large (and intermediate) LOCAs, the frequencies were estimated based on the χ^2 variate at the 50% cumulative probability level, using the following expression:

$$\phi(A) = \frac{\chi^2_{50}(2n+1)}{2T}$$

The frequencies of the LOCA initiators were therefore calculated as follows:

BWR

$$\phi(A) = \frac{0.455}{2 \cdot 390} = 5.8 \times 10^{-4}/yr \qquad \phi(S) = \frac{2}{390} = 5.1 \times 10^{-3}/yr$$

PWR

$$\phi(A) = \phi(S_1) = \frac{0.455}{2 \cdot 660} = 3.4 \times 10^{-4}/yr \qquad \phi(S_2) = \frac{2}{660} = 3.0 \times 10^{-3}/yr$$

$$\phi(R) = \frac{3}{660} = 4.5 \times 10^{-3}/yr \qquad \phi(T_4) = \frac{1}{660} = 1.5 \times 10^{-3}/yr$$

The results for all initiating events for both plants are summarized in Table A1-3. It should be noted that the ALWR requirements specify that the design result in a total frequency of reactor trips of not more than 1.0 per year, and that the frequencies presented in Table A1-3 exceed that figure for both types of plant. The nature of this particular design requirement is such that it will not be possible to demonstrate conclusively that it has been met in the absence of actual operating experience. The reliance on recent experience of current generation plants, with trip frequencies reduced to reflect specific design requirements and other considerations, is considered to be the most appropriate approach to the development of initiator frequencies for the PRAs for the ALWRs.

ANNEX A
RELIABILITY DATA BASE FOR ALWR PRAS

Table A1-3
SUGGESTED INITIATOR FREQUENCIES FOR ALWRS

Event	Description	Suggested Frequency
BWR		
T_1	Turbine trip	2.3
T_2	Loss of main condenser	0.49
T_3	Loss of feedwater	0.37
T_4	Loss of normal offsite power*	0.035
T_5	Loss of a major ac power bus	1.5×10^{-3}
A	Large loss-of-coolant accident	5.8×10^{-4}
S	Small loss-of-coolant accident	5.1×10^{-3}
PWR		
T_1	Reactor/turbine trip	2.8
T_2	Loss of main feedwater	0.46
T_3	Loss of normal offsite power*	0.035
T_4	Steamline break	1.5×10^{-3}
T_5	Loss of a major ac power bus	1.5×10^{-3}
A	Large loss-of-coolant accident	3.4×10^{-4}
S_1	Intermediate loss-of-coolant accident	3.4×10^{-4}
S_2	Small loss-of-coolant accident	3.0×10^{-3}
R	Steam-generator tube rupture	6.1×10^{-3}

* For total loss of off-site power, the conditional unavailability of the reserve supply (0.22) must also be multiplied by this value. In addition, for the advanced PWR the frequency of demand for emergency power must also reflect the conditional unavailability of the full-load rejection capability.

A2 Loss of Off-site Power

Because of the potential importance of sequences involving failures of off-site and on-site ac power, it was considered desirable to examine the available sources of information to obtain the most appropriate characterization of the frequency of losses of off-site power as initiating events, as well as the conditional probability of restoring off-site power as a function of time following the event. NSAC/144 (Ref. 4) contains an excellent summary of all of the partial and complete losses of off-site power that have occurred at nuclear power plants through 1988, and is the most up-to date source of information available in this area. However, the treatment of some events required some modification in order to ensure that the data are applied in a manner consistent with the nature of the models in a PRA for the ALWR designs.

Chapter 11 of the Requirements Document provides requirements for the arrangement of off-site power supplies that go beyond the features generally found in current-generation plants. Among those features likely to be most important for the PRAs for ALWRs are the following:

- The use of a generator-output breaker is specified so that, upon tripping of the main turbine-generator, off-site power is continuously supplied from the main switchyard via the auxiliary transformers, with no switching required.

- A reserve transformer must be provided that is fed from a separate substation that is, to the extent practical, independent of the portion of the grid feeding the main switchyard. If possible, the feed to the reserve transformer is to be underground, providing further protection against severe-weather phenomena. The reserve transformer would normally be in a standby mode and, upon deenergization of the buses, would pick up the loads before a signal is generated to start the emergency diesel generators.

- For the advanced PWR, a full-load rejection capability is required. Therefore, upon loss of the normal off-site power supply, the reactor and main turbine-generator should run back to a nominal power level sufficient to continue to supply the plant auxiliary loads. For the advanced BWR, the ability to sustain operation following a loss of load up to 40% of full power is specified.

These features combine to present an arrangement that is potentially much more reliable than might be reflected in a generic assessment of operating experience for current-generation plants. For current plants, it is required that two different off-site supplies to plant loads be provided. However, this requirement is met in many different ways by different plants. For example, some plants have two different supplies from the same main switchyard. Others have transformers fed from two different switchyards on-site, but with substantially less independence between the switchyards than is called for for the ALWRs. Only one existing plant has a full-load rejection capability that has been successfully used. For some plants, the auxiliary transformer is deenergized upon a plant trip, and switching to an alternative transformer is required. Still other plants normally use the startup transformer to supply some or all plant loads during normal operation. While this reduces the potential for a loss of power following a plant trip, it also limits the ability to use the main switchyard for auxiliary loads in the event that the switchyard feeding the startup transformer is lost.

Therefore, it was necessary to examine the events in NSAC/144 in more detail in order to assess their relationships to the features required for the ALWRs. The first step was to reclassify the events according to the following factors:

- Whether or not the event corresponded to a loss of the normal off-site supply for an ALWR;
- Whether or not a supply at least roughly analogous to the reserve transformer was provided, and whether or not the event constituted a loss of this equivalent reserve supply alone or in addition to the loss of normal power; and
- Whether or not the event itself could have precluded the use of full-load rejection, if it had been provided (e.g., due to a failure in the step-up or auxiliary transformer).

In general, the switchyard connected to the main generator was considered analogous to the main switchyard for the ALWR, and if a supply was also provided from a separate (although not necessarily independent) switchyard, it was considered to be analogous to the ALWR's reserve transformer.

ANNEX A
RELIABILITY DATA BASE FOR ALWR PRAS

Only the experience for the ten years, 1978 through 1988, was examined. This was done primarily to reflect improvements in off site power reliability that have been exhibited by current plants in recent years, as a result of upgrading switchyards and off-site grids. The reclassification of the events is provided as Table A2-1. The table also includes the original NSAC/144 classifications for reference purposes. These categories are as follows:

1. No off-site power available and unit trip;

2. Loss of backup off-site power, but if on line, the unit remained connected to the normal off-site system and the plant received auxiliary power from the unit transformer or its equivalent; and

3. Loss of normal off-site power but backup off-site power available.

It should be noted that the experience for two plants was deleted from the data base. The two losses of off-site power that have occurred at Palo Verde were determined to be due to a unique arrangement and plant-specific switching considerations, and were judged not to apply directly to the consideration of loss of off-site power for the ALWR. Therefore, both the events and the operating years were removed from the data base. The other plant that was not included was Turkey Point. Turkey Point had previously experienced a number of unique problems with off-site power, but has taken substantial steps to resolve them. The limited experience since these steps were taken indicates that they appear to have been successful. Therefore, it was judged that the plant was not representative for the ALWR, and the corresponding experience for Turkey Point was also removed.

The relevant results of the data review are as follows:

- In approximately 630 site-years, there have been 22 events corresponding to loss of normal off-site power, for an annual frequency of 0.035.

- Of the 22 events, the failures that occurred would have precluded the use of full-load rejection in 5 cases; this results in a contribution to the conditional unavailability of full-load rejection of 0.23 (which does not include the probability that full-load rejection itself would not function when demanded).

- Of the 22 events, 18 occurred at sites at which there was a source roughly analogous to the reserve transformer for the ALWRs. Of these 18, there were 4 events in which the reserve feed was also unavailable. This yields a contribution to the conditional unavailability of the reserve source of 0.22 (which does not include unavailability of the transformer itself or failure of breakers, etc.).

Table A2-1
SUMMARY OF EVENTS INVOLVING LOSSES OF OFF-SITE POWER*

Plant	*Date*	*NSAC category*	*Loss of normal off-site power?*	*Full-load rejection precluded?*	*Site has reserve transformer?*	*Loss of reserve power?*	*Duration*
Arkansas Nuclear One	4/7/80	3	yes	no	yes	no	0:22
	6/24/80	3	yes	no	yes	no	unknown
Browns Ferry	3/1/80	-	no	–	yes	yes	unknown
Calvert Cliffs	7/23/87	1	yes	no	yes	yes	1:58
Connecticut Yankee	8/1/84	1	no	–	yes	no	–
Cook	2/1/86	3	yes	yes	yes	no	unknown
Cooper	1/29/84	3	yes	yes	yes	no	1:49
Crystal River-3	6/16/81	3	no	–	yes	yes	unknown
	2/28/84	3	no	–	yes	yes	0:00:05
Davis-Besse	10/15/79	1	yes	yes	no	–	0:26
Diablo Canyon	7/17/88	1	no	–	yes	yes	0:38

* This summary is based on information provided in NSAC/144 for events i the 10 years, 1979 through 1988.

Table A2-1
SUMMARY OF EVENTS INVOLVING LOSSES OF OFF-SITE POWER*

Plant	*Date*	*NSAC category*	*Loss of normal off-site power?*	*Full-load rejection precluded?*	*Site has reserve transformer?*	*Loss of reserve power?*	*Duration*
Dresden	8/16/85	1	no	–	yes	no	–
Farley	10/8/83	1	no	–	yes	yes	2:45
Ft. St. Vrain	5/17/83	1	yes	no	no	–	1:45
Ginna	4/18/81	2	no	–	yes	yes	unknown
Indian Point-2	6/3/80	1	no	–	yes	yes	1:45
Indian Point-2	10/4/83	3	no	–	yes	yes	0:15
Indian Point-3	7/12/84	3	yes	yes	yes	no	unknown
Maine Yankee	4/25/83	2	no	–	yes	yes	2:45
	7/2/83	2	no	–	yes	yes	0:04
McGuire	8/21/84	1	yes	no	no	–	0:20
Millstone	9/27/85	1	yes	no	yes	yes	3:31
Monticello	4/27/81	1	no	–	yes	no	–

Table A2-1
SUMMARY OF EVENTS INVOLVING LOSSES OF OFF-SITE POWER*

Plant	*Date*	*NSAC category*	*Loss of normal off-site power?*	*Full-load rejection precluded?*	*Site has reserve transformer?*	*Loss of reserve power?*	*Duration*
Nine Mile Point	2/7/82	2	no	–	yes	yes	0:00:10
	12/26/88	1	no	–	yes	yes	0:00:00
Oyster Creek	11/14/83	2	no	–	yes	yes	4:00
Palisades	7/14/87	1	no	–	yes	yes	7:26
Pilgrim	7/27/79	3	yes	no	yes	no	0:14
	8/28/79	3	yes	no	yes	no	unknown
	10/12/82	3	yes	no	yes	no	11:33
	2/13/83	3	yes	no	yes	no	unknown
	11/19/86	3	yes	no	yes	no	3:14
	12/23/86	3	yes	no	yes	no	0:27
	11/12/87	1	yes	no	yes	yes	11:00
Prairie Island	7/15/80	1	yes	yes	yes	yes	1:02
Quad Cities	6/22/82	3	no	–	yes	no	–
River Bend	1/1/86	1	yes	no	no	–	0:46
Robinson	1/28/86	1	no	–	yes	yes	1:40

Table A2-1
SUMMARY OF EVENTS INVOLVING LOSSES OF OFF-SITE POWER*

Plant	*Date*	*NSAC category*	*Loss of normal off-site power?*	*Full-load rejection precluded?*	*Site has reserve transformer?*	*Loss of reserve power?*	*Duration*
San Onofre	11/22/80	1	yes	no	yes	no	0:00:15
	11/21/85	1	no	–	yes	yes	0:04
Susquehanna	7/15/84	3	yes	no	yes	no	unknown
	7/26/84	1	–	–	(intentional test)	–	–
WNP-2	1/31/85	3	yes	no	yes	no	unknown
Totals			22	5	–	20	–

There were an additional 16 events that involved loss of only the reserve feed, and this information could be used to estimate an additional unavailability contribution. However, because the average duration of these outages is less than 2 hours, this contribution is very small compared to the likelihood of failure in common with the normal supply.

Another point worth noting is the potential that the failure mode might be of such a nature that it could affect both the normal and reserve feeds, as well as preclude use of the full-load rejection capability. Such an event might be postulated, for example, due to the propagation of some bus fault that did not clear before the reserve source attempted to close in. In the data base, there was one event that involved failure of both sources and that would have precluded the use of full-load rejection. The conditional unavailability of both full-load rejection and the reserve source based on this limited data would be 0.056. This compares very favorably to the combined conditional probabilities obtained when treating the full-load rejection and reserve source as independent ($0.22 \times 0.23 = 0.051$). This provides some level of check on this aspect of the data treatment.

Finally, the times reported for initial recovery of off-site power were evaluated to derive a distribution of non-recovery probability as a function of time. In examining the recovery times, it was noted that, for the four events involving a loss of both the normal and reserve supply, three involved severe-weather phenomena away from the site (i.e., hurricane, tornado, etc.). Furthermore, the recovery times for these four events were all at or above the average recovery time for all events considered together. Therefore, the question of what data constituted an appropriate set to use for analysis of recovery of a total loss of off-site power arose. It was concluded that the four data points alone were not sufficient to support a recovery-time distribution. The use of only the recovery times for events involving severe weather was also considered. However, that distribution is strongly affected by two long events (both of which occurred at Pilgrim), neither of which involved a loss of the reserve source. In addition, the requirements for the reserve feed should tend to reduce the effects of severe-weather events somewhat, although it is difficult to characterize the degree to which this will be realized. Finally, the recovery-time distribution for all events and that for only weather-related events are relatively close to each other in probability (within a factor of two). Therefore, it was decided to develop a single recovery-time distribution to be used for all losses of off-site power.

The resulting distribution is provided as Table A2-2. Entries for times beyond 12 hours are taken from a curve fit based on a gamma distribution, which has previously been shown to provide a relatively good fit to these data (Ref. 5).

It should be pointed out that this data treatment is useful only for considering events initiated from power operation. During cold shutdown, and especially during extended refueling outages, less stringent restrictions regarding the outages of transformers and other key equipment typically apply; this could correspond to increased frequency of total losses of off-site power and/or longer durations of the outages.

ANNEX A

RELIABILITY DATA BASE FOR ALWR PRAS

Two other points are also important. First, this data treatment may be somewhat conservative, in the sense that the degree of independence for the off-site sources for the ALWRs is greater than that generally found for current sites. Furthermore, it is reasonable to assume that an actual advanced reactor will employ grid connections comparable to the better and more recent of the current-generation plants. Therefore, overall, this treatment of the available data is considered to be appropriate.

A3 COMPONENT FAILURE DATA

As a result of the desire to recommend a consistent set of reliability data to be used in the ALWR PRAs, several data sources were reviewed, and a representative set of failure rates was compiled. For each component type and failure mode, the failure rates were extracted from the available sources, and a suitable value was selected based on judgment regarding applicability to the anticipated ALWR designs. The primary sources of generic data examined included the following:

- The Oconee PRA (Ref. 6), whose generic data base represents the synthesis of data from a variety of generic sources;

- The Seabrook Probabilistic Safety Study (PSS) (Ref. 7), which reflects both earlier generic sources such as those that led to the Oconee PRA data base, and detailed data from a number of individual plants;

- Data estimated from licensee-event reports, and reported in NUREG/CR-1363 for valves (Ref. 8), NUREG/CR-1205 for pumps (Ref. 9), and NUREG/CR-1362 for diesel generators (Ref. 10);

- Additional data compiled for diesel generators and reported in NUREG/CR-2989 (Ref. 11);

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RELIABILITY DATA BASE FOR ALWR PRAS

Table A2-2
CUMULATIVE NON-RECOVERY PROBABILITIES

Time (hr)	Probability of not recovering power	Time (hr)	Probability of not recovering power
0.5	0.61	13	0.013
1	0.54	14	9.1×10^{-3}
2	0.32	15	6.1×10^{-3}
3	0.25	16	4.1×10^{-3}
4	0.18	17	2.7×10^{-3}
5	0.14	18	1.8×10^{-3}
6	0.14	19	1.2×10^{-3}
7	0.14	20	7.5×10^{-4}
8	0.11	21	4.8×10^{-4}
9	0.11	22	3.1×10^{-4}
10	0.11	23	1.9×10^{-4}
11	0.071	24	1.3×10^{-4}
12	0.019		

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RELIABILITY DATA BASE FOR ALWR PRAS

- The data for diesel generators reported in NSAC/108 (Ref. 12);
- The data base compiled for the Accident Sequence Evaluation Program (Ref. 13), which is based largely on data from the Reactor Safety Study (Ref. 14);
- The data provided for the Northeast Utilities system, as reported in the draft version of the ALWR PRA Key Assumptions and Groundrules Document (Ref. 15);
- Military data for non-nuclear installations reported in NPRD-2 (Ref. 16);
- The data for some electrical components and instrumentation reported in IEEE-500 (Ref. 17);
- The Browns Ferry PRA (Ref. 18);
- The PSA Procedures Guide (Ref. 19);
- The elicitation of expert opinion obtained for NUREG-1150 (Ref. 20); and
- Data collected by Ontario Hydro for combustion turbine-generators (Ref. 21).

In addition, raw data were extracted from available sources for several specific plants. These sources included the following:

- The plant-specific experience summarized in the Oconee PRA (Ref. 6);
- The data reported for Indian Point Units 2 and 3 in the Indian Point PSS (Ref. 22);
- The operating experience for Zion reported in the Zion PSS (Ref. 23);
- Experience described for Millstone in a recent paper (Ref. 24);
- The experience for Browns Ferry reported in the Browns Ferry PRA (Ref. 18);
- The data compiled for a particular PWR for which a PRA is currently underway (designated as PWR X), and

- The evidence of relief-valve reliability for LaSalle provided to the Risk Methods Integration and Evaluation Program (Ref. 25).

It is recognized that there is overlap among some of these data sources, and that none (with the possible exception of NPRD-2) is completely independent of all of the others. An attempt was made to take these factors into account in selecting the recommended values. The values extracted from surveying these sources are tabulated in the forms provided at the end of this annex. The results are summarized in Table A3-1. For each component type and failure mode, a reference is provided to the entry in the survey sheets.

It was also judged to be desirable to provide suggested values to be used for maintenance unavailabilities. A limited survey was conducted of available PRAs, and maintenance unavailabilities were estimated on a train level for selected systems. In addition to the sources noted above for failure data, some maintenance unavailabilities for BWRs were extracted from the Shoreham PRA (Ref. 26). The maintenance unavailabilities are summarized in Table A3-2 for BWRs and Table A3-3 for PWRs.

A4 COMMON-CAUSE FACTORS

Common-cause factors were evaluated according to the procedures presented in the EPRI report NP-5613 (Ref. 27). This procedure involves reviewing specific events that have occurred to determine whether or not similar events could occur at the plant of interest. Common-cause factors are then estimated from the relative frequencies of multiple failures compared to overall failures, including independent faults. The events summarized in EPRI NP-3967 (Ref. 28) served as the input data base for the review. In this assessment, the multiple-Greek letter approach was utilized to obtain common-cause parameters for failure of component combinations of interest. The systems analyst must select the component groups to which the common-cause factors should be applied.

ANNEX A

RELIABILITY DATA BASE FOR ALWR PRAS

Table A3-1
COMPONENT FAILURE DATA

Component	Failure Mode	Failure Rate	Survey Entry
Motor-operated valve	Fails to operate on demand	4.0×10^{-3}/d	1
	Transfers closed	1.4×10^{-7}/hr	2
Air-operated valve	Fails to operate on demand	2.0×10^{-3}/d	3
	Transfers closed	1.5×10^{-7}/hr	4
Check valve (other than stop)	Fails to operate on demand	2.0×10^{-4}/d	5
	Transfers closed	2.0×10^{-7}/hr	6
	Reverse leakage (gross)	6.0×10^{-7}/hr	7
Stop-check valve	Fails to operate on demand	1.0×10^{-3}/d	8
	Transfers closed	2.0×10^{-7}/hr	9
	Reverse leakage (gross)	6.0×10^{-7}/hr	10
Check valve	Internal rupture	5.0×10^{-9}/hr	11
Manual valve	Plugs/transfers closed	3.7×10^{-8}/hr	12
Pressurizer safety valve (PWR)	Fails to open on demand	1.0×10^{-3}/d	13
	Fails to reclose	7.0×10^{-3}/d	14
Safety/relief valve (BWR)	Fails to open on demand	6.0×10^{-3}/d	15
	Fails to reclose	6.5×10^{-3}/d	16
Pilot-operated relief valve	Fails to open on demand	7.0×10^{-3}/d	17
	Fails to reclose	2.5×10^{-2}/d	18
Motor-driven pump (all types)	Fails to start on demand	2.0×10^{-3}/d	19
	Fails to run	2.5×10^{-5}/hr	20
Motor-driven pump (LPI/RHR)	Fails to start on demand	2.3×10^{-3}/d	21
	Fails to run	1.3×10^{-5}/hr	22
Motor-driven pump (safety inj.)	Fails to start on demand	1.0×10^{-3}/d	23
	Fails to run	5.0×10^{-5}/hr	24

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Table A3-1
COMPONENT FAILURE DATA

Component	Failure Mode	Failure Rate	Survey Entry
Motor-driven pump (emerg. feed)	Fails to start on demand	3.0×10^{-3}/d	25
	Fails to run	1.5×10^{-4}/hr	26
Motor-driven pump (service water)	Fails to start on demand	2.4×10^{-3}/d	27
	Fails to run	3.2×10^{-5}/hr	28
Motor-driven pump (comp. cooling)	Fails to start on demand	1.3×10^{-3}/d	29
	Fails to run	5.0×10^{-6}/hr	30
Motor-driven pump (BWR CRD)	Fails to start on demand	2.4×10^{-3}/d	31
	Fails to run	2.4×10^{-6}/hr	32
Motor-driven pump (cont. spray)	Fails to start on demand	5.0×10^{-3}/d	33
	Fails to run	5.0×10^{-5}/hr	34
Turbine-driven pump (AFW)	Fails to start on demand	1.5×10^{-2}/d	35
	Fails to run	3.0×10^{-4}/hr	36
Turbine-driven pump (RCIC)	Fails to start on demand	2.0×10^{-2}/d	37
	Fails to run	4.0×10^{-4}/hr	38
Diesel-driven pump	Fails to start on demand	2.0×10^{-2}/d	39
	Fails to run	1.0×10^{-4}/hr	40
Motor-driven air compressor	Fails to start on demand	1.0×10^{-2}/d	41
	Fails to run	1.0×10^{-4}/hr	42
Blower/ventilation	Fails to start on demand	6.0×10^{-4}/d	43
	Fails to run	1.0×10^{-5}/hr	44
Room chiller unit	Fails to start on demand	8.1×10^{-3}/d	45
	Fails to run	5.0×10^{-6}/hr	46
Motor-driven strainer	Fails to start on demand	2.7×10^{-5}/d	47
	Fails to run	5.0×10^{-6}/hr	48
Filter/strainer	Plugs	2.0×10^{-6}/hr	49

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Table A3-1

COMPONENT FAILURE DATA

Component	Failure Mode	Failure Rate	Survey Entry
Heat exchanger	Fails while operating (leaks, plugs)	1.0×10^{-6}/hr	50
Tank	Fails catastrophically	1.0×10^{-7}/hr	51
Diesel generator	Fails to start and load	1.4×10^{-2}/d	52
	Fails to run	2.4×10^{-3}/hr	53
Gas turbine-generator	Fails to start on demand	2.5×10^{-2}/d	54
	Fails to run	2.0×10^{-6}/hr	55
Battery	Fails to provide output on demand	5.0×10^{-4}/d	56
Battery charger	Fails to maintain output	7.0×10^{-6}/hr	57
Circuit breaker (4 kv)	Fails to close on demand	3.0×10^{-4}/d	58
	Opens spuriously	6.0×10^{-7}/hr	59
Circuit breaker (600 v)	Fails to close on demand	4.0×10^{-4}/d	60
	Opens spuriously	5.0×10^{-7}/hr	61
Transformer (high voltage)	Fails to continue operating	1.2×10^{-6}/hr	62
Transformer (4 kv to 600/480 v)	Fails to continue operating	7.0×10^{-7}/hr	63
Transformer (lower voltage)	Fails to continue operating	8.0×10^{-7}/hr	64
Fuse	Opens spuriously	5.0×10^{-7}/hr	65
Electrical buswork	Fails during operation	2.0×10^{-7}/hr	66
Inverter	Fails during operation	2.0×10^{-5}/hr	67
Relay	Fails to operate on demand	1.0×10^{-4}/d	68
	Operates spuriously	6.0×10^{-7}/hr	69
Flow transmitter	Output fails during operation	6.0×10^{-6}/hr	70
Pressure transmitter	Output fails during operation	5.0×10^{-6}/hr	71

Table A3-1
COMPONENT FAILURE DATA

Component	Failure Mode	Failure Rate	Survey Entry
Temperature transmitter	Output fails during operation	1.0×10^{-6}/hr	73
Pressure switch	Fails during operation	3.0×10^{-7}/hr	74
	Fails to respond on demand	2.0×10^{-4}/d	75
Level switch	Fails during operation	3.0×10^{-7}/hr	76
	Fails to respond on demand	1.0×10^{-5}/d	77

Table A3-2

MAINTENANCE UNAVAILABILITIES FOR THE BWR

System	Shoreham PRA	Train Unavailability NUREG/CR-4550	Value Selected
Reactor-core isolation cooling	1.1×10^{-2}	3.5×10^{-3}	4.0×10^{-3}
High-pressure injection	4.0×10^{-3}	3.5×10^{-3}	4.0×10^{-3}
Low-pressure injection	4.0×10^{-3}	1.9×10^{-3}	2.0×10^{-3}
Emergency service water	2.0×10^{-3}	1.9×10^{-3}	2.0×10^{-3}
Standby-liquid control	2.5×10^{-3}	3.5×10^{-3}	3.0×10^{-3}
Diesel generator*	–	6.0×10^{-3}	6.0×10^{-3}
Gas turbine-generator**	—	—	6.8×10^{-2}

Table A3-3

MAINTENANCE UNAVAILABILITIES FOR THE PWR

System	Train Unavailability Oconee PRA	Seabrook PSS	NUREG/CR-4550	Value Selected
Turbine-driven AFW	3.8×10^{-3}	4.6×10^{-3}	6.0×10^{-3}	5.0×10^{-3}
Motor-driven AFW	1.5×10^{-3}	1.8×10^{-3}	1.9×10^{-3}	2.0×10^{-3}
Safety injection	6.3×10^{-4}	1.8×10^{-3}	1.9×10^{-3}	2.0×10^{-3}
Residual-heat removal	2.0×10^{-3}	2.3×10^{-3}	1.9×10^{-3}	2.0×10^{-3}
Containment spray	2.0×10^{-3}	1.8×10^{-3}	1.9×10^{-3}	2.0×10^{-3}
Diesel generator*	–	4.6×10^{-3}	6.0×10^{-3}	6.0×10^{-3}
Gas turbine-generator**	—	—	—	6.8×10^{-2}

*The unavailability for diesel generators was taken from NUREG/CR-2989, which was also the source for NUREG/CR-4550.

**Total maintenance unavailability (forced outages plus preventive maintenance) is based on 90 generator years of experience with emergency combustion generators from Ontario Hydro system.

ANNEX A

RELIABILITY DATA BASE FOR ALWR PRAS

Table A3-4
COMMON-CAUSE FACTORS

Component	Failure Mode	Number of Failures	Survey Entry
Safety-injection pump	Fails to start	2 of 2	1.4×10^{-1}
		2 of 4	4.7×10^{-2}
		3 of 4	7.6×10^{-3}
		4 or 4	3.6×10^{-3}
	Fails to run	2 of 2	8.0×10^{-3}
		2 of 4	7.6×10^{-3}
		3 of 4	1.7×10^{-4}
		4 of 4	7.4×10^{-6}
Emergency feedwater pump	Fails to start	2 of 4	3.0×10^{-2}
		3 of 4	1.3×10^{-3}
		4 of 4	4.1×10^{-5}
	Fails to run	2 of 4	3.0×10^{-3}
		3 of 4	2.6×10^{-5}
		4 of 4	7.1×10^{-7}
Low-pressure injection pump	Fails to start	2 of 2	1.4×10^{-1}
		2 of 3	5.4×10^{-2}
		3 of 3	1.4×10^{-2}
	Fails to run	2 of 2	3.9×10^{-2}
		2 of 3	1.9×10^{-2}
		3 of 3	1.6×10^{-3}
Containment-spray pump	Fails to start	2 of 2	1.3×10^{-1}
	Fails to run	2 of 2	(no evidence)
Service-water/CCW pump	Fails to start	2 of 3	5.6×10^{-2}
		3 of 3	1.7×10^{-2}
		2 of 4	3.8×10^{-2}
		3 of 4	4.9×10^{-3}
		4 of 4	2.2×10^{-3}
	Fails to run	2 of 3	3.6×10^{-2}
		3 of 3	3.9×10^{-3}
		2 of 4	2.2×10^{-2}
		3 of 4	1.1×10^{-3}
		4 of 4	1.8×10^{-4}

Table A3-4 (continued)
COMMON-CAUSE FACTORS

Component	Failure Mode	Number of Failures	Survey Entry
Motor-operated valve	Fails to operate on demand	2 of 2	6.8×10^{-2}
		2 of 3	3.2×10^{-2}
		3 of 3	4.5×10^{-3}
		2 of 4	2.1×10^{-2}
		3 of 4	1.4×10^{-3}
		4 of 4	2.9×10^{-4}
	Transfers closed	2 of 4	1.6×10^{-2}
		3 of 4	8.5×10^{-4}
		4 of 4	1.4×10^{-4}
Diesel generator	Fails to start	2 of 2	3.8×10^{-2}
		2 of 3	1.9×10^{-2}
		3 of 3	1.3×10^{-3}
	Fails to run	2 of 2	6.8×10^{-2}
		2 of 3	3.2×10^{-2}
		3 of 3	3.8×10^{-3}
Dc battery	Fails on demand	2 of 2	7.3×10^{-2}
		2 of 3	9.2×10^{-2}
		3 of 3	1.0×10^{-2}

ANNEX A

ALWR COMPONENT FAILURE DATA SURVEY

1. Motor-operated valves: failure to operate on demand

Generic Sources			Failure Rate (/d)
NUREG/CR-4550			3.0E-3
NUREG/CR-1363			4.0E-3
Oconee PRA			4.0E-3
Seabrook PSS			4.3E-3
Five plants (below)			4.6E-3
Arithmetic Average			*4.0E-3*
Geometric Average			*3.9E-3*
Plant-Specific Evidence			
	Failures	Demands	Failure Rate
Oconee	42	6,725	6.2E-3
Zion	31	14,577	2.1E-3
Indian Point	3	1,505	2.0E-3
Millstone	60	11,732	5.1E-3
PWR X	69	10,052	6.9E-3
Total:	205	44,691	4.6E-3

Value selected: **4.0E-3**

Rationale: Value is representative of both generic data sources and plant-specific failure rates.

2. Motor-operated valves: transfer closed

Generic Sources			Failure Rate (/hr)
NUREG/CR-4550			1.3E-7
NUREG/CR-1363			5.7E-8
NUREG/CR-2815			2.0E-7
Oconee PRA			2.3E-7
Seabrook PSS			9.3E-8
Fourplants (below)			1.4E-7
Arithmetic Average			*1.4E-7*
Geometric Average			*1.4E-7*
Plant-Specific Evidence			
	Failures	Hours	Failure Rate
Oconee	0	1,890,000	1.8E-7
Zion	0	3.220,000	1.0E-7
Indian Point	0	1,429,000	2.3E-7
PWR X	1	817,399	1.2E-6
Total	1	7,356,399	1.4E-7

Value selected: **1.4E-7**

Rationale: Value is representative of both generic data sources and plant-specific failure rates.

3. Air-operated valves: failure to operate on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	3.0E-3
NUREG/CR-1363	6.6E-4
Oconee PRA	9.0E-4
Seabrook PSS	1.5E-3
Five plants (below)	6.2E-3
Four plants (below, not X)	1.6E-3
Arithmetic Average with X	*2.5E-3*
Geometric Average with X	*1.8E-3*
Arithmetic Average without X	*1.5E-3*
Geometric Average without X	*1.3E-3*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	3	1,349	2.2E-3
Zion	3	1,540	1.9E-3
Indian Point	1	1,440	6.9E-4
Millstone	--	--	--
PWR X	35	2,433	1.4E-2
Total:	42	6,762	6.2E-3

Value selected: **2.0E-3**

Rationale: Value is consistent with most data sources. PWR X saw repetitive failures in the past that have apparently been corrected, and are of questionable applicability for ALWRs.

4. Air-operated valves: transfer closed

Generic Sources	Failure Rate (/hr)
NUREG/CR-4550	1.3E-7
NUREG/CR-1363	1.0E-7
Oconee PRA	2.3E-7
Seabrook PSS	2.7E-7
Four plants (below)	9.0E-8
Arithmetic Average	*1.6E-7*
Geometric Average	*1.5E-7*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	0	194,000	1.7E-6
Zion	0	2,130,000	1.6E-7
Indian Point	0	444,000	7.5E-7
PWR X	0	954,171	3.5E-7
Total	0	3,722,171	9.0E-8

Value selected: **1.5E-7**

Rationale: Value is representative of generic sources, and also reflects plant-specific experience with no failures.

5. Check valves (other than stop-check): failure to operate on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	1.0E-4
NUREG/CR-1363	1.1E-4
Oconee PRA	1.0E-4
Seabrook PSS	2.7E-4
Five plants (below)	3.4E-4
Arithmetic Average	*1.8E-4*
Geometric Average	*1.6E-4*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	1	6,279	1.6E-4
Zion	0	6,968	4.8E-5
Indian Point	0	1,444	2.3E-4
Millstone	3	3,896	7.7E-4
PWR X	3	1,923	1.6E-3
Total	7	20,510	3.4E-4

Value selected: **2.0E-4**

Rationale: Value reflects more recent generic data and plant-specific experience.

---

6. Check valves (other than stop-check): transfer closed

Generic Sources	Failure Rate (/hr)
Oconee PRA	2.3E-7
Seabrook PSS	1.0E-8
Two plants (below)	9.5E-7
Arithmetic Average	*4.0E-7*
Geometric Average	*1.3E-7*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	0	387,000	8.6E-7
PWR X	1	665,016	1.5E-6
Total	1	1,052,016	9.5E-7

Value selected: **2.0E-7**

Rationale: Rare mode, very uncertain failure rate; limited available data from plant-specific sources.

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7. Check valves (other than stop-check): reverse leakage

Generic Sources	Failure Rate (/hr)
NUREG/CR-1363	6.6E-7
Seabrook PSS	5.4E-7
Arithmetic Average	*6.0E-7*
Geometric Average	*6.0E-7*

Plant-Specific Evidence

Not available.

Value selected: **6.0E-7**

Rationale: Limited data available. Current expert opinion is that failure rate for sufficient leakage to constitute gross rupture is lower.

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8. Stop check valves: failure to operate on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	1.0E-4
NUREG/CR-1363	1.1E-4
Oconee PRA	1.0E-4
Seabrook PSS	9.1E-4
Two plants (below)	5.7E-3
Arithmetic Average	*1.4E-3*
Geometric Average	*3.6E-4*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	1	572	1.7E-3
PWR X	5	476	1.1E-2
Total	6	1,048	5.7E-3

Value selected: **1.0E-3**

Rationale: Most older data sources did not distinguish among check-valve types; generic sources were therefore weighted less.

9. Stop check valves: transfer closed

Generic Sources	Failure Rate (/hr)
Oconee PRA	2.3E-7
Seabrook PSS	1.0E-8
Two plants (below)	4.9E-7
Arithmetic Average	*2.4E-7*
Geometric Average	*1.1E-7*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	0	342,000	9.7E-7
PWR X	0	345,047	9.7E-7
Total	0	687,047	4.9E-7

Value selected: **2.0E-7**

Rationale: Limited applicable data, no failures in plant-specific evidence. Value is consistent with that for other check valves.

10. Stop check valves: reverse leakage

Generic Sources	Failure Rate (/hr)
NUREG/CR-1363	6.6E-7
Seabrook PSS	5.4E-7
Arithmetic Average	*6.0E-7*
Geometric Average	*6.0E-7*

Plant-Specific Evidence

None available.

Value selected: **6.0E-7**

Rationale: Limited applicable data. Value is also consistent with that for other check valves.

11. Check valves: internal rupture

Generic Sources	Failure Rate (/hr)
NUREG/CR-5116	5.0E-9
NUREG/CR-2815	1.0E-7
Arithmetic Average	5.3E-8
Geometric Average	2.2E-8

Plant-Specific Evidence

None available.

Value selected: **5.0E-9**

Rationale: Value from detailed review by experts for NUREG-1150; reviews found this to be a very rare failure mode.

12. Manual valves: plug/transfer closed

Generic Sources	Failure Rate (/hr)
Oconee FRA	3.4E-8
Seabrook PSS	4.2E-8
Four plants (below)	3.5E-8
Arithmetic Average	*3.7E-8*
Geometric Average	*3.7E-8*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	1	3,090,000	3.2E-7
Zion	0	7,870,000	4.2E-8
Indian Point	0	8,270,000	4.0E-8
PWR X	0	9,510,241	3.5E-8
Total	1	28,740,241	3.5E-8

Value selected: **3.7E-8**

Rationale: Data sources in very close agreement, despite rare nature of failure mode.

13. Pressurizer safety valves (PWR): failure to open on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-1363	6.2E-3
Oconee PRA	2.7E-4
Seabrook PSS	3.3E-4
Two plants (below)	1.5E-2
Arithmetic Average	*5.5E-3*
Geometric Average	*1.7E-3*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	0	10	3.3E-2
PWR X	0	12	2.8E-2
Total	0	22	1.5E-2

Value selected: **1.0E-3**

Rationale: Plant-specific data of limited use, wide range in generic sources.

14. Pressurizer safety valves (PWR): failure to reclose on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	1.0E-2
Oconee PRA (steam)	4.9E-3
Oconee PRA (water)	1.0E-1
Seabrook PSS (steam)	2.9E-3
Seabrook PSS (water)	1.0E-1
Two plants (below)	1.5E-2
Arithmetic Average	*3.9E-2*
Geometric Average	*1.7E-2*
Arithmetic Average (steam only)	*8.2E-3*
Geometric Average (steam only)	*1.0E-2*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	0	10	3.3E-2
PWR X	0	12	2.8E-2
Total	0	22	1.5E-2

Value selected: **7.0E-3**

Rationale: Plant-specific data again of limited use. Problems with failure after liquid flow should be eliminated in ALWRs.

15. Safety/relief valves (BWR): fail to open on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-1363	7.9E-3
Plant-Specific Evidence	
Browns Ferry PRA	8.0E-3
One plant (below)	3.4E-3
Arithmetic Average	6.4E-3
Geometric Average	6.0E-3

Plant-Specific Evidence

Browns Ferry	1	290	3.4E-3

Value selected: **6.0E-3**

Rationale: Value selected is representative of all sources.

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16. Safety/relief valves (BWR): fail to reclose

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	1.0E-2
NUREG/CR-1363	4.5E-3
Browns Ferry PRA	5.0E-3
One plant (below)	6.9E-3
Arithmetic Average	*6.6E-3*
Geometric Average	*6.3E-3*

Plant-Specific Evidence

Browns Ferry	2	290	6.9E-3

Value selected: **6.5E-3**

Rationale: Available values reasonably close; value selected is representative.

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17. Pilot-operated relief valves: failure to open on demand

Generic Sources	Failure Rate (/d)
Oconee PRA	8.0E-3
Seabrook PSS	4.3E-3
Two plants (below)	8.5E-3
Arithmetic Average	*6.9E-3*
Geometric Average	*6.6E-3*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	0	31	1.1E-2
PWR X	0	8	4.2E-2
Total	0	39	8.5E-3

Value selected: **7.0E-3**

Rationale: Sources are quite close together, value is representative.

---

18. Pilot-operated relief valves: failure to reclose on demand

Generic Sources	Failure Rate (/d)
Oconee PRA	5.0E-3
Seabrook PSS	2.5E-2
Two plants (below)	5.1E-2
Arithmetic Average	*2.7E-2*
Geometric Average	*1.9E-2*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	1	31	3.2E-2
PWR X	1	8	1.3E-1
Total	2	39	5.1E-2

Value selected: **2.5E-2**

Rationale: Plant-specific evidence and more recent generic source given higher weight.

19. Motor-driven pumps (all): failure to start on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	3.0E-3
NUREG/CR-1205	4.2E-4
Oconee PRA	5.0E-4
Seabrook PSS, standby	2.4E-3
Seabrook PSS, normally-operating	3.3E-3
Northeast Utilities	1.3E-3
Six plants (below)	2.0E-3
Arithmetic Average	*2.0E-3*
Geometric Average	*1.5E-3*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	4	972	4.1E-3
Zion	7	3,600	1.9E-3
Indian Point	9	1,593	5.6E-3
Millstone	22	5,129	4.3E-3
Browns Ferry	13	8,330	1.6E-3
PWR X	2	835	2.4E-3
Total	57	20,459	2.8E-3

Value selected: **2.0E-3**

Rationale: Value is consistent with most available sources of data.

20. Motor-driven pumps (all): failure to run

Generic Sources	Failure Rate (/hr)
NUREG/CR-4550	3.0E-5
NUREG/CR-1205	6.0E-6
Oconee PRA	2.0E-5
Seabrook PSS	3.4E-5
Northeast Utilities	4.0E-5
Six plants (below)	2.0E-5
Arithmetic Average	*2.5E-5*
Geometric Average	*2.1E-5*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	3	98,120	3.1E-5
Zion	1	340,412	2.9E-6
Indian Point	9	258,684	3.5E-5
Millstone	20	953,038	2.1E-5
Browns Ferry	9	284,134	3.2E-5
PWR X	0	191,577	1.7E-6
Total	42	2,126,145	2.0E-5

Value selected: **2.5E-5**
Rationale: Value is consistent with all of the available sources of data.

21. Motor-driven LPI/RHR pumps: failure to start on demand

Generic Sources	Failure Rate (/d)
Northeast Utilities	2.0E-3
Four plants (below)	2.5E-3
Arithmetic Average	*2.3E-3*
Geometric Average	*2.3E-3*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	0	223	1.5E-3
Millstone	3	259	1.2E-2
Browns Ferry	3	1,688	1.8E-3
PWR X	0	199	1.7E-3
Total	6	2,369	2.5E-3

Value selected: **2.3E-3**
Rationale: Available sources of data agree reasonably well.

22. Motor-driven LPI/RHR pumps: failure to run

Generic Sources	**Failure Rate (/hr)**
Northeast Utilities	9.6E-6
Six plants (below)	1.7E-5
Arithmetic Average	*1.3E-5*
Geometric Average	*1.3E-5*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	1	11,287	8.9E-5
Zion	0	32,500	1.0E-5
Indian Point	2	8,065	2.5E-4
Millstone	0	15,050	2.2E-5
Browns Ferry	0	88,900	3.7E-6
PWR X	0	17,211	1.9E-5
Total	3	173,013	1.7E-5

Value selected: **1.0E-5**

Rationale: Value reasonably reflects available sources of data that apply directly for this type of pump. Plant-specific experience is strongly affected by Indian Point.

23. Motor-driven safety injection pumps: failure to start on demand

Generic Sources	**Failure Rate (/d)**
Northeast Utilities	2.0E-3
Four plants (below)	3.1E-4
Arithmetic Average	*1.2E-3*
Geometric Average	*7.8E-4*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	1	530	1.9E-3
Millstone	0	954	3.5E-4
Browns Ferry	0	1,631	2.0E-4
PWR X	0	134	2.5E-3
Total	1	3,249	3.1E-4

Value selected: **1.0E-3**

Rationale: Value reasonably reflects limited available sources of data that apply directly for this type of pump.

24. Motor-driven safety-injection pumps: failure to run

Generic Sources	Failure Rate (/hr)
Northeast Utilities	8.0E-5
Five plants (below)	2.6E-5
Arithmetic Average	*5.3E-5*
Geometric Average	*4.5E-5*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	0	38,787	8.6E-6
Zion	0	46	7.2E-3
Indian Point	1	124	8.1E-3
Browns Ferry	0	78	4.3E-3
PWR X	0	67	5.0E-3
Total	1	39,102	2.6E-5

Value selected: **5.0E-5**

Rationale: Value reasonably reflects limited available sources of data that apply directly for this type of pump.

25. Motor-driven emergency feedwater pumps: failure to start on demand

Generic Sources	Failure Rate (/d)
Northeast Utilities	1.3E-3
One plant (below)	8.6E-3
Arithmetic Average	*5.0E-3*
Geometric Average	*3.3E-3*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Zion	4	464	8.6E-3

Value selected: **3.0E-3**

Rationale: Limited available data applying directly to this type of pump. Value influenced more by value for motor-driven pumps in general.

26. Motor-driven emergency feedwater pumps: failure to run

Generic Sources	Failure Rate (/hr)
Northeast Utilities	8.0E-5
Two plants (below)	2.0E-4
Arithmetic Average	*1.4E-4*
Geometric Average	*1.3E-4*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Zion	1	3,800	2.6E-4
Indian Point	1	6,320	1.6E-4
Total	2	10,120	2.0E-4

Value selected: **1.5E-4**

Rationale: Plant-specific data given more weight, since only generic value is from WASH-1400.

27. Motor-driven service-water pumps: failure to start on demand

Generic Sources	Failure Rate (/d)
Northeast Utilities	1.5E-3
Three plants (below)	7.7E-3
Arithmetic Average	*4.6E-3*
Geometric Average	*3.4E-3*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	0	61	5.5E-3
Millstone	9	1,085	8.3E-3
Browns Ferry	9	4,387	2.1E-3
PWR X	[illegible]	160	6.3E-3
Total	19	5.693	3.3E-3

Value selected: **2.4E-3**

Rationale: Value reasonably reflects limited available sources of data that apply directly for this type of pump.

28. Motor-driven service-water pumps: failure to run

Generic Sources	Failure Rate (/hr)
Northeast Utilities	3.8E-5
Five plants (below)	2.6E-5
Arithmetic Average	*3.2E-5*
Geometric Average	*3.2E-5*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	2	47,991	4.2E-5
Zion	0	152,000	2.2E-6
Indian Point	5	122,000	4.1E-5
Browns Ferry	9	195,000	4.6E-5
PWR X	0	87,072	3.8E-6
Total	16	604,063	2.6E-5

Value selected: **3.2E-5**

Rationale: Value reasonably reflects limited available sources of data that apply directly for this type of pump, and that are quite close together.

29. Motor-driven component-cooling water pumps: failure to start on demand

Generic Sources	Failure Rate (/d)
Northeast Utilities	1.8E-3
Two plants (below)	8.9E-4
Arithmetic Average	*1.3E-3*
Geometric Average	*1.3E-3*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Millstone	0	915	3.6E-4
PWR X	1	209	4.8E-3
Total	1	1,124	8.9E-4

Value selected: **1.3E-3**

Rationale: Value reasonably reflects available data sources.

30. Motor-driven component-cooling water pumps: failure to run

Generic Sources	Failure Rate (/hr)
Northeast Utilities	1.0E-5
Three plants (below)	1.3E-6
Arithmetic Average	*5.7E-6*
Geometric Average	*3.6E-6*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Zion	0	76,000	4.4E-6
Indian Point	0	122,096	2.7E-6
PWR X	0	52,232	6.4E-6
Total	0	250,328	1.3E-6

Value selected: **5.0E-6**

Rationale: Limited data available suggests relatively wide range of values. Value selected represents average value.

31. Motor-driven control-rod drive pumps: failure to start on demand

Generic Sources	Failure Rate (/d)
Northeast Utilities	1.8E-3
One plant (below)	2.9E-3
Arithmetic Average	*2.4E-3*
Geometric Average	*2.3E-3*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Millstone	1	342	2.9E-3

Value selected: **2.4E-3**

Rationale: Limited data available, significant (if not complete) overlap in data sources. Value is consistent with that for motor-driven pumps in general.

---

32. Motor-driven control-rod drive pumps: failure to run

Generic Sources	Failure Rate (/hr)
Northeast Utilities	1.6E-6
One plant (below)	3.3E-6
Arithmetic Average	*2.4E-6*
Geometric Average	*2.3E-6*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Millstone	0	101,652	3.3E-6

Value selected: **2.4E-6**

Rationale: Limited data available, significant overlap in sources.

---

33. Motor-driven containment-spray pumps: failure to start on demand

Generic Sources	Failure Rate (/d)
Northeast Utilities	1.0E-3
One plant (below)	2.1E-2
Arithmetic Average	*1.1E-2*
Geometric Average	*4.6E-3*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	3	140	2.1E-2

Value selected: **5.0E-3**

Rationale: Limited data available, wide spread in values. Value selected is consistent with geometric mean of available sources.

---

34. Motor-driven containment-spray pumps: failure to run

Generic Sources	Failure Rate (/hr)
Northeast Utilities	1.5E-5
Three plants (below)	1.9E-3
Arithmetic Average	*9.3E-4*
Geometric Average	*1.7E-4*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	0	40	8.3E-3
Zion	0	66	5.1E-3
Indian Point	0	74	4.5E-3
Total	0	180	1.9E-3

Value selected: **5.0E-5**

Rationale: Limited data available, very limited value in plant-specific sources due to limited experience and no failures. Value selected weighted Northeast data most heavily.

35. Turbine-driven auxiliary feedwater pumps: failure to start on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	3.2E-3
NUREG/CR-1205	9.6E-3
Oconee PRA	4.0E-3
Seabrook PSS	3.3E-2
Northeast Utilities	2.3E-2
Four plants (below)	2.1E-2
Arithmetic Average	*1.6E-2*
Geometric Average	*1.1E-2*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	6	113	5.3E-2
Zion	6	231	2.6E-2
Indian Point	0	57	5.8E-3
PWR X	2	260	7.7E-3
Total	14	661	2.1E-2

Value selected: **1.5E-2**

Rationale: Older generic data sources tended to underestimate this rate; value selected is more consistent with more recent data sources.

ALWR COMPONENT FAILURE DATA SURVEY

36. Turbine-driven auxiliary feedwater pumps: failure to run

Generic Sources	Failure Rate (/hr)
NUREG/CR-4550	1.3E-4
NUREG/CR-1205	4.3E-5
Oconee PRA	2.0E-5
Seabrook PSS	1.0E-3
Northeast Utilities	7.6E-6
Four plants (below)	2.0E-3
Arithmetic Average	*5.5E-4*
Geometric Average	*1.1E-4*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	1	94	1.1E-2
Zion	0	1,900	1.8E-4
Indian Point	1	1,240	8.1E-4
PWR X	5	194	2.6E-2
Total	7	3,428	2.0E-3

Value selected: **3.0E-4**

Rationale: Wide range in available sources of data. Northeast experience is much better than general industry experience. PWR X experience is much worse than other plants. Value selected appears to be reasonable.

37. Turbine-driven RCIC pumps: failure to start on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	3.2E-3
NUREG/CR-1205	1.2E-2
Browns Ferry PRA	4.0E-2
One plant (below)	3.4E-2
Arithmetic Average	*2.2E-2*
Geometric Average	*1.5E-2*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Browns Ferry	21	614	3.4E-2

Value selected: **2.0E-2**

Rationale: Sources generally agree, except for NUREG/CR-4550, which is much lower. Value selected is representative.

38. Turbine-driven RCIC pumps: failure run

Generic Sources	Failure Rate (/hr)
NUREG/CR-4550	1.3E-4
Browns Ferry	4.1E-4
One plant (below)	4.4E-3
Arithmetic Average	*1.6E-3*
Geometric Average	*6.2E-4*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Browns Ferry	0	76	4.4E-3

Value selected: **4.0E-4**

Rationale: Very limited plant-specific data available. Generic data from Browns Ferry PRA given greater weight.

39. Diesel-driven pumps: failure to start on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-1205	3.0E-2
Northeast Utilities	3.1E-3
Two plants (below)	2.6E-2
Arithmetic Average	*2.0E-2*
Geometric Average	*1.3E-2*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Zion	1	183	5.5E-3
Millstone	8	158	5.1E-2
Total	9	341	2.6E-2

Value selected: **2.0E-2**

Rationale: Available sources are generally consistent, value selected is representative.

40. Diesel-driven pumps: failure to run

Generic Sources	Failure Rate (/hr)
NUREG/CR-1205	2.6E-5
Northeast Utilities	8.0E-5
One plant (below)	6.1E-2
Arithmetic Average	*2.0E-2*
Geometric Average	*5.0E-4*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Zion	2	33	6.1E-2

Value selected: **1.0E-4**

Rationale: Zion experience is very different from generic data. Value selected is weighted heavily toward generic sources.

41. Air compressors: failure to start on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	5.3E-2
Oconee PRA	5.0E-3
Seabrook PSS	3.3E-3
Arithmetic Average	*2.0E-2*
Geometric Average	*9.6E-3*

Plant-Specific Evidence

Not available.

Value selected: **1.0E-2**

Rationale: Wide range in values; value selected is representative.

42. Air compressors: failure to run

Generic Sources	Failure Rate (/hr)
NUREG/CR-4550	4.8E-5
NPRD-2	2.1E-5
Oconee PRA	2.9E-4
Seabrook PSS	9.8E-5
Arithmetic Average	*1.1E-4*
Geometric Average	*7.3E-5*

Plant-Specific Evidence

Not available.

Value selected: **1.0E-4**

Rationale: Most values are reasonably close; value selected is representative.

43. Blower/ventilation fans: failure to start on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	3.8E-4
Oconee PRA	5.0E-4
Seabrook PSS	4.8E-4
Four plants (below)	1.1E-3
Arithmetic Average	*6.1E-4*
Geometric Average	*5.6E-4*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	3	237	1.3E-2
Zion	2	1,155	1.7E-3
Indian Point	0	45	7.4E-3
PWR X		4,086	2.4E-4
Total	6	5,523	1.1E-3

Value selected: **6.0E-4**

Rationale: Most values are reasonably close; value selected is representative.

44. Ventilation fans: failure to run

Generic Sources	Failure Rate (/hr)
NUREG/CR-4550	1.3E-5
NPRD-2	2.6E-6
Oconee PRA	1.9E-5
Seabrook PSS	7.9E-6
Four plants (below)	9.6E-6
Arithmetic Average	*1.0E-5*
Geometric Average	*8.6E-6*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	1	81,351	1.2E-5
Zion	0	152,000	2.2E-6
Indian Point	2	122,000	1.6E-5
PWR X	1	60,723	1.6E-5
Total	4	416,074	9.6E-6

Value selected: **1.0E-5**

Rationale: Most values are reasonably close; value selected is representative.

45. Room chiller unit: failure to start on demand

Generic Sources	Failure Rate (/d)
Seabrook PSS	8.1E-3

Plant-Specific Evidence
Not available.
Value selected: **8.1E-3**
Rationale: Only value readily available.

---

46. Room chiller unit: fails to continue operating

Generic Sources	Failure Rate (/hr)
NPRD-2	1.0E-6
Seabrook PSS	7.9E-6
Arithmetic Average	*4.4E-6*
Geometric Average	*2.8E-6*

Plant-Specific Evidence
Not available.
Value selected: **5.0E-6**
Rationale: Limited data available; greater weight given to Seabrook since it reflects nuclear power plant experience. NPRD-2 reflects significant level of operating experience, but no nuclear experience.

---

47. Strainer: fails to start

Generic Sources	Failure Rate (/d)
IEEE-500	2.7E-5

Plant-Specific Evidence
Not available.
Value selected: **2.7E-5**
Rationale: Only value readily available. Value seems low in comparison to other motor-driven components.

48. Strainer: fails to continue operating

Generic Sources	Failure Rate (/hr)
IEEE-500	3.8E-6
Seabrook PSS	6.2E-6
Arithmetic Average	*5.0E-6*
Geometric Average	*4.9E-6*

Plant-Specific Evidence

Not available.

Value selected: **5.0E-6**

Rationale: Generic values are quite close, value selected is very representative.

---

49. Strainer or filter: plugs

Generic Sources	Failure Rate (/hr)
NPRD-2	3.0E-6
Seabrook PSS	1.1E-6
Arithmetic Average	*2.0E-6*
Geometric Average	*1.9E-6*

Plant-Specific Evidence

Not available

Value selected: **2.0E-6**

Rationale: Limited data available. Generic values are quite close, value selected is representative.

---

50. Heat exchanger: fails while operating (severe leakage, plugging)

Generic Sources	Failure Rate (/hr)
NPRD-2	9.0E-7
Seabrook PSS	2.0E-6
Two plants (below)	6.9E-7
Arithmetic Average	*1.2E-6*
Geometric Average	*1.1E-6*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Zion	0	236,000	1.4E-6
Indian Point	0	244,000	1.4E-6
Total	0	480,000	6.9E-7

Value selected: **1.0E-6**

Rationale: Values are reasonably close, value selected is very representative.

51. Tanks: fail catastrophically

Generic Sources	Failure Rate (/hr)
NPRD-2	1.6E-6
Seabrook PSS	2.7E-8
Arithmetic Average	*8.2E-7*
Geometric Average	*2.1E-7*

Plant-Specific Evidence
Not available.

Value selected: **1.0E-7**

Rationale: Wide spread in sources, uncertain and rare failure rate. Value selected weights Seabrook more heavily due to uncertainty in nature of NPRD-2 data.

52. Diesel Generators: fail to start on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	3.8E-2
NUREG/CR-1362	4.4E-2
NUREG/CR-2989	3.3E-2
NSAC-108*	1.4E-2
Seabrook PSS**	3.8E-2
Northeast Utilities	7.0E-3
Four plants (below)	1.3E-2
Arithmetic Average	*2.7E-2*
Geometric Average	*2.2E-2*

*Includes some failures to run, but not dominant.
**Includes failure to run during first hour of operation.

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Zion	30	1,693	1.8E-2
Indian Point	6	609	9.9E-3
Millstone	3	652	4.6E-3
PWR X	5	502	1.0E-2
Total	44	3,456	1.3E-2

Value selected: **1.4E-2**

Rationale: NSAC-108 provides extensive review of recent operating experience, reflecting most current maintenance practices, and accounting well for actual demands. Failure rate reflects some failures in load/run phases of operation, but these are not expected to impact the result substantially. Therefore, NSAC-108 value recommended.

53. Diesel Generators: fail to run

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	1.3E-3
NUREG/CR-1362	2.6E-2
NUREG/CR-2989	2.4E-3
Seabrook PSS	2.5E-3
Northeast Utilities	1.5E-3
Four plants (below)	3.9E-3
Arithmetic Average	*6.3E-3*
Geometric Average	*3.3E-3*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Zion	6	1,340	4.5E-3
Indian Point	0	408	8.2E-4
Millstone	1	1,018	9.8E-4
PWR X	7	846	8.3E-3
Total	14	3,612	3.9E-3

Value selected: **2.4E-3**

Rationale: Many data sources include failures to load or other failures immediately after starting that are not appropriate for long (e.g., 24-hr) mission times. NUREG/CR-2989 collected data specifically from long-duration tests; value is also consistent with several other data sources.

54. Energy combustion turbine-generators: failure to start on demand

Generic Sources	Failure Rate (/d)
Ontario Hydro system	2.5E-2
One plant (below)	3.4E-2
Arithmetic Average	*2.9E-2*
Geometric Average	*2.9E-2*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Millstone	28	834	3.4E-2

Value selected: **2.5E-2**

Rationale: Data sources are quite similar; Ontario Hydro data represents 90 generator-yr of experience, and weighed more heavily.

55. Emergency combustion turbine-generators: failure to run

Generic Sources	Failure Rate (/hr)
Ontario Hydro system	1.7E-6
One plant (below)	1.8E-4
Arithmetic Average	*8.9E-5*
Geometric Average	*1.7E-5*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Millstone	1	5,697	1.8E-4

Value selected: **2.0E-6**

Rationale: Data sources very different; Ontario Hydro data represents 90 generator-yr of experience, and weighed more heavily.

56. Batteries: failure of output on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	1.4E-3
NUREG-0666	3.3E-4
Oconee PRA*	3.2E-5
Seabrook PSS	4.8E-4
NPRD-2*	1.6E-4
Three plants (below)	1.5E-3
Arithmetic Average	*6.6E-4*
Geometric Average	*3.5E-4*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	0	96,426	3.5E-6
Zion	0	202,000	1.7E-6
Indian Point	2	167,800	1.2E-5
Total	2	466,226	4.3E-6
Total (/d)*			1.5E-3

*Assuming monthly testing.

Value selected: **5.0E-4**

Rationale: Values are generally quite close, and value selected is representative. Quarterly vs. monthly testing would drive value closer to 1 - 2 E-3/d.

57. Battery charger: failure to maintain output

Generic Sources	Failure Rate (/hr)
NUREG/CR-4550	4.0E-6
NUREG-0666	2.8E-6
Oconee PRA	3.1E-6
Seabrook PSS	1.9E-5
Four plants (below)	1.1E-5
Arithmetic Average	*8.0E-6*
Geometric Average	*5.9E-6*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	1	96,426	1.0E-5
Zion	0	202,000	1.7E-6
Indian Point	2	167,800	1.2E-5
Millstone	5	229,488	2.2E-5
Total	8	695,714	1.1E-5

Value selected: **7.0E-6**

Rationale: Values are generally quite close, and value selected is representative.

58. Circuit breaker (4 kv): fails to close on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	1.3E-4
Northeast Utilities System	3.4E-4
Oconee PRA	4.3E-5
Seabrook PSS	1.6E-3
Five plants (below)	6.2E-5
Arithmetic Average	*4.4E-4*
Geometric Average	*1.8E-4*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	2	1,192	1.7E-3
Zion	0	202,000	1.7E-6
Indian Point	2	167,800	1.2E-5
Millstone	3	34,333	8.7E-5
PWR X	18	1,144	1.6E-2
Total	25	406,469	6.2E-5

Value selected: **3.0E-4**

Rationale: Values are generally close; experience for Zion and Indian Point is for all types of breakers, and is therefore given slightly less weight.

59. Circuit breaker (4 kv): opens spuriously

Generic Sources	Failure Rate (/hr)
NPRD-2	6.8E-7
Northeast Utilities System	1.3E-6
Oconee PRA	1.6E-7
Seabrook PSS	8.3E-7
Four plants (below)	3.7E-7
Arithmetic Average	*6.7E-7*
Geometric Average	*5.3E-7*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	0	888,000	3.8E-7
Zion	0	910,000	3.7E-7
Indian Point	1	732,000	1.4E-6
PWR X	0	191,577	1.7E-6
Total	1	2,721,577	3.7E-7

Value selected: **6.0E-7**

Rationale: Values are generally close, and value selected is representative.

60. Circuit breaker (600 v): fails to close on demand

Generic Sources	Failure Rate (/d)
NUREG/CR-4550	1.3E-4
Northeast Utilities System	1.3E-3
Seabrook PSS	2.3E-4
Arithmetic Average	*5.5E-4*
Geometric Average	*3.4E-4*

Plant-Specific Evidence

Not available.

Value selected: **4.0E-4**

Rationale: Value selected reasonably reflects the available sources.

---

61. Circuit breaker (600 v): opens spuriously

Generic Sources	Failure Rate (/hr)
NPRD-2	6.8E-7
Northeast Utilities System	1.3E-6
Oconee PRA	1.6E-7
Seabrook PSS	2.7E-7
One plant (below)	6.6E-7
Arithmetic Average	*6.1E-7*
Geometric Average	*4.8E-7*

Plant-Specific Evidence

	Failures	Demands	Failure Rate
Oconee	2	3,040,000	6.6E-7

Value selected: **5.0E-7**

Rationale: Value selected reasonably reflects the available sources.

---

62. Transformer (high voltage): fails to continue operating

Generic Sources	Failure Rate (/hr)
Oconee PRA	1.7E-6
Seabrook PSS	1.6E-6
IEEE-500	3.2E-7
Three plants (below)	1.4E-6
Arithmetic Average	*1.3E-6*
Geometric Average	*1.1E-6*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	0	81,900	4.1E-6
Zion	1	301,000	3.3E-6
Indian Point	0	313,000	1.1E-6
Total	1	695,900	1.4E-6

Value selected: **1.2E-6**

Rationale: Available data sources are reasonably close, and value selected is representative.

---

63. Transformer (4 kv to 600/480 v): fails to continue operating

Generic Sources	Failure Rate (/hr)
Oconee PRA	9.1E-7
Seabrook PSS	6.9E-7
IEEE-500	3.4E-7
Three plants (below)	9.5E-7
Arithmetic Average	*7.2E-7*
Geometric Average	*6.7E-7*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	0	434,000	7.7E-7
Zion	1	301,000	3.3E-6
Indian Point	0	313,000	1.1E-6
Total	1	1,048,000	9.5E-7

Value selected: **7.0E-7**

Rationale: Available data sources are reasonably close, and value selected is representative.

---

64. Transformer (lower voltage): fails to continue operating

Generic Sources	Failure Rate (/hr)
Oconee PRA	1.1E-6
Seabrook PSS	1.6E-6
IEEE-500	2.4E-7
Three plants (below)	7.0E-7
Arithmetic Average	*9.0E-7*
Geometric Average	*7.3E-7*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	0	820,000	4.1E-7
Zion	1	301,000	3.3E-6
Indian Point	0	313,000	1.1E-6
Total	1	1,434,000	7.0E-7

Value selected: **8.0E-7**

Rationale: Available data sources are reasonably close, and value selected is representative.

---

65. Fuse: opens spuriously

Generic Sources	Failure Rate (/hr)
NUREG-0666	1.0E-6
NPRD-2	1.4E-7
Seabrook PSS	9.2E-7
IEEE-500	1.5E-7
Arithmetic Average	*5.5E-7*
Geometric Average	*3.7E-7*

Plant-Specific Evidence

Not available.

Value selected: **5.0E-7**

Rationale: Available data sources are somewhat close, and value selected is representative.

66. Electrical buswork: fails during operation

Generic Sources	Failure Rate (/hr)
Oconee PRA	3.6E-6
Seabrook PSS	5.0E-7
IEEE-500	1.2E-7
Three plants (below)	4.6E-8
Arithmetic Average	*1.1E-6*
Geometric Average	*3.2E-7*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	0	2,604,000	1.3E-7
Zion	0	3,030,000	1.1E-7
Indian Point	0	1,575,000	2.1E-7
Total	0	7,209,000	4.6E-8

Value selected: **2.0E-7**

Rationale: Wide variation in reported failure rates. Value selected is influenced by plant-specific experience.

67. Inverter: fails during operation

Generic Sources	Failure Rate (/hr)
Oconee PRA	1.3E-4
Seabrook PSS	1.8E-5
Three plants (below)	1.6E-5
Arithmetic Average	*5.5E-5*
Geometric Average	*3.4E-5*

Plant-Specific Evidence

	Failures	Hours	Failure Rate
Oconee	9	337,000	2.7E-5
Zion	3	304,000	9.9E-6
Indian Point	1	167,800	6.0E-6
Total	13	808,800	1.6E-5

Value selected: **2.0E-5**

Rationale: Sources agree well, except for Oconee generic. Value is most heavily influenced by other sources.

68. Relay: fails to operate on demand

Generic Sources	Failure Rate (/d)
Oconee PRA	2.4E-4
Seabrook PSS	2.4E-4
IEEE-500	4.5E-6
Arithmetic Average	*1.6E-4*
Geometric Average	*6.4E-5*

Plant-Specific Evidence

Not available.

Value selected: **1.0E-4**

Rationale: Limited sources; IEEE-500 value is not consistent with other sources. Other two sources weighted most heavily.

---

69. Relay: failure to operate (per hr)

Generic Sources	Failure Rate (/hr)
NPRD-2	1.4E-6
Oconee PRA	8.1E-7
Seabrook PSS	4.2E-7
IEEE-500	6.0E-8
Arithmetic Average	*6.8E-7*
Geometric Average	*4.1E-7*

Plant-Specific Evidence

Not available.

Value selected: **6.0E-7**

Rationale: Available sources are similar, except for IEEE-500, which is much lower.

---

70. Flow transmitter: output fails during operation

Generic Sources	Failure Rate (/hr)
Oconee PRA	2.6E-6
Seabrook PSS	6.3E-6
NPRD-2	8.4E-6
Arithmetic Average	*5.7E-6*
Geometric Average	*5.1E-6*

Plant-Specific Evidence

Not available.

Value selected: **6.0E-6**

Rationale: Available sources are similar, and value selected is representative.

---

71. Pressure transmitter: output fails during operation

Generic Sources	Failure Rate (/hr)
Oconee PRA	1.4E-5
Seabrook PSS	7.6E-6
IEEE-500	8.8E-7
NPRD-2	2.6E-6
Arithmetic Average	*6.3E-6*
Geometric Average	*4.0E-6*

Plant-Specific Evidence
Not available.

Value selected: **5.0E-6**

Rationale: Available sources are somewhat similar, and value selected is representative.

72. Level transmitter: output fails during operation

Generic Sources	Failure Rate (/hr)
Oconee PRA	3.2E-6
Seabrook PSS	1.6E-5
IEEE-500	1.4E-6
Arithmetic Average	*6.8E-6*
Geometric Average	*4.1E-6*

Plant-Specific Evidence
Not available.

Value selected: **5.0E-6**

Rationale: Seabrook value is higher than other sources. Value selected is representative.

---

73. Temperature transmitter: output fails during operation

Generic Sources	Failure Rate (/hr)
Oconee PRA	5.7E-6
IEEE-500	1.6E-7
Arithmetic Average	*2.9E-6*
Geometric Average	*9.5E-7*

Plant-Specific Evidence

Not available.

Value selected: **1.0E-6**

Rationale: Limited data available, and sources are not very close. Value selected is representative.

---

74. Pressure switch: failure during operation

Generic Sources	Failure Rate (/hr)
Oconee PRA	3.4E-7
NPRD-2	9.8E-7
IEEE-500	7.0E-8
Arithmetic Average	*4.6E-7*
Geometric Average	*2.9E-7*

Plant-Specific Evidence

Not available.

Value selected: **3.0E-7**

Rationale: Limited sources available; value is reasonable, with greater weight given to nuclear plant sources.

---

75. Pressure switch: fails to respond on demand

Generic Sources	Failure Rate (/d)
Oconee PRA	2.4E-4
Seabrook PSS	2.7E-4
IEEE-500	1.4E-7
Arithmetic Average	*1.7E-4*
Geometric Average	*2.1E-5*

Plant-Specific Evidence

Not available.

Value selected: **2.0E-4**

Rationale: IEEE-500 data seems very low for demand failure rate. Other sources given more weight.

---

76. Level switch: failure during operation

Generic Sources	Failure Rate (/hr)
Oconee PRA	3.4E-7
NPRD-2	5.3E-6
IEEE-500	2.0E-7
Arithmetic Average	*1.9E-6*
Geometric Average	*7.1E-7*

Plant-Specific Evidence
Not available.

Value selected: **3.0E-7**

Rationale: NPRD-2 value is much higher than others, and reflects only non-nuclear experience (although the experience is substantial). Greater weight is given to the other sources.

---

77. Level switch: fails to respond on demand

Generic Sources	Failure Rate (/d)
Oconee PRA	2.4E-4
IEEE-500	3.3E-7
Arithmetic Average	*1.2E-4*
Geometric Average	*8.9E-6*

Plant-Specific Evidence
Not available.

Value selected: **1.0E-5**

Rationale: Very limited data available. IEEE-500 value again seems quite low for a demand failure rate, but both sources must be considered.

ANNEX A
RELIABILITY DATA BASE FOR ALWR PRAS

References

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ANNEX B
ALWR REFERENCE SITE

The ALWR reference site is expected to conservatively represent the consequences of most potential sites. Characteristics of 91 U.S. reactor sites are tabulated in the NRC document, *Technical Guidance for Siting Criteria Development* (NUREG CR-2239). Below are listed several of these characteristics which are correlated with high off-site consequences. The values for the ALWR reference site are shown, as well as the approximate percentile for the values:

PARAMETER	ALWR VALUE	PERCENTILE
Population density 0-200 miles	182/sq. mi.	80
Population density 0-20 miles	370/sq. mi.	90
Population center 5-10 miles	1600/sq. mi.	90
Population center 10-20 miles	2700/sq. mi.	95
Rainfall – hours annually	540 hours	80

The following ALWR "reference site" characteristics are required as input to the CRAC2 computer code:

- Meteorological Data (see Table A.B-1);
- Population Data (see Table A.B-2);
- Evacuation and Sheltering Data (see Table A.B-3).

Meteorological Data

CRAC2 requires a file of hourly meteorological data consisting of wind speed, wind direction, atmospheric stability category, and intensity of precipitation. A CRAC2 meteorological data file contains data for one year, which consists of 8760 entries for a 365-day year. The weather data assessment is done by sorting the file into weather categories. The categories must provide a realistic representation of the year's weather without overlooking those kinds of weather that are instrumental in producing major consequence impacts. A set of 29 weather categories has been selected for the CRAC2 model to reflect these requirements.

ANNEX B
ALWR REFERENCE SITE

The entire year of data, 8760 hourly recordings, are sorted into the 29 weather categories. Each sequence is examined to determine (1) the first occurrence of rain within 30 miles of the site, or (2) the first occurrence of a wind speed slowdown within 30 miles of the accident site, or (3) the stability category and wind speed at the start of the sequence. The first of these conditions that is satisfied by the sequence determines the weather category to which it is assigned. Following the assessment process, the start hour of each weather sequence will have been assigned to one and only one weather category. Each of the weather categories then includes a set of weather sequences representing the corresponding weather type. The probability of occurrence of that weather type is the ratio of the total number of weather sequences in the year's data set.

The sampling procedure now has two key items of information available to it: (1) the category of each weather sequence and (2) the probability of occurrence of each category of weather. A sample consists of a set of weather sequences selected from each of the categories. Four sequences are selected from each category by the "Latin hypercube" sampling scheme [1]. With this sampling method, random samples are drawn from sets evenly spaced within the weather category. This assures that the model uses an event representation of the weather data over the full year.

Rather than present the entire file in CRAC2 input format, the summary tables are attached for review. These tables give statistics for 29 bins derived from the 8760 hours of data.

Bins 1 through 7 represent cases where rain occurs over the distance intervals 0 (site), 0-5, 5-10, 10-15, 15-20, 20-25, and 25-30 miles, respectively.

Bins 8 through 12 represent cases where slowdowns (periods of low wind speed) occur over the distance intervals 0-10, 10-15, 15-20, 20-25, and 25-30 miles, respectively.

Bins 13 and 14 represent cases with stability class A, B, or C and initial wind speeds of $\leq$ 3 and $>$ 3 meters/sec, respectively.

Bins 15 through 19 represent cases with stability class D and initial wind speeds of $<$ 1, 1-2, 2-3, 3-5, and $>$ 5 meters/sec, respectively.

[1] Inman, R.L. and Conover, W.J. (1982) *Short Course on Sensitivity Analysis Techniques*, NUREG/CR-2350, SAND81-1978.

ANNEX B
ALWR REFERENCE SITE

Bins 20 and 24 represent cases with stability class E and initial wind speeds of < 1, 1-2, 2-3, 3-5, and > 5 meters/sec, respectively.

Bins 25 and 29 represent cases with stability class F and initial wind speeds of < 1, 1-2, 2-3, 3-5, and > 5 meters/sec, respectively.

All bins are further divided to provide statistics for the 16 different wind directions corresponding to 22.5-degree sectors. The first of these sectors is centered on due north, the second 22.5 degrees east of north, and so on.

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

METEOROLOGICAL DATA FILE CONTAINS 513 HOURS OF OBSERVED RAIN DATA.
ACCUMULATED RAIN MEASUREMENTS TOTALED 47.64 INCHES FOR THE YEAR.
HOLZWORTH AFTERNOON MIXING HEIGHT 1500 METERS.

* * * METEOROLOGICAL BIN SUMMARY * * *

BIN PRIORITIES
- R - RAIN WITHIN INTERVALS
- S - SLOWDOWNS WITHIN INTERVALS
- C D E F – STABILITY CATEGORIES
- 1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

WIND DIRECTION

METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
1 R 0	0.136	0.111	0.090	0.047	0.041	0.021	0.049	0.097	0.078	0.107	0.068	0.029	0.029	0.019	0.035	0.041	513	5.8562
2 R 5	0.114	0.086	0.043	0.014	0.114	0.029	0.100	0.114	0.057	0.086	0.029	0.043	0.057	0.029	0.071	0.014	70	0.7991
2 R 10	0.075	0.075	0.082	0.034	0.075	0.052	0.110	0.068	0.062	0.137	0.082	0.027	0.027	0.007	0.041	0.034	146	1.6667
4 R 15	0.076	0.101	0.076	0.059	0.067	0.042	0.084	0.092	0.109	0.118	0.050	0.050	0.008	0.017	0.017	0.034	119	1.3584
5 R 20	0.054	0.045	0.116	0.045	0.027	0.009	0.107	0.098	0.089	0.116	0.089	0.027	0.036	0.018	0.045	0.080	112	1.2785
6 R 25	0.080	0.070	0.090	0.060	0.040	0.070	0.080	0.110	0.140	0.100	0.070	0.020	0.010	0.010	0.030	0.020	100	1.1416
7 R 30	0.063	0.116	0.074	0.0	0.063	0.063	0.084	0.074	0.147	0.126	0.116	0.042	0.011	0.0	0.011	0.011	95	1.0845
8 S 10	0.085	0.136	0.068	0.051	0.0	0.017	0.0	0.0	0.017	0.153	0.051	0.051	0.017	0.119	0.102	0.136	59	0.6735
9 S 15	0.175	0.050	0.100	0.075	0.0	0.025	0.0	0.025	0.025	0.050	0.075	0.075	0.175	0.050	0.025	0.075	40	0.4566

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

BIN PRIORITIES

R - RAIN WITHIN INTERVALS
S - SLOWDOWNS WITHIN INTERVALS
C D E F – STABILITY CATEGORIES
1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

WIND DIRECTION

METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
10 S 20	0.184	0.041	0.061	0.0	0.0	0.020	0.0	0.020	0.061	0.122	0.041	0.061	0.061	0.020	0.184	0.122	49	0.5534
11 S 25	0.087	0.109	0.109	0.0	0.043	0.022	0.022	0.022	0.022	0.065	0.217	0.022	0.022	0.0	0.109	0.130	46	0.5251
12 S 30	0.058	0.038	0.038	0.0	0.058	0.019	0.0	0.358	0.038	0.135	0.096	0.058	0.038	0.019	0.135	0.212	52	0.5936
13 C 3	0.052	0.058	0.067	0.052	0.069	0.053	0.076	0.073	0.067	0.091	0.131	0.056	0.059	0.044	0.023	0.028	1126	12.8539
14 C 4	0.057	0.079	0.052	0.026	0.018	0.002	0.019	0.031	0.029	0.116	0.246	0.085	0.039	0.063	0.072	0.066	1136	12.9680
15 D 1	0.065	0.043	0.087	0.065	0.098	0.087	0.043	0.033	0.033	0.065	0.054	0.065	0.054	0.065	0.065	0.076	92	1.0502
16 D 2	0.056	0.062	0.074	0.052	0.076	0.064	0.048	0.052	0.095	0.110	0.099	0.064	0.056	0.027	0.031	0.035	484	5.5251
17 D 3	0.048	0.081	0.116	0.048	0.057	0.032	0.041	0.057	0.063	0.154	0.125	0.063	0.027	0.032	0.023	0.032	559	6.3813
18 D 4	0.128	0.137	0.117	0.023	0.011	0.0	0.007	0.036	0.069	0.200	0.144	0.034	0.016	0.007	0.029	0.040	554	6.3242
19 D 5	0.063	0.215	0.018	0.0	0.0	0.0	0.004	0.031	0.094	0.135	0.229	0.045	0.004	0.0	0.072	0.090	223	2.5457
20 E 1	0.123	0.058	0.076	0.035	0.058	0.064	0.076	0.047	0.070	0.047	0.088	0.076	0.041	0.047	0.047	0.047	171	1.9521
21 E 2	0.070	0.041	0.038	0.019	0.035	0.019	0.041	0.086	0.089	0.153	0.152	0.072	0.064	0.038	0.029	0.053	627	7.1575

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

BIN PRIORITIES

R - RAIN WITHIN INTERVALS
S - SLOWDOWNS WITHIN INTERVALS
C D E F – STABILITY CATEGORIES
1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

WIND DIRECTION

METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
22 E 3	0.135	0.055	0.045	0.022	0.012	0.010	0.015	0.075	0.077	0.185	0.137	0.055	0.050	0.035	0.035	0.057	401	4.5776
23 E 4	0.155	0.082	0.034	0.010	0.007	0.0	0.0	0.082	0.103	0.258	0.117	0.021	0.003	0.0	0.052	0.076	291	3.3219
24 E 5	0.081	0.210	0.032	0.0	0.0	0.0	0.016	0.032	0.355	0.145	0.065	0.016	0.0	0.0	0.032	0.016	62	0.7078
25 F 1	0.078	0.073	0.065	0.039	0.057	0.035	0.071	0.043	0.092	0.082	0.086	0.057	0.065	0.049	0.057	0.051	510	5.8219
26 F 2	0.103	0.057	0.021	0.006	0.013	0.005	0.025	0.057	0.112	0.149	0.169	0.113	0.072	0.042	0.021	0.034	793	9.0525
27 F 3	0.107	0.020	0.008	0.004	0.004	0.0	0.0	0.055	0.091	0.154	0.154	0.091	0.134	0.059	0.043	0.075	253	2.8881
28 F 4	0.213	0.115	0.016	0.016	0.0	0.0	0.0	0.0	0.213	0.016	0.049	0.0	0.033	0.082	0.098	0.148	61	0.6963
29 F 5	0.0	0.250	0.0	0.0	0.0	0.0	0.0	0.0	0.250	0.500	0.0	0.0	0.0	0.0	0.0	0.0	16	0.1826
30 ALL	0.085	0.078	0.063	0.031	0.037	0.024	0.040	0.059	0.079	0.131	0.138	0.062	0.046	0.036	0.041	0.050	8760	

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

* * * METEOROLOGICAL BIN SUMMARY * * * (Page 4 of 7)

BIN PRIORITIES

R - RAIN WITHIN INTERVALS
S - SLOWDOWNS WITHIN INTERVALS
C D E F – STABILITY CATEGORIES
1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

	WIND DIRECTION																
METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
1 R 0	70	57	46	24	21	11	25	50	40	55	35	15	15	10	18	21	513	5.8562
2 R 5	8	6	3	1	8	2	7	8	4	6	2	3	4	2	5	1	70	0.7991
3 R 10	11	11	12	5	11	9	16	10	9	20	12	4	4	1	6	5	146	1.6667
4 R 15	9	12	9	7	8	5	10	11	13	14	6	6	1	2	2	4	119	1.3584
5 R 20	6	5	13	5	3	1	12	11	10	13	10	3	4	2	5	9	112	1.2785
6 R 25	8	7	9	6	4	7	8	11	14	10	7	2	1	1	3	2	100	1.1416
7 R 30	6	11	7	0	6	6	8	7	14	12	11	4	1	0	1	1	95	1.0845
8 S 10	5	8	4	3	0	1	0	0	1	9	3	3	1	7	6	8	59	0.6735
9 S 15	7	2	4	3	0	1	0	1	1	2	3	3	7	2	1	3	40	0.4566
10 S 20	9	2	3	0	0	1	0	1	3	6	2	3	3	1	9	6	49	0.5594
11 S 25	4	5	5	0	2	1	1	1	1	3	10	1	1	0	5	6	46	0.5251

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

BIN PRIORITIES

R - RAIN WITHIN INTERVALS
S - SLOWDOWNS WITHIN INTERVALS
C D E F – STABILITY CATEGORIES
1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

WIND DIRECTION

METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
12 S 30	3	2	2	0	3	1	0	3	2	7	5	3	2	1	7	11	52	0.5936
13 C 13	59	65	76	58	78	60	86	82	75	103	148	63	66	49	26	32	1126	12.8539
14 C 4	65	90	59	29	21	2	22	35	33	132	279	97	44	71	82	75	1136	12.9680
15 D 1	6	4	8	6	9	8	4	3	3	6	5	6	5	6	6	7	92	1.0502
16 D 2	27	30	36	25	37	31	23	25	46	53	48	31	27	13	15	17	484	5.5251
17 D 3	27	45	65	27	32	18	23	32	35	86	70	35	15	18	13	18	559	6.3813
18 D 4	71	76	65	13	6	0	4	20	38	111	80	19	9	4	16	22	554	6.3242
19 D 5	14	48	4	0	0	0	1	7	21	30	51	10	1	0	16	20	223	2.5457
20 E 1	21	10	13	6	10	11	13	8	12	8	15	13	7	8	8	8	171	1.9521
21 E 2	44	26	24	12	22	12	26	54	56	96	95	45	40	24	18	33	627	7.1575
22 E 3	54	22	18	9	5	4	6	30	31	74	55	22	20	14	14	23	401	4.5776
23 E 4	45	24	10	3	2	0	0	24	30	75	34	6	1	0	15	22	291	3.3219

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

BIN PRIORITIES

R - RAIN WITHIN INTERVALS
S - SLOWDOWNS WITHIN INTERVALS
C D E F – STABILITY CATEGORIES
1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

WIND DIRECTION

METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
24 E 5	5	13	2	0	0	0	1	2	22	9	4	1	0	0	2	1	62	0.7078
25 F 1	40	37	33	20	29	18	36	22	47	42	44	29	33	25	29	26	510	5.8219
26 F 2	82	45	17	5	10	4	20	45	89	118	134	90	57	33	17	27	793	9.0525
27 F 3	27	5	2	1	1	0	0	14	23	39	39	23	34	15	11	19	253	2.8881
28 F 4	13	7	1	1	0	0	0	0	13	1	3	0	2	5	6	9	61	0.6963
29 F 5	0	4	0	0	0	0	0	0	4	8	0	0	0	0	0	0	16	0.1826

TABLE A.B-1.
CRAC2 METEOROLOGICAL BIN SUMMARY

BIN PRIORITIES

R - RAIN WITHIN INTERVALS
S - SLOWDOWNS WITHIN INTERVALS
C D E F – STABILITY CATEGORIES
1 (0-1), 2 (1-2), 3 (2-3), 4 (3-5), 5 (GT 5) – WIND SPEED INTERVALS (M/S)

WIND DIRECTION

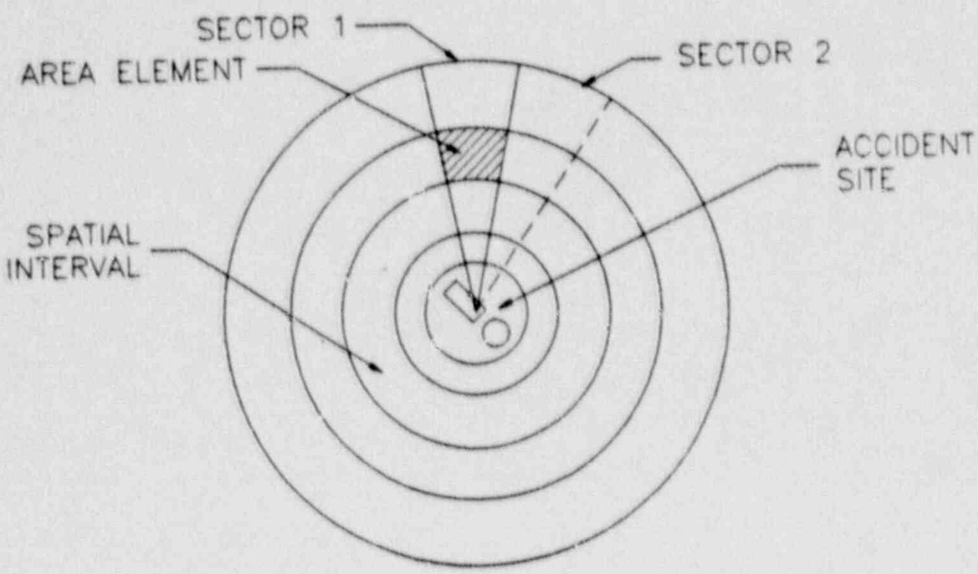
METBIN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	TOTAL	PERCENT
* * * SUMMARIES * * *																		
R	118	109	99	48	61	41	86	108	104	130	83	37	30	18	40	43	1155	13.1849
S	28	19	18	6	5	5	1	6	8	27	23	13	14	11	28	34	246	2.8082
C	124	155	135	87	99	62	108	117	108	235	427	160	110	120	108	107	2262	25.8219
D	145	203	178	71	84	57	55	87	143	286	254	101	57	41	66	84	1912	21.8265
E	169	95	67	30	39	27	46	113	151	262	203	87	68	46	57	87	1552	17.7169
F	162	98	53	27	40	22	56	81	176	208	220	142	125	78	63	81	1633	18.6415
1	70	51	55	34	53	38	57	34	64	57	69	50	47	41	43	41	804	9.1781
2	174	114	104	63	107	75	107	163	230	301	306	184	146	92	56	88	2310	26.3699
3	143	124	133	72	73	53	73	118	123	267	278	123	111	72	58	81	1902	21.7123
4	174	163	130	42	27	2	24	76	105	296	323	101	49	60	75	92	1739	19.8516
5	39	99	11	4	2	0	4	12	56	70	128	32	8	20	62	57	604	6.8950

ANNEX B
ALWR REFERENCE SITE

Population Data

The population data which describes the ALWR reference site is contained in the Site Data file. The population distribution around the reactor site was assigned to elements of a grid defined by sixteen 22.5-degree sectors and thirty-four annuli. The first of these sectors is centered on due north, the second 22.5 degrees east of north, and so on. These directions correspond to the wind rose generated from the meteorological file, with the wind blowing **toward** the given directions. The annuli have the following radii in miles: 0.47, 1.0, 2.0, 3.0, 4.0, 5.0, 6.0, 7.0, 8.0, 9.0, 10.0, 11.0, 12.0, 13.0, 14.0, 15.0, 16.0, 17.0, 18.0, 19.0, 20.0, 30.0, 40.0, 50.0, 55.0, 60.0, 65.0, 70.0, 85.0, 100.0, 150.0, 200.0, 350.0, 500.0.

Attached is the population distribution for the ALWR reference site. Information on format can be obtained from the CRAC2 Computer Code Users Manual.



Representation of the CRAC2 Geometry

TABLE A.B-2.
ALWR CRAC2 REFERENCE SITE — POPULATION DATA

Sector Distance Intervals (miles)	#1	#2	#3	#4	#5	#6	#7	#8
0.0—0.47	0	0	0	0	0	0	0	0
0.47—1.0	3	6	0	3	3	15	0	0
1.0—2.0	44	30	35	41	11	75	27	7
2.0—3.0	76	31	38	19	50	935	229	256
3.0—4.0	819	113	70	89	156	566	726	465
4.0—5.0	435	461	100	139	219	146	413	777
5.0—6.0	255	161	178	71	376	300	406	1279
6.0—7.0	223	189	173	87	140	603	2025	4563
7.0—8.0	237	188	52	59	638	2762	414	6780
8.0—9.0	435	377	25925	25409	472	2188	254	4277
9.0—10.0	537	542	1054	257	1108	852	255	6276
10.0—11.0	731	704	1587	1634	1156	216	661	2530

TABLE A.B-2.
ALWR CRAC2 REFERENCE SITE — POPULATION DATA

Sector Distance intervals (miles)	#1	#2	#3	#4	#5	#6	#7	#8
11.0—12.0	2305	783	2160	5760	2508	525	752	1300
12.0—13.0	4946	1588	4516	8019	2037	556	503	697
13.0—14.0	7747	2001	8474	9310	399	577	935	431
14.0—15.0	5996	2542	15120	10564	205	224	1738	771
15.0—16.0	6818	2955	17177	8195	436	417	217	304
16.0—17.0	6422	5506	21995	12552	2217	444	231	323
17.0—18.0	2761	4247	22467	12366	1729	471	245	343
18.0—19.0	3071	3052	23250	12254	783	497	260	362
19.0—20.0	1717	2452	23709	12438	1101	524	274	382
20.0—30.0	29136	25042	143872	104941	56858	18654	51951	2771
30.0—40.0	27439	39969	132594	21792	42640	14732	30022	15879
40.0—50.0	48856	40643	64239	24214	17771	20822	19065	3685

TABLE A.B-2.
ALWR CRAC2 REFERENCE SITE – POPULATION DATA

Sector Distance Intervals (miles)	#1	#2	#3	#4	#5	#6	#7	#8
50.0 – 55.0	52079	45879	72858	47698	12162	7242	1954	790
55.0 – 60.0	25051	19981	40315	21113	18059	9587	5288	19660
60.0 – 65.0	24084	19444	18256	8228	9979	11453	7197	46008
65.0 – 70.0	11886	22036	56997	10456	6983	14747	10546	6927
70.0 – 85.0	121342	213636	238550	70567	97396	70866	99108	71870
85.0 – 100.	37489	328113	556800	79135	94778	86191	211826	135627
100. – 150.	329656	430709	907321	1215270	801702	447183	278209	284248
150. – 200.	656250	965756	328122	780078	594809	377605	140354	738702
200. – 350.	1425219	860867	3388006	1565834	368272	2738	0	0
350. – 500.	7457921	2880548	11226251	17599	0	0	0	0

TABLE A.B-2.
ALWR CRAC2 REFERENCE SITE – POPULATION DATA

Sector Distance Intervals (miles)	#9	#10	#11	#12	#13	#14	#15	#16
0.0 – 0.47	0	0	0	0	0	0	0	0
0.47 – 1.0	9	0	3	6	3	0	0	0
1.0 – 2.0	11	31	0	0	15	68	61	30
2.0 – 3.0	113	236	73	39	0	15	27	30
3.0 – 4.0	290	265	184	39	60	69	36	119
4.0 – 5.0	595	392	85	39	90	74	262	80
5.0 – 6.0	834	386	126	130	103	100	180	152
6.0 – 7.0	2156	607	271	157	120	145	163	279
7.0 – 8.0	2317	432	201	115	140	255	333	350
8.0 – 9.0	3278	105	260	205	275	498	290	343
9.0 – 10.0	4199	353	110	2146	375	2263	238	215
10.0 – 11.0	2479	530	160	3135	320	2037	150	3232

TABLE A.B-2.
ALWR CRAC2 REFERENCE SITE – POPULATION DATA

Sector / Distance Intervals (miles)	#9	#10	#11	#12	#13	#14	#15	#16
11.0 – 12.0	1053	220	225	1427	389	171	451	2241
12.0 – 13.0	629	175	250	340	346	230	1567	2046
13.0 – 14.0	512	215	190	197	215	290	1265	7624
14.0 – 15.0	331	177	155	133	200	339	2111	11128
15.0 – 16.0	257	325	116	247	225	107	1507	13046
16.0 – 17.0	274	345	124	263	239	114	1465	15289
17.0 – 18.0	290	366	132	279	254	121	2517	7189
18.0 – 19.0	307	387	139	295	269	127	1694	4992
19.0 – 20.0	323	408	147	310	283	134	8411	3369
20.0 – 30.0	4453	37878	5618	3593	14417	34231	47823	35411
30.0 – 40.0	4145	3906	35154	16059	59503	75906	29496	56468
40.0 – 50.0	19643	5506	17736	44895	126121	54872	16930	113123

TABLE A.B-2.
ALWR CRAC2 REFERENCE SITE – POPULATION DATA

Sector Distance Intervals (miles)	#9	#10	#11	#12	#13	#14	#15	#16
50.0 – 55.0	11545	1900	2808	17413	119787	33333	6987	94554
55.0 – 60.0	9375	9720	10567	22609	42633	19489	48768	70357
60.0 – 65.0	31158	36735	44829	9934	43459	10529	27050	52473
65.0 – 70.0	227613	16251	24852	35436	76259	8241	41715	43795
70.0 – 85.0	522468	53220	72841	234790	361906	142008	49147	26493
85.0 – 100.	55514	41546	88142	239710	133399	327358	62105	98301
100. – 150.	266650	746697	145073	458692	176912	347401	788962	487580
150. – 200.	289005	236081	264759	1505036	273317	1346685	497565	662998
200. – 350.	1039589	620871	1097589	3070908	1631176	1364143	2600059	1976559
350. – 500.	2698919	673150	1081859	1260638	1435912	1629589	2988924	5122181

ANNEX B
ALWR REFERENCE SITE

Evacuation and Sheltering Data

The ALWR off-site consequences analysis requires six distinct evacuation schemes in order to adequately represent evacuation time estimates for the permanent resident population, the transient population, and the special facility population (schools, hospitals, etc.). The evacuation data includes an evacuation scheme that assumes 5 percent of the population would delay evacuation for 24 hours after being warned to evacuate. This very conservative assumption is used so that the ALWR risk estimates can be compared with the IDCOR and NUREG-1150 analyses which both use this assumption.

Cloud and ground shielding factors are based on information given in WASH-1400. Breathing rate data is obtained from the PRA Procedures Guide.

Table A.B-3.
Evacuation and Sheltering Data (Normal)

(Page 1 of 3)

EVCONI(1.1)	PROBABILITY OF STRATEGY (0-1)	5.000E−02
EVCONI(2.1)	TIME DELAY BEFORE EVACUATION (HRS)	2.400E+01
EVCONI(3.1)	EVACUATION SPEED (M/S)	4.470E+00
EVCONI(4.1)	MAXIMUM DISTANCE OF EVACUATION (M)	1.609E+04
EVCONI(5.1)	DISTANCE MOVED BY EVACUEES (M)	3.219E+04
EVCONI(6.1)	SHELTERING RADIUS (M)	8.047E+04
EVCONI(7.1)	EVACUATION SCHEME (1 OR 2)	2.000E+00
EXPD(1)	EXPOSURE DURATION (DAYS)	1.000E+00
EVCONI(1.2)	PROBABILITY OF STRATEGY (0-1)	5.600E−01
EVCONI(2.2)	TIME DELAY BEFORE EVACUATION (HRS)	1.000E+00
EVCONI(3.2)	EVACUATION SPEED (M/S)	4.470E+00
EVCONI(4.2)	MAXIMUM DISTANCE OF EVACUATION (M)	1.609E+04
EVCONI(5.2)	DISTANCE MOVED BY EVACUEES (M)	3.219E+04
EVCONI(6.2)	SHELTERING RADIUS (M)	8.047E+04
EVCONI(7.2)	EVACUATION SCHEME (1 OR 2)	2.000E+00
EXPD(2)	EXPOSURE DURATION (DAYS)	1.000E+00
EVCONI(1.3)	PROBABILITY OF STRATEGY (0-1)	3.400E−01
EVCONI(2.3)	TIME DELAY BEFORE EVACUATION (HRS)	1.500E+00
EVCONI(3.3)	EVACUATION SPEED (M/S)	4.470E+00
EVCONI(4.3)	MAXIMUM DISTANCE OF EVACUATION (M)	1.609E+04
EVCONI(5.3)	DISTANCE MOVED BY EVACUEES (M)	3.219E+04
EVCONI(6.3)	SHELTERING RADIUS (M)	8.047E+04
EVCONI(7.3)	EVACUATION SCHEME (1 OR 2)	2.000E+00
EXPD(3)	EXPOSURE DURATION (DAYS)	1.000E+00

Table A.B-3.
Evacuation and Sheltering Data (Normal)

(Page 2 of 3)

EVCONI(1.4)	PROBABILITY OF STRATEGY (0-1)	3.000E−02
EVCONI(2.4)	TIME DELAY BEFORE EVACUATION (HRS)	2.000E+00
EVCONI(3.4)	EVACUATION SPEED (M/S)	4.470E+00
EVCONI(4.4)	MAXIMUM DISTANCE OF EVACUATION (M)	1.609E+04
EVCONI(5.4)	DISTANCE MOVED BY EVACUEES (M)	3.219E+04
EVCONI(6.4)	SHELTERING RADIUS (M)	8.047E+04
EVCONI(7.4)	EVACUATION SCHEME (1 OR 2)	2.000E+00
EXPD(4)	EXPOSURE DURATION (DAYS)	1.000E+00
EVCONI(1.5)	PROBABILITY OF STRATEGY (0-1)	1.000E−02
EVCONI(2.5)	TIME DELAY BEFORE EVACUATION (HRS)	2.500E+00
EVCONI(3.5)	EVACUATION SPEED (M/S)	4.470E+00
EVCONI(4.5)	MAXIMUM DISTANCE OF EVACUATION (M)	1.609E+04
EVCONI(5.5)	DISTANCE MOVED BY EVACUEES (M)	3.219E+04
EVCONI(6.5)	SHELTERING RADIUS (M)	8.047E+04
EVCONI(7.5)	EVACUATION SCHEME (1 OR 2)	2.000E+00
EXPD(5)	EXPOSURE DURATION (DAYS)	1.000E+00
EVCONI(1.6)	PROBABILITY OF STRATEGY (0-1)	1.000E−02
EVCONI(2.6)	TIME DELAY BEFORE EVACUATION (HRS)	3.000E+00
EVCONI(3.6)	EVACUATION SPEED (M/S)	4.470E+00
EVCONI(4.6)	MAXIMUM DISTANCE OF EVACUATION (M)	1.609E+04
EVCONI(5.6)	DISTANCE MOVED BY EVACUEES (M)	3.219E+04
EVCONI(6.6)	SHELTERING RADIUS (M)	8.047E+04
EVCONI(7.6)	EVACUATION SCHEME (1 OR 2)	2.000E+00
EXPD(6)	EXPOSURE DURATION (DAYS)	1.000E+00

Table A.B-3.
Evacuation and Sheltering Data (Normal)

(Page 3 of 3)

SHFAC(1.1)	CLOUD SHIELDING — STATIONARY PEOPLE	8.300E−01
SHFAC(2.1)	CLOUD SHIELDING — MOVING EVACUEES	8.300E−01
SHFAC(3.1)	CLOUD SHIELDING — SHELTERING	7.100E−01
SHFAC(4.1)	CLOUD SHIELDING — NO EMERGENCY ACTION	7.400E−01
SHFAC(1.2)	GROUND SHIELDING — STATIONARY PEOPLE	4.300E−01
SHFAC(2.2)	GROUND SHIELDING — MOVING EVACUEES	4.300E−01
SHFAC(3.2)	GROUND SHIELDING — SHELTERING	2.500E−01
SHFAC(4.2)	GROUND SHIELDING — NO EMERGENCY ACTION	3.100E−01
BRATE(1)	BREATHING RATE STATIONARY EVACUEES	2.660E−04
BRATE(2)	BREATHING RATE MOVING EVACUEES	2.660E−04
BRATE(3)	BREATHING RATE SHELTERING REGION ONE	1.330E−04
BRATE(4)	BREATHING RATE SHELTERING REGION TWO	2.660E−04
EVCOST(1)	RADIUS OF CIRCULAR AREA EVAC NEAR REACTOR	1.609E+04
EVCOST(2)	WIDTH OF EVACUATED ARC (DEGREES)	9.000E+01
EVCOST(3)	EVACUATION DIRECT COST (3/EVACUEE/DAY)	1.650E+02
EVCOST(4)	MAX DURATION OF RELEASE FOR KEY SHAPED EVAC	3.000E+00
IEXPD	DURATION OF EXPOSURE SWITCH	1