Advanced Light Water Reactor Program

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Program Management and Staff Review Methodology

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ABSTRACT

This report summarizes the NRC/EPRI coordinated effort to develop design requirements for a standardized advanced light water reactor (ALWR) and the procedures for screening and applying new generic safety issues to this program. The end-product will be an NRC-approved ALWR Requirements Document for use by the nuclear industry in generating designs of LWRs to be constructed for operation in the 1990s and beyond.

ACKNOWLEDGEMENT

The Advanced Light Water Reactor (ALWR) Program has been essentially from inception a closely coordinated effort of the Electric Power Research Institute (EPRI) and the Nuclear Regulatory Commission (NRC) staff. The ALWR Program management and staff review methodology have evolved through efforts of the EPRI staff, NRC staff and particularly through the constant attention and technical management capability of: (1) the EPRI project managers: Dr. Robert E. Nickell, Warren J. Bilanin, and Daniel N. Noble; (2) the consultant to EPRI, Ronald E. Engel of S. Levy Incorporated; and (3) the NRC staff member, Ronald E. Emrit of the Office of Nuclear Reactor Regulation.

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AIF	Atomic Industrial Forum
ALWR	advanced light water reactor
APTS	Three Mile Island Action Plan Tracking System
BOP	balance of plant
BWR	boiling water reactor
DBE	design-basis earthquake
DOE	Department of Energy
EPRI	Electric Power Research Institute
GIMCS	Generic Issue Management Control System
GITS	Generic Issue Tracking System
IE	Office of Inspection and Enforcement, NRC
INPO	Institute for Nuclear Power Operations
LOCA	loss-of-coolant accident
LWR	light water reactor
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation, NRC
NSSS	nuclear steam supply system
OL	operating license
OSTG	once-through steam generator
PWR	pressurized water reactor
SER	Safety Evaluation Report
SRP	Standard Review Plan
SRV	safety/relief valve
SSW	system service water
TMI	Three Mile Island
USI	unresolved safety issue

EXECUTIVE SUMMARY

To address the need for a viable nuclear power generation option, the Electric Power Research Institute (EPRI), as directed by its ALWR Utility Steering Committee, embarked on the EPRI/Industry Advanced Light Water Reactor (ALWR) Program. It is EPRI's objective for the ALWR Program to develop improved LWR designs and to work with the staff in this closely coordinated effort to bring about stabilized regulatory and utility industry requirements for future LWR plant designs.

The output of the ALWR Program includes a utility-reviewed and Nuclear Regulatory Commission (NRC) approved Requirements Document for future nuclear power plants. The Requirements Document is to consist of 13 chapters and is to be applicable to both PWRs and BWRs of 400 MWe to 1350 MWe. The Requirements Document will define both licensing and utility requirements for a facility that is: (1) less subject to accidents; (2) more tolerant of operating transients; (3) simplified compared to current plants with respect to constructability, maintainability, and operability; (4) based on fully proven technology; (5) lower in lifetime costs than current plants; and (6) reduce, relative to previous plants, the probability and consequences of accidents that could endanger the safety of in-plant personnel or the general public, cause radioactive release, or damage plant equipment. It is intended that the ALWR Requirements Document will be prepared by EPRI and reviewed by both the utilities and the nuclear industry, and that the licensing requirement aspects will be approved by the NRC.

The following are considered to be the major components of the program management and review methodology for the ALWR Program.

- (1) The ALWR Program provides a mechanism for establishing a standard baseline of regulatory and industry requirements for the total plant in the Requirements Document which appropriately address current regulatory requirements including resolution of applicable generic safety and licensing issues. The licensing process for a plant utilizing the Requirements Document primarily consists of a demonstration that the plant requirements have been appropriately implemented at an acceptable site and considering factors beyond the scope of the Requirements Document.
- (2) A process for treating current and new generic safety issues has been implemented consistent with the intent of establishing the foundation of regulatory requirements for the ALWR. This process is consistent with NUREG-0933 and is closely coordinated with pertinent procedures in NUREG-0933. As a result, two separate and distinct processes have been developed.
 - (a) For current issues, the resolution proposed for all applicable generic safety and licensing issues is included in the Requirements Document.
 - (b) For issues identified after July 1, 1986, appropriate requirements will be identified after their resolution if one or more of the following criteria are satisfied.

- (ii) Would the offsite radiological consequences dose requirements established in the Requirements Document be exceeded as a result of this issue?
 - (iii) Would the Commission's safety goals be exceeded as a result of this issue?
- (3) Technically supportable alternatives to current regulatory requirements, called "plant optimization subjects," have been identified for application to the ALWR Program. These alternatives are proposed to achieve the program goals and are identified in the Requirements Document. Review of the plant optimization subjects is considered a part of the Requirements Document review.
- (4) The NRC staff review of the Requirements Document will be documented in an SER. Assuming the Requirements Document meets staff approval, that approval will include the following: The staff has reviewed the Requirements Document and finds that it contains the necessary requirements that, if properly translated into a design in accordance with current practice and licensing guidance, it will generate a nuclear power plant design which will have all the attributes required by NRC regulations to assure there is no undue risk to the public health and safety as required by the regulations. The SER will, where appropriate, state that the Requirements Document has provided resolution for generic safety and licensing issues consistent with the level of detail provided. The SER will also document the acceptability of alternative approaches to current regulatory requirements identified in the Requirements Document. The SER will provide that approval, including assurance that non-safety subjects such as "constructability" are not in conflict with regulations, and therefore are acceptable.
- (5) The schedule for EPRI submittal and NRC review of the Requirements Document is shown in Table 1-2.

1 INTRODUCTION

To address the need for a viable nuclear power generation option, the Electric Power Research Institute (EPRI) has embarked on an EPRI/Industry Advanced Light Water Reactor Program in close coordination with the NRC. It is the objective of the ALWR Program to promote a standardized design that is easier to construct and license, simpler in design, easier to operate and maintain, and more reliable than are current nuclear power plants. In addition, the ALWR Program is focused on the use of more passive safety features to minimize investor risk and improve plant safety.

The end-product of the EPRI/Industry ALWR Program is a Requirements Document for future nuclear power plants that is to be reviewed by the nuclear industry and transmitted to the NRC staff for approval. The Requirements Document is to consist of 13 chapters and is to be applicable to both PWRs and BWRs of 400 MWe to 1350 MWe.

The Requirements Document is intended to define a licensable, complete plant based on fully proven technology, when compared with current plants, that is: (1) greatly simplified with respect to constructability, maintainability, and operability; (2) less subject to accidents; (3) more stable during operating transients; and (4) lower in lifetime costs.

The purpose of this report is to document the ALWR Program management and staff review methodology. This documentation will provide an introduction and agency memory of the NRC review and approval process as it will be applied to the EPRI/Industry ALWR Program. It should be recognized that the ALWR Program represents a novel approach to nuclear power plant specification, design, and construction in that a comprehensive Requirements Document will be used to establish the detailed plant requirements. In addition, it is intended that the Requirements Document be reviewed by the NRC staff to establish at the level of detail provided, that the applicable regulatory requirements can be satisfied by proper implementation and that an appropriate resolution of all applicable unresolved safety issues and that high- and medium-priority generic safety issues have been incorporated. Also, procedures have been developed to address applicability of new generic safety issues to the ALWR. Such an approach should lead to a high degree of regulatory stability.

Therefore, because of the unique approach being taken by the ALWR Program, it is necessary to develop and document a review process consistent with the goals of this program.

1.1 Preparation and Review of the ALWR Requirements Document

An overview of the Requirements Document preparation and review process for the EPRI/Industry ALWR Program is shown in Figure 1-1. This process has evolved under the guidance and direction of the NRC Policy Committee, the ALWR Utility Steering Committee, and EPRI.

The staff review of the ALWR Program is focused on the Requirements Document. In addition, the staff has the opportunity to provide regulatory insights to the Requirements Document at each key development stage.

As shown in Figure 1-1, the primary set of inputs to the Requirements Document is comprised of the current regulatory requirements and guidance. Another source of regulatory inputs is the requirements established by the resolution of generic safety and licensing issues. A final source of inputs is the proposed optimization of current regulatory requirements supportable by current technology. The Requirements Document uses current NRC requirements as a reference point. The primary regulatory requirements and guidance are documented in the applicable regulations (Code of Federal Regulations, Title 10, "Energy") along with additional guidance provided by the Standard Review Plan (NUREG-0800) and regulatory guides. Other regulatory guidance may be contained in Generic Letters, Office of Inspection and Enforcement Bulletins, and NRC reports in the NUREG series.

The resolution of generic safety and licensing issues provides an important set of inputs to the Requirements Document. As shown in Figure 1-1, there are basically three steps that are performed in the treatment of generic safety and licensing issues in developing inputs to the Requirements Document: (1) the identification of all generic safety and licensing issues; (2) an evaluation to determine issue applicability and priority; and (3) the resolution of applicable issues. These steps apply to both current and future generic safety and licensing issues. The process begins with the NRC identification and definition of generic safety and licensing issues from all available sources. These issues are prioritized to establish their safety significance; applicability of an issue to the ALWR Program is established by the staff. Work continues on the applicable unresolved generic safety issues (USIs) until acceptable resolutions are identified. The resolution may serve all plants (generic) or only specific plants. The effort to resolve issues for the ALWR Program is applicable only to plants designed and constructed in accordance with the ALWR Requirements Document. The resolutions of issues applicable to the ALWR are incorporated into the Requirements Document as appropriate. During the issue-resolution process, there is a high degree of interaction between EPRI and the staff to ensure that the resolution's applicability to the ALWR is clearly understood.

A similar process is followed for identifying and assessing acceptable approaches to plant optimization subjects. Plant optimization subjects are first identified by EPRI as technically supportable alternatives to current regulatory requirements that represent optimization of plant design, construction, and operation. Potential plant optimization subjects are proposed by EPRI and reviewed by the staff for applicability to the ALWR Program. Plant optimization subjects that satisfy the EPRI criteria for the ALWR Program and are found acceptable by the staff are incorporated into the Requirements Document as appropriate.

The ALWR Requirements Document consists of 13 chapters, each addressing a different aspect of the plant. The staff began its review of the ALWR Requirements Document in the third quarter of calendar year (CY) 1986 with the receipt of Chapter 1, "Overall Requirements." The staff will continue its review on a chapter-by-chapter basis for approximately a two-and-one-half-year period, during which the Requirements Document is planned to be submitted (see Table 1-1). The staff will document its review by means of a single SER that has individual

chapters (or sections) addressing each of the 13 chapters of the Requirements Document. The SER that reports on each chapter will contain statements about the acceptability of each chapter. During the review, each chapter will be treated in a final draft SER chapter (or section). After all chapters have been reviewed, a final SER will be issued as the staff's end-product covering the review of the entire Requirements Document.

1.2 Interaction With NRC Tracking Programs for Generic Safety and Licensing Issues

The EPRI/Industry ALWR Program utilizes existing NRC programs to identify and resolve generic safety and licensing issues. Two documents are the primary interfaces of the program: "A Prioritization of Generic Safety Issues" (NUREG-0933) and the "Generic Issues Management Control System" (GIMCS) (NRC memorandum dated April 24, 1986).

NUREG-0933 provides for the identification and prioritization of all generic safety and licensing issues considering all plants, both operating and under construction. Because this report provides the most comprehensive discussion of generic safety and licensing issues, it is a key document used in the identification of remaining and resolved issues requiring action in the ALWR Program. Each of the current issues identified in NUREG-0933 is considered in the ALWR Program, and the issue priority is consistent with that identified in NUREG-0933. To aid in establishing issue applicability to the ALWR Program, Appendix B has been added to NUREG-0933 which identifies applicability to nuclear steam supply system vendors and to operating and future plants.

To make the process for treating new issues more effective, a set of screening and implementation criteria is used to process new issues before any new issue is considered in the ALWR Program. The screening criteria focus the intent of new issues on potential safety concerns rather than on research-related, environmental, or regulatory improvement issues and eliminate the reintroduction of issues that have been considered previously under other titles. The implementation criteria are intended to identify those safety issues that represent a substantial improvement in plant safety and have a favorable risk improvement/cost-benefit relationship. July 1, 1986, marks the beginning of formal application of the screening and implementation criteria.

The "Generic Issue Management Control System" (GIMCS) is published quarterly by NRC and provides the information necessary to manage the technical resolution and completion of generic safety issues. The system is used by the ALWR Program to provide the current status of remaining issues and completion status. This information is then used by EPRI to aid in establishing the resources required to develop the elements of resolution to be used as input to the ALWR Requirements Document to formally resolve the issues with respect to the ALWR.

It should be noted that a number of other NRC reports* and tracking systems** provide information on generic safety and licensing issues. These systems, along with NUREG-0933 and GIMCS, provide a comprehensive set of information regarding generic safety and licensing issues. It is this information together with NRC reports on specific issues that forms the primary basis for the acceptable resolution of generic safety and licensing issues and for the current activities under way to resolve the remaining issues applicable to the ALWR.

1.3 Applicability to Future Plants

In the "NRC Policy on Future Reactor Designs" (NUREG-1070, p.9) which recognizes the continuing work between the NRC and EPRI, it is stated:

It is assumed in this Policy Statement that, over the next 10 to 15 years, utility and commercial interest in the United States will focus on advanced light water reactors that involve improvements but are essentially based on the technology that was demonstrated in the design, construction, and operation of more than 100 of these plants in the United States. This policy should not be viewed as prejudicial to more extensive changes in reactor designs that might be demonstrated during or beyond that time period. Indeed, the Commission encourages the development and commercialization of any standard designs that might realize safety benefits, such as those achieved through greater simplicity; slower dynamic response to upset conditions involving accident precursor events; passive heat removal for loss-of-coolant accidents; and other characteristics that promote more efficient construction, operation, and maintenance procedures to enhance safety, reliability, and economy.

The current EPRI approach to the ALWR is consistent with NRC policy. NRC will review the EPRI Requirements Document to ensure that appropriate requirements are established for resolving all applicable unresolved safety and licensing issues and high- and medium-priority generic safety issues, as well as the other requirements established as a result of NRC policy.

Therefore, the NRC review of a reactor of the future designed in accordance with the EPRI Requirements Document will be limited to a review of the proposed designs to confirm that the requirements have been appropriately translated into the design. No further NRC action will be taken with regard to the application of additional generic safety issues without appropriate determinations in accordance with the implementation criteria described in Section 3.1 of this NUREG report.

*NUREGs-0371, -0410, -0471, -0606, -0660, -0737, and -0985.

**"APTS - Three Mile Island Action Plan Tracking System," August 30, 1983 (available from NRC Office of Resource Management), and "GITS - Generic Issues Tracking System," September 1, 1982 (available from NRC Office of Nuclear Reactor Regulation, Safety Program Evaluation Branch).

Figure 1-1 Overview of NRC review process for the ALWR

Table 1-1 Staff review schedule for
ALWR Requirements Document

Chapter	Date to be transmitted	Date review to be completed
1	July 1986	January 1987
2	September 1986	March 1987
3, 4, & 5	June 1987	December 1987
6	December 1987	June 1988
7-12	March 1988	September 1988
13	September 1988	March 1989
1-13	-	June 1989

2 BACKGROUND AND STRUCTURE OF THE ALWR PROGRAM

2.1 Organizational Interfaces

The key organizational interfaces for the ALWR Program are shown in Figure 2-1. The primary interface is between the NRC Policy Committee and the ALWR Utility Steering Committee. The primary technical interface is between the NRC staff and EPRI. More functional interfaces exist between EPRI and other organizations.

Since its inception, the ALWR Program has been carried out under the direction of the ALWR Utility Steering Committee, a body of nuclear utility executives who have substantial nuclear experience. The primary points of contact of the ALWR Utility Steering Committee are with the NRC Policy Committee and EPRI. Through the relationship with the NRC Policy Committee, the ALWR Utility Steering Committee establishes program policy, formally requests topic paper and Requirements Document review, and concurs with required actions on generic safety and licensing issues. Through the relationship with EPRI, the ALWR Utility Steering Committee provides program direction and provides approval of documentation submitted to the NRC.

The NRC Policy Committee consists of the Director of the Office of Nuclear Reactor Regulation and the responsible Division Directors. The NRC Policy Committee is responsible for establishing the regulatory policy for the ALWR. The NRC Policy Committee, acting in concert with the ALWR Utility Steering Committee, provides for management review of the regulatory requirement for the ALWR and ensures the necessary NRC staff resources to enable the review of the EPRI topic papers and Requirements Document to be accomplished in an effective and timely manner. The NRC staff performs the detailed review of the topic papers and the Requirements Document.

EPRI is responsible for developing the topic papers and the Requirements Document, as directed by the ALWR Utility Steering Committee. Through technical interface with the NRC staff, EPRI evaluates the applicability of generic safety and licensing issues to the ALWR, recommends resolution to the NRC, and provides technical and schedule information in response to NRC staff requests. EPRI also provides direction to its contractors for the performance of specific work packages related to the ALWR and obtains review of the topic papers and Requirements Document from the Atomic Industrial Forum (AIF), utilities, and nuclear industry, as directed by the ALWR Utility Steering Committee.

2.2 Modes for Accomplishing Tasks in the Program

The ALWR Utility Steering Committee and the NRC Policy Committee have met quarterly. After publication of this NUREG report, the two groups will meet about three times a year, as required. The periodic meetings provide the primary forum for establishing the regulatory policy for the ALWR. Meeting summaries*

*See NRC memoranda from D. H. Moran to C. O. Thomas dated September 16, 1983; November 29, 1983; April 3, 1984; June 14, 1984; January 23, 1985; June 4, 1985; and October 15, 1985. Also see NRC memoranda from D. H. Moran to K. Kniel dated December 27, 1985, and July 5, 1986.

are used to document the agreements, commitments, and action items. These meetings are management level discussions regarding the needs of the ALWR that generally review the progress being made on the ALWR, establish the applicability of generic safety and licensing issues, review the progress being made on the resolution of issues, and cover other administrative and technical concerns. This report is intended to formalize the agreements made in the quarterly meetings.

The Commission was briefed on the ALWR Program by the ALWR Utility Steering Committee, EPRI, and the staff on February 7, 1985. During the briefing, the utility representatives explained how the ALWR Program provided the opportunity for the industry to ensure that the nuclear option is viable and emphasized that the current LWR technology is sound, proven, and should be preserved. The Commissioners agreed that future plants should be simpler, have greater design margin, and be standardized. The Commissioners also endorsed the ALWR Program and urged the staff to continue its close coordination with EPRI and to ensure that the ALWR Program is properly represented in the budget.

The Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Standardization and the Full Committee were briefed on the ALWR Program by the AIF and EPRI on September 5 and 6, 1984. In the ACRS letter (September 11, 1984), in addition to other detailed comments, the following favorable conclusions and recommendations are contained:

We believe that the general approach of attempting to categorize safety and licensing issues into groups applicable or inapplicable to new LWR designs, as EPRI now envisages such designs, should be useful in several ways. It helps to provide a mechanism for identifying those issues warranting attention for resolution and is leading to a useful exchange of information and ideas between the NRC staff and industry. We support continuation of this effort.

EPRI and AIF plan to attempt to develop proposed means for resolution of the issues EPRI has identified as unresolved and applicable to new LWR designs. The EPRI program is intended to lead to accepted means of resolving currently identified issues and minimizing the chance of backfit for future LWRs. If the ambitious EPRI plan for the resolution of such issues for future LWRs is to receive adequately informed review and evaluation by the NRC staff, a very considerable allocation of NRC resources will be required in addition to the current NRC effort on unresolved safety issues and high priority generic issues focused on plants in operation or under construction.

Finally, there have been a number of working meetings between the NRC staff and EPRI. Working meetings are usually scheduled for planning, coordinating, and resolving problems. Variances between staff and industry positions are articulated at working meetings until common ground is found in order to resolve the issue. Early in the program, working meetings were generally directed toward developing information on generic safety and licensing issue status and developing the information necessary to define the elements of resolution for the remaining issues. Working meetings are intended to provide information to the ALWR Program in a timely manner to make possible the more effective utilization of both NRC and EPRI resources.

Figure 2-1 EPRI/Industry ALWR Program organizational interfaces

3 PREPARATION OF THE ALWR REQUIREMENTS DOCUMENT

EPRI and the utility industry, desiring to facilitate the revitalization of the nuclear option, determined that generic safety and licensing issues represented a substantial impact on future plants and lack of an established baseline for regulations impacting future plant design contributed significantly to the uncertainty of the nuclear option. In addition, they perceived that in several aspects of the licensing process current regulatory requirements could be made less stringent ("optimization of current regulatory requirements"). Finally, they recognized that staff review and approval of a performance specification (Requirements Document) was potentially a key step toward certification of a design and construction of a new plant. It was recognized that this represented a new approach which required a high level of NRC involvement during all phases.

The ALWR Program initial phases began under the name "The Advanced LWR Design Evaluation Program." EPRI completed the initial phase before asking the NRC to join the effort. Phase I included a compilation of about 550 generic safety and licensing issues in varying states from unprioritized to resolved. EPRI categorized these issues as to (1) priority (high, medium, low, or drop), (2) resolved or not resolved, and (3) applicable or not applicable to future designs. In addition, EPRI initiated work to identify subjects that could be used to optimize current regulatory requirements. At the same time, EPRI obtained from the utility industry an exhaustive listing of the design configuration and operational attributes desired in future reactors. Phase I involving basic elements was completed internally by EPRI early in 1983 and consisted of:

- determination of preferred characteristics for future LWRs
- a compilation and categorization of generic safety and licensing issues
- a survey of standardization concepts
- development of relationships with vendors, NRC, and other key and interested entities

The closely coordinated NRC and industry/EPRI effort began with Phase II in mid-1983 with the principal thrust toward establishing a regulatory baseline for future plant design. Chief elements of this phase at the startup of the coordinated ALWR Program were the development of an issue resolution methodology for the ALWR Program and the development of plant optimization subjects.

Phase III was initiated in 1985 with the recognition that the Requirements Document would provide the vehicle for NRC approval of the program results. The issue resolution methodology and treatment of plant optimization subjects then became an inherent part of the Requirements Document preparation process. EPRI is responsible for preparing the Requirements Document as directed by the ALWR Utility Steering Committee. The ALWR Utility Steering Committee requested that the NRC review and approve the Requirements Document for use by any qualified group to generate design specifications and a design for a complete nuclear

power plant. The Requirements Document is intended to be the initial documentation necessary to enable the utility industry to exercise the nuclear option to construct new nuclear power plants to go on line in the mid-1990s.

3.1 Treatment of Generic Safety and Licensing Issues

In the initial phases, much of the ALWR Program resources were focused on identifying and treating generic safety and licensing issues. This was necessary because it was recognized that the genesis of most new regulatory requirements stemmed from the resolution of generic safety and licensing issues. Thus an important step toward establishing a regulatory foundation for future plants was the coordinated development of a comprehensive process for treating generic safety and licensing issues which were recognized as having a significant impact on the preparation of the Requirements Document.

The process for treating current and new generic safety and licensing issues for application in the Requirements Document has a number of key elements. The key elements for the treatment of an issue include: (1) identification of the issue; (2) evaluation of the issue; and (3) resolution of the issue. The general process differs slightly in treatment for current and new issues because of their impact on the preparation of the Requirements Document. The general process for treating current generic safety and licensing issues is shown in Figure 3-1. All generic issues identified by July 1, 1986, have been reviewed for their applicability to the ALWR and the issues that are not applicable to future plant design are identified (applicability is documented in Appendix A). The general process for treating new issues identified after July 1, 1986, is shown in Figure 3-2. These key elements in the treatment of current and new issues are described in more detail below.

3.1.1 Issue Identification

The process for treating current and new generic safety and licensing issues (NRR Office Letter No. 40) begins with the identification and definition of the issues. Issues may have their origin in NRC staff concerns, ACRS letters, NRC reports or program plans, and operating experience reports. The current issues with their definition and status are contained in NRC reports* and tracking systems.**

Issues continue to be generated at a rate of approximately 30 issues per year. New issues are identified and defined in NUREG-0933, updated semiannually.

3.1.2 Issue Evaluation

Experience has shown that generic safety and licensing issues may have a number of different objectives. Issues may include the identification of new safety concerns, areas requiring further research to determine if a safety concern exists, actions to be taken by the NRC, and relaxation of current regulatory requirements. The issue prioritization process, documented in NUREG-0933, was primarily developed to prioritize safety issues, but it is used to evaluate the entire spectrum of issues. Because of the ALWR Program needs, it was established that the applicability of generic safety and licensing issues to the ALWR was

*NUREGs-0371, -0410, -0471, -0606, -0660, -0737, -0933, and -0985.

**GIMCS, APTS, GITS.

an important consideration in establishing the regulatory input to the issues to be considered in the development of the Requirements Document. The NRC's issue prioritization can then be used to identify those issues which must be considered in the development of the Requirements Document on the basis of their evaluated priority.

3.1.2.1 Issue Applicability

The process used for determining issue applicability depends on the date the issue is identified. For current issues, the issue applicability was the subject of an agreement between the ALWR Utility Steering Committee and the NRC Policy Committee and was documented in the meeting summary. For issues identified after July 1, 1986, a formal screening process has been developed for application by the NRC to the ALWR.

Before an issue requires action with respect to the Requirements Document, its applicability must be determined. For the purpose of the ALWR Program, issues defined as not applicable are those generic issues which by their definition have no impact on the ALWR plant design, schedule, or construction costs, assuming the decision to build the plant has been made and an acceptable site has been selected. No further action is taken on issues so defined unless a design feature for the ALWR is used which, upon further review, requires the not applicable classification to be changed. Once an issue is determined to be not applicable, it is classified into one of the six categories identified below.

- (1) Superseded - Issues that are redundant to or interrelated with other issues
- (2) Licensing - Issues that address NRC training programs, policy developments, organizational changes, studies, code development and maintenance, and research activities not directly affecting plant design
- (3) Not a Current Standardized Design - Issues that are applicable only to reactor types and design features not included in current standardized designs
- (4) Operating Plants Only - Issues that are applicable only to operating plants
- (5) Not Applicable to Plant Design - Issues that do not affect plant design, such as: plant operation and operating procedures, management of operations, accident management and emergency preparedness, operator training and qualifications, inspection and maintenance, and operating experience reporting
- (6) Regulatory Impact - Issues that address optimization of current regulatory requirements but do not require resolution because the current requirements are acceptable

3.1.2.2 Issue Screening

Because candidate safety issues are submitted to NRC at the rate of approximately 30 per year, it was determined that a set of screening criteria should

be applied to potential issues, starting July 1, 1986, to improve the utilization of resources. Only those issues that represent potential safety issues, as evidenced by passing the screening criteria, will be considered for application to the Requirements Document.

The issue screening performed by the NRC staff is based on input provided by EPRI. The screening criteria to establish issue applicability are listed below. If the answer to any of the following questions is "yes," then the issue shall not be considered further as having the potential for impacting the ALWR design or construction and will not be considered in the review of the Requirements Document.

- (1) Does the issue duplicate an issue previously identified or prioritized?
- (2) Is the issue considered a non-safety issue (e.g., environmental, economic, etc.)?
- (3) Is the issue applicable only to existing plant(s) or plant features not included in the ALWR standard design?
- (4) Is the issue beyond the scope of the ALWR Requirements Document?
- (5) Is the issue research related (i.e., is additional research required to understand or resolve the issue)? It is assumed that appropriate new issues will be identified as applicable once the appropriate research is completed.
- (6) Is the issue considered as a regulatory impact issue?
- (7) Is insufficient information available to evaluate the issue? Review of these issues will be deferred until sufficient information is available.

Any issue that falls into categories 2 or 4 above is considered potentially applicable to an ALWR licensee at a later date (i.e., at time of site-specific or operating license (OL) review) and should be retained for evaluation of its applicability and potential safety significance at that time. The NRC will maintain a list of these issues in a specifically identified document to be used in the review of any application referencing the Requirements Document.

3.1.2.3 Issue Prioritization

The NRC's issue prioritization process for all generic issues, including operating plants, plants under construction, and future plants, is documented in NUREG-0933. The prioritization process was established to develop a priority list of generic safety issues based on the potential safety significance and cost of implementing each issue. The primary purpose of the prioritization process is to assist in the timely and efficient allocation of resources to those safety issues that have a high potential for reducing risk and to remove from further consideration issues that have little safety significance and little potential for worthwhile safety enhancement.

Once the issue applicability has been established, the NRC's prioritization process is used in the ALWR Program to establish the priority of the issue to determine any further action necessary. On the basis of the results of the

issue prioritization process, each applicable issue is placed in one of the following three categories: (1) resolved issues, (2) drop issues, and (3) remaining issues.

- Resolved Issues

Resolved issues are those applicable issues which are sufficiently resolved or complete enough to enable their impact on the cost and schedule of the ALWR design to be evaluated. Resolved issues include the following NRC categories:

- (1) resolution available and documented in an NRC document (NUREG, NRC memorandum, safety evaluation report, or equivalent)
- (2) technical resolution complete as documented in NUREG-0606
- (3) a requirement as identified in NUREG-0737
- (4) a resolution that resulted in a new regulatory requirement (by rule, Standard Review Plan change, or equivalent)
- (5) resolution resulted in no new requirements

Once an issue has had its resolution identified, the resolution is treated as described in Section 3.1.3.

- Drop Issues

Drop issues are those applicable issues which have little or no prospect of improvements that are substantial and worthwhile. These issues are identified as having a low or drop priority in NUREG-0933 and are not to be pursued by the NRC. Therefore, drop issues are not considered in the development of the Requirements Document.

- Remaining Issues

Remaining issues are those applicable issues having a priority other than low or drop that have entered the NRC prioritization process and have not been resolved. Issues that have not been prioritized are treated as remaining issues until they have been prioritized.

Because of the detailed evaluations, reviews, and concurrences that have to be made to establish the priority of issues, a long time period is frequently required before priority is formally documented. In addition, the period of time between the identification of an acceptable resolution to an issue and formal completion of an issue can be several months. These delays could result in the expenditure of unnecessary resources by the ALWR Program, both staff time and utility/EPRI effort.

To address this situation, a forecasting process has been developed for use in the ALWR Program. The process consists of holding periodic working meetings in which the responsible NRC staff members provide current information on issue status and priority. This information is treated in the same manner as is formal NRC-documented information, except that progress on the issue is closely

tracked to ensure that the forecast does not unexpectedly change. It is recognized that it is somewhat risky to use this preliminary information because of the potential for change.

The forecasting process is very effective in ensuring the current issue status and priority are available to the ALWR Program. The availability of this information provides a vehicle for effective application of resources to the ALWR Program. Experience with the forecasting process has demonstrated that the benefits outweigh any potential risk.

3.1.3 Issue Resolution

The final step in the process for treating generic safety and licensing issues is the identification of an acceptable resolution of the issues for the ALWR. The resolution process is highly dependent on the timing of issue identification. This situation exists because of the sensitivity of the Requirements Document development to the incorporation of the resolution of issues. To accommodate this consideration, the issue resolution process has been separated into two processes depending on the timing of issue identification: (1) current issues - issues identified before July 1, 1986, and (2) new issues - issues identified after July 1, 1986.

3.1.3.1 Current Issue Resolution

Two paths can lead to resolution of current issues for the ALWR. These paths are shown in Figure 3-1. The first path consists of using the generic issue resolution identified by the NRC, and the second path consists of developing an issue resolution applicable to the ALWR.

For a large number of generic safety and licensing issues, a generic issue resolution has been identified, and the resolution is to be incorporated into the Requirements Document. EPRI accomplishes this by maintaining a current list of resolved issues that have been corroborated by the NRC. It is first determined if the resolution of the issue resulted in a new regulatory requirement or additional guidance. Next, each resolved issue is assigned to a Requirements Document chapter. Finally, appropriate references describing the issue resolution are identified. This information is used to develop appropriate plant requirements to reflect the resolution of the issues.

Even with the issue prioritization process and the large number of issues that are resolved and issues that are not applicable, the number of remaining issues requiring resolution for the ALWR is sufficient to challenge the concept of regulatory stabilization. Further, the relatively rapid rate of introducing new issues combined with the relatively slow and thorough process for issue prioritization and resolution makes it difficult to reduce the backlog of remaining issues.

In order to proceed with the development of the Requirements Document in an efficient and timely manner, it is necessary to identify an acceptable issue resolution for the ALWR for the current remaining issues. Thus, a unique ALWR issue resolution process has been developed. This process consists of using topic papers to identify the elements of resolution for an issue or group of linked, similar issues.

Topic papers are written to identify the elements of resolution for each of the remaining generic safety and licensing issues. The identification of the remaining issues to be the subject of the topic papers is obtained from a list of remaining issues which have been grouped into similar topics by Requirements Document chapter. This is referred to as the "short list" and was developed to ensure that all facets of an issue or group of issues are addressed to obtain complete closure. Topic papers are intended to provide a short summary of the linked issues, the current regulatory status, and the proposed elements of resolution along with the rationale for their selection. The topic papers are reviewed by the NRC staff and an acceptable set of elements of resolution is agreed to by the ALWR Utility Steering Committee and the NRC Policy Committee.

The acceptable sets of elements of resolution for the remaining generic safety and licensing issues are used as input to the development of the Requirements Document. In the Requirements Document, the elements of resolution are translated into specific plant requirements. The acceptability of the plant requirements based on the elements of resolution identified in the topic papers are established during the NRC staff's review of the Requirements Document and are documented in the Safety Evaluation Report.

3.1.3.2 Resolution of New Issues

It is recognized that new issues will continue to be generated during and after the review process for the ALWR. However, to maintain accurately the baseline of generic issues impacting future plants, additional control over these "new issues" is necessary. To achieve this control, a unique process applicable only to the ALWR Program has been developed. This process is shown in Figure 3-2.

An important characteristic of the program is that the ALWR must be relatively immune to backfits. EPRI and the LWR Steering Committee have taken an important step to reduce the potential for backfits by requiring a design that is more conservative and is more resistant to transients and accidents than current plants are. Additional assurance is needed that regulatory changes resulting from resolution of generic issues will not impact the ALWR unless the need is demonstrated. To address this concern, a set of implementation criteria has been identified for, and will be applied to, the generic resolution of any issue identified after July 1, 1986. The implementation criteria are intended to identify those safety issues that represent a substantial improvement in plant safety and have a favorable relationship between risk improvement and cost benefit.

Any new issue which passes the initial screening criteria will be judged to have the potential for affecting the ALWR design or construction requirements. Accordingly, to stabilize the ALWR requirements as much as possible, only those new issues which are directed toward correcting a major safety problem, so as to maintain an acceptable level of safety, will be considered applicable to and implemented in the ALWR Requirements Document. The initial screening of new issues will be performed by the NRC staff on a periodic basis consistent with the level of issue generation. If necessary, EPRI will provide input to aid in the screening of new issues.

To determine if an issue has major safety significance and is applicable to the ALWR Requirements Document, it will be evaluated as described below. After an issue has passed the initial screening criteria and has been resolved by the

NRC, it will be evaluated by the staff for application to the ALWR Requirements Document using the following implementation criteria. Insofar as its Requirements Document is concerned, EPRI will provide information necessary for the evaluation.

If one or more of the following criteria apply to the issue, it will be considered to apply to the ALWR and a specific set of plant requirements will be added to the Requirements Document to address the issue.

- (1) Would the core melt frequency goal established in the ALWR Requirements Document be exceeded as a result of this issue?
- (2) Would the offsite accident radiological consequences dose requirements established in the Requirements Document be exceeded as a result of this issue?
- (3) Would the Commission's safety goals be exceeded as a result of this issue?

The primary evaluation basis for issue resolution applicability will be the value/impact assessment that is provided for the resolution of generic issues by the NRC. In those cases in which the generic value/impact evaluation does not clearly establish that the issue need not be considered in the ALWR design, EPRI will give further information to provide the basis for not considering the issue in the design or to place appropriate new requirements in the Requirements Document consistent with the issue resolution.

3.2 Treatment of Plant Optimization Subjects

Another part of the ALWR Program is to identify and pursue the updating of those subjects that would result in optimization of plant design, construction, and operation. A number of technically supportable alternatives to current regulatory requirements have been identified for application to the ALWR Program. These alternatives are identified as "plant optimization subjects."

The process for treating plant optimization subjects for application to the Requirements Document has two key elements: (1) subject identification and (2) subject assessment. The general process for treating plant optimization subjects is shown in Figure 3-3, and the key elements are described in more detail below.

3.2.1 Identifying Optimization Subjects

The process for treating plant optimization subjects begins with the identification and definition of the subjects. Proposed plant optimization subjects are initially identified by the ALWR Program contractors and utilities. The proposed subjects are clearly defined by the ALWR Program Office and the ALWR position on the subject is identified so that the subject assessment can proceed.

A very conservative approach has been taken in identifying plant optimization subjects to ensure that only a very limited number of important subjects will be pursued by the ALWR Program. There are three primary considerations with regard to the acceptability of plant optimization subjects for application to the ALWR through the Requirements Document: (1) the subject must have a high payoff; (2) there must be a high likelihood of NRC acceptance based on

information available independent of the ALWR Program; and (3) there must be essentially no impact on ALWR Program resources. By adopting this conservative approach, only a very limited number of plant optimization subjects will be pursued by the ALWR Program; these subjects, however, will have a very high likelihood of acceptance and high payoff for the ALWR Program.

3.2.2 Assessing Optimization Subjects

Once adequately defined, the plant optimization subject is screened and categorized. The screening is performed using a set of screening criteria established by the ALWR Program and approved by the ALWR Utility Steering Committee. When the screening is completed, recommendations are developed for considering or not considering the subject in the ALWR Program. Subject papers are then developed for each subject recommended for inclusion in the program.

The criteria used in the assessment of plant optimization subjects are as follows:

- (1) The subject must be based on established technology. Thus, the data must be available supporting acceptance of the ALWR position as applied to the ALWR Program.
- (2) The subject must not degrade the safety of the plant. There should be no increase in core damage frequency.
- (3) There must be a significant improvement in plant performance or economics. The subject should represent a substantial reduction in life-cycle cost, a substantial simplification in design, or a substantial improvement in operability.
- (4) There must be a substantial improvement in the institutional environment. There must be a perception of reduced public risk or a better balance of safety requirements.

These screening criteria are then used to separate plant optimization subjects into four categories by potential safety or economic benefits and resource requirements. The categories are identified and described below. Only subjects falling into Categories A, B, and C are candidates for pursuit by the ALWR Program.

Category A - The subjects have attributes which provide substantial benefits, likely near-term NRC acceptance, and require no additional program resources.

Category B - The subjects have attributes which provide substantial benefits, likely long-term NRC acceptance, and in fact are being pursued independent of the program.

Category C - The subjects have attributes which provide nominal benefits, likely near-term NRC acceptance, and require no additional program resources.

Category D - The subjects were considered and evaluated but do not satisfy the screening criteria and are not being pursued.

Plant optimization subject papers are prepared for all subjects recommended for inclusion in the ALWR Program. The subject papers are reviewed by the ALWR Utility Steering Committee and are provided to the NRC staff for comment. Upon receipt of NRC comments, EPRI incorporates the subject into the Requirements Document, as appropriate.

3.3 Preparing the Requirements Document

The process used in the preparation of the Requirements Document is shown in Figure 3-4. The key elements include: (1) establishing the current regulatory requirements; (2) incorporating generic safety and licensing issue and plant optimization subject inputs; and (3) preparing the Requirements Document. These key elements are described in more detail below.

3.3.1 Establishing Current Regulatory Requirements

The current regulatory requirements and guidance used as input to the Requirements Document are contained in the regulations (Code of Federal Regulations, Title 10, "Energy,"), regulatory guides, the Standard Review Plan (NUREG-0800), and in the resolution of generic safety issues as described in Section 3.1.

Regulations

The regulations contain all of the requirements the NRC has placed on nuclear power plant design and the ALWR design must satisfy the regulations. The primary regulation associated with the level of technical information required in support of a proposed design is 10 CFR 50.34. The requirements of 10 CFR 50.34 include provisions that a report be provided which describes the design, identifies the design bases and any limits on plant operation, and provides a safety analysis of the design. Other key requirements relating to the technical information required for the ALWR design identified in 10 CFR 50.34 include: design criteria; emergency core cooling system performance; Technical Specifications and their bases; codes and standards; and radioactive effluents and radiation exposures. The additional requirements identified as a result of the accident at Three Mile Island, which are applicable to ALWR design, are identified in 10 CFR 50.34(f). Also, 10 CFR 50.34(g) requires that the design be evaluated against the SRP and, where differences exist, an evaluation must be provided which describes how the alternative proposed provides an acceptable method of complying with the NRC's rules and regulations.

Regulatory Guides

Regulatory guides provide acceptable methods of satisfying key regulatory requirements. Regulatory guides are not substitutes for regulations, and compliance with them is not specifically required. Methods and solutions different from those set out in regulatory guides are acceptable if an acceptable basis is provided. For the ALWR, the method of compliance with the applicable regulatory guides will be identified. Where alternative approaches are taken, the proposed alternatives will be suitably identified.

Standard Review Plan

The Standard Review Plan is a guidance document for NRC staff reviewers in performing safety reviews. The principal purpose of the Standard Review Plan is

to ensure quality and uniformity of NRC staff reviews. The SRP provides review procedures, acceptance criteria, and evaluation findings for specific areas of review. As required by 10 CFR 50.34(g), the ALWR will be evaluated against the SRP. EPRI will follow the SRP, as applicable, in writing the Requirements Document and the staff will follow the plan, as applicable, in the review of the Requirements Document. It must be made clear, the ALWR Requirements Document is a performance specification or design envelope which will be followed by designers who write design specifications and generate an ALWR standard design. The SRP is a plan for the review of a safety analysis report and not for the review of a performance specification. Therefore, the staff will use the SRP in the review of the Requirements Document only as a reference to NRC regulations and guidance to ensure that the ALWR Requirements Document thoroughly addresses all the considerations covered therein.

3.3.2 Requirements Document Description

The Requirements Document will be the key output document of the EPRI/Industry ALWR Program. The Requirements Document will establish the detailed requirements that will be imposed on the ALWR design. The primary goals of the Requirements Document as enumerated by EPRI are to establish requirements that define a plant that is: (1) greatly simplified compared to current plants with respect to constructability, (2) based on fully proven technology, (3) less subject to accident, (4) more stable during operating transients, (5) lower in lifetime costs, and (6) licensable.

The Requirements Document is scheduled to consist of 13 chapters placing requirements on the major ALWR features. These chapters are identified in Table 3-1. The Requirements Document is being prepared by EPRI contractors and is being reviewed by the ALWR Utility Steering Committee and the nuclear industry and utilities. The staff review plan for the Requirements Document is described in Section 4.

EPRI is developing the Requirements Document utilizing applicable, current regulatory requirements, to structure specific plant requirements. Further, new regulatory requirements and additional guidance resulting from the resolution of currently identified and future generic safety and licensing issues are incorporated into the Requirements Document as specific plant requirements as described in Section 3.1. Optimization subjects identify changes to regulatory requirements adopted in the Requirements Document.

It is recognized that a number of technically supportable alternatives to current regulatory requirements are available for future plants. As a part of the process for preparing the Requirements Document, these alternatives are evaluated to determine which represent substantial potential improvements in the ALWR economics. The issues identified through this process are called plant optimization subjects. The number of subjects has intentionally been limited to provide a better opportunity for stabilizing regulatory requirements. The process for treating plant optimization subjects is described in Section 3.2.

Figure 3-1 ALWR process for treating current issues

Figure 3.2 ALWR process for treating new issues

Figure 3-3 ALWR process for treating optimization subjects

Figure 3-4 Requirements Document preparation and review process

Table 3-1 Requirements Document chapters

Chapter	Title
1	Overall Requirements
2	Power Generation Systems
3	Reactor Coolant and Non-Safety Reactor Auxiliary Systems
4	Reactor Systems
5	Safety Systems
6	Building Design and Arrangements
7	Fueling and Refueling
8	Plant Cooling Water Systems
9	Site Support Systems
10	Instrumentation and Control Systems and Plant Control Stations
11	Electric Power Systems
12	Radioactive Waste Processing Systems
13	Turbine Generator Systems

4 NRC REVIEW PROCESS FOR THE ALWR

The Requirements Document will be reviewed by the staff as described below and shown in Figure 1-1. The primary documentation of the review will be contained in a Safety Evaluation Report (SER). The staff's SER will be used to document the results of the staff review of the Requirements Document. The NRC staff will review each of the 13 chapters and will also review the Requirements Document in its entirety. The Requirements Document is to be reviewed using the Standard Review Plan as guidance, as modified by the resolution of applicable, generic safety and licensing issues and the acceptable elements of resolution of the remaining issues and plant optimization issues. Specific sections of the Standard Review Plan will be modified only if necessary to accommodate the ALWR design requirements.

Because of the unique approach of the ALWR Program, the NRC staff review is limited to approval of the specific plant requirements as laid out in the Requirements Document. Once the Requirements Document has been approved by the NRC, it will be used by the staff in reviewing future LWRs which commit to comply with these requirements as guidance regarding the design criteria, applicable SRP sections, and regulatory guides, etc., which apply to the design.

4.1 Safety Evaluation Report

The primary NRC documentation of the ALWR Program review will be contained in an SER. The SER will specifically cover each of the 13 chapters and generally evaluate the entire Requirements Document. Drafts of the SER covering each chapter will be prepared upon completion of the specific chapter review. The SER on the entire document will be prepared after the NRC has completed its review of all chapters. The completed SER will serve to document NRC's position on the safety and licensing requirements for the ALWR.

4.2 ALWR Requirements Document Staff Review Plan

4.2.1 The Concept of the Advanced Light Water Reactor

EPRI and the nuclear utilities that sponsor EPRI asked the NRC to enter into a closely coordinated effort to facilitate the U.S. nuclear industry's efforts toward providing the U.S. electric utilities with a viable nuclear option when ordering new plant construction during the 1990 time frame. The concept is predicated on a compilation of performance specifications, which, when complied with, will result in a plant design embodying consideration for all NRC rules, policies, regulations, and generic safety issues and which reduces as much as possible, the potential for backfit.

EPRI is compiling a Requirements Document in 13 chapters, which is a performance specification or design envelope. The Requirements Document will apply to a range of ALWRs, both BWR and PWR, with output from 400 MWe to 1350 MWe.

The term "advanced" is used in the context of the most advanced within the existing LWR technology. Only proven technology, based on demonstrated experience to ensure high reliability, will be considered.

4.2.2 Specific Preparation for the Review of Chapter 1 of the Requirements Document

The Standard Review Plan (NUREG-0800) will be used as guidance as applicable. However, the LWR edition of the SRP is written for the review of safety analysis reports for nuclear power plants. Reviewing a performance specification while following the SRP will require interpretation and the development of supplemental guidance because of the differences between a safety analysis and a performance specification. Reviewers must keep in mind that the Requirements Document will contain requirements for an entire plant including: nuclear steam supply system (NSSS), balance of plant (BOP), site arrangements, auxiliary structures and connections within the site, and offsite service connections.

4.2.3 The Review of Chapter 1 of the Requirements Document

The ALWR Requirements Document delineates the essential performance requirements EPRI has determined will satisfy the nuclear utility industry, while meeting applicable NRC rules, regulations, and generic safety issue resolutions. The text format of the Requirements Document is side-by-side "Requirements" and corresponding "Engineering Rationale" for each chapter subheading. The Chapter 1 content has three levels of detail. It is assumed this format is representative of the entire document.

The staff reviewer is interested in the applicant's treatment of specific nuclear power plant attributes and capabilities which are the subject of NRC rules, regulations, and resolutions to generic safety issues. The staff reviewers will use the SRP to guide the review and will provide reasonable assurance that all applicable regulatory requirements have been addressed.

For each chapter subheading, the "Requirements" section will be reviewed with the understanding that a performance specification is being presented such that, if one were to use it to generate a design specification and then generate a design, the design would conform to or meet the pertinent NRC regulations. If questions arise during this process, the corresponding engineering rationale will be used as guidance.

Three tests will be applied to each subsection containing a discrete "Requirement" and corresponding "Engineering Rationale."

- (1) Is the specification complete?
- (2) Is it clearly stated?
- (3) If the requirement is to be implemented in a design, using the Engineering Rationale to guide the designer, will the NRC criteria (regulations) be met? An alternate to this test would be: "Can the NRC criteria (regulations) be met?"

As end-product for this review, the staff will produce a single SER which includes a chapter or section for each corresponding chapter in the Requirements Document.

Assuming the Requirements Document meets staff approval, the staff approval will include the following:

(1) The Overall Approval

The staff has reviewed the Requirements Document and finds that it contains the necessary requirements that, if properly translated into a design in accordance with current practice and guidance documents, will generate a nuclear power plant design which will have all the attributes required by NRC regulations to assure there is no undue risk to the public health and safety as required by the regulations.

(2) Chapter or Section Approval

Chapter/Section [supply appropriate number] of the Requirements Document contains requirements that, if properly translated into a design in accordance with current practice and guidance documents, will generate a nuclear power plant design which will have all the attributes required by NRC regulations to assure there is no undue risk to the public health and safety as required by the regulations.

5 HISTORICAL SUMMARY OF UTILITY INDUSTRY INITIATIVES LEADING TO ALWR PROGRAM

5.1 Introduction

EPRI and the electric utilities which support EPRI concluded in 1981 it is in the national interest that the nuclear power option be revitalized to support future electric power generation. The utility industry perceived that the need arises because even with low electric growth rates, active conservation efforts, and peak load leveling, many utilities predict an electric power shortage to occur in the mid-1990s unless new power plants can be built quickly and economically. Industry determined that coal and nuclear power plants are the two major options available to satisfy the base plant power needs of the 1990s.

To address the need for a viable nuclear power generation option, the Electric Power Research Institute (EPRI), as directed by its ALWR Utility Steering Committee, embarked on the EPRI/Industry Advanced Light Water Reactor (ALWR) Program.

The electric utilities structured the ALWR program to give them an opportunity to establish the configuration of light water reactor (LWR) power plants of the future. This approach provides better assurance that utility requirements will be considered in the design process. The electric utilities also concluded that early review by the NRC staff will ensure that regulatory requirements are appropriately considered in the design process, lending greater stability to the regulatory process.

5.2 Industry Assessment of the Future for Nuclear Power

The electric utilities determined there is consensus that the nuclear power industry is facing a period of extreme uncertainty with respect to the prospects for new nuclear power plants. All nuclear power plants operating or under construction in 1986 were ordered before 1975. By 1990, almost all the nuclear power plant projects presently under construction will have been built or canceled, and there are currently no prospects for new domestic orders.

A February 1984 paper prepared by the Office of Technology Assessment (1984) reported

Without significant changes in technology, management and level of public acceptance, nuclear power in the United States is unlikely to be expanded in this century beyond the reactors already under construction. Currently, nuclear powerplants present too many financial risks as a result of uncertainties in electric demand growth, very high capital costs, poor construction management, operating problems, increasing regulatory requirements, and growing public opposition.
(p.xi)

This finding is consistent with a June 1984 Atomic Industrial Forum (AIF) Study Group conclusion:

Nuclear power cannot at this time be considered a viable option on which to base new electric generating capacity in the U.S. If nuclear power is to compete for new electric generation, ways must be found to reduce the uncertainties and financial risks associated with capital-intensive and long lead-time construction. (p.iii)

This is true despite the fact that nuclear power accounted for 380 billion kWh or approximately 15.5% of the national supply in 1985, despite the fact that nuclear power is expected to account for the addition of more capacity over the next three years than any other type of power generation, and despite the fact that nuclear power is being aggressively pursued on other parts of the planet.

The AIF report further states:

Responsibility for revitalizing the nuclear power option rests principally with the private sector. However, until such time as more certainty and stability have been introduced into the licensing and regulation of nuclear power, until lead times have been reduced, and until there is a greater public consensus in favor of nuclear power, the private sector cannot take on the open-ended financial risks that now attend the nuclear power option.

5.3 Background and Initial Formulation of the ALWR Program

The initial objective of the EPRI/Industry ALWR Program involved determining the desirable characteristics of the next generation of LWRs. To satisfy this objective, EPRI held a number of meetings with nuclear utility executives and plant managers to determine their concerns about current LWRs and to obtain their ideas about improvements needed in future plants. It was evident from the initial discussions that the major concerns involved fundamental and interrelated factors of a political, economic, and regulatory nature.

The EPRI report (RP-1585, pp. 9 and 10) prepared as a result of the utility meetings contained the following conclusions and recommendations:

- (1) Utility comments received during the trips involved interrelated political, economic, regulatory, and technical matters that addressed prerequisites for new LWR orders as well as the characteristic features of the designs. The prerequisites for new LWR orders involve stabilization of the licensing requirements and a more favorable climate for financing new construction. The design characteristics desired in new LWRs were preservation of current technology and experience with emphasis on simplification of present backfitted designs and with more consideration given to conservatism, reliability, operability, and maintainability as they affect the safety and availability of the plant.
- (2) Stabilization of licensing requirements is a paramount and prerequisite step toward the goal of new LWR plant designs. Achieving this requires a concerted industry and NRC effort at coming to terms with open licensing issues including rollback of some, and then finalizing the details of licensing requirements

within the context of a detailed plant design. Durable stabilization requires standardization of plant designs along with some form of organizational continuity in the utilities and the NRC to manage and control changes to the standardized plant designs.

- (3) The nature and viability of the effort described in item 2 above requires active participation of the utilities and the NRC early [in] and throughout the process. The participation should be at a policy and an engineering level to ensure that utility interests are adequately and appropriately considered and that there is timely and effective input of the utility lessons learned into standardized designs of new LWRs.
- (4) Stabilizing licensing requirements along with standardization of plant designs is a major undertaking that should be done as a coupled effort since they are interrelated. Consideration should be given to use of present plant designs to serve as the design base for new LWR designs unless final licensing requirements or utility mandated betterments require more than minor changes to overall plant design. The outcome would tend to invalidate continuing this approach.

On the basis of the conclusions and recommendations listed above and on initial ALWR Program efforts, EPRI established that there were two key elements that should be pursued in dealing with the NRC. These were (1) establish a list of stable regulatory requirements that must be satisfied by any new LWR design and (2) generate a set of utility-reviewed and NRC-approved plant requirements for the ALWR. These elements involve a high level of interaction between the NRC and EPRI to ensure the establishment of a regulatory baseline for future plant design and have been the major focus of the ALWR Program for the past three years.

The ALWR Program has been developed recognizing that for new nuclear plants to be operational in the mid-1990s, plant technology must be based on proven LWR technology, considering the important technical lessons learned from the extensive industry experience in constructing, licensing, operating, and maintaining the present generation of nuclear plants. The new designs are to be safer, easier to construct, of simpler configuration, easier to operate and maintain, and more reliable than current nuclear power plants. In addition, the ALWR Program is placing special emphasis on reducing plant investor risks.

6 REFERENCES

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APPENDIX A

ISSUES THAT ARE NOT APPLICABLE TO THE ALWR PROGRAM

The not applicable issues identified with respect to the ALWR Program are identified in Table A-1 and Tables A-1a through A-1f. These tables identify the issue number, the not applicable category, and the issue title. These categories have been reviewed by the NRC staff and have been corroborated by the NRC Policy Committee as documented in the meeting summary of the quarterly meeting with ALWR Utility Steering Committee.

Issue Applicability

Before a currently identified issue requires further action with respect to the ALWR Program, it must be determined if the issue is applicable. For the purposes of the ALWR Program, not applicable issues are defined as those generic issues which by their definition have no impact on the ALWR plant design, schedule, or construction costs, assuming the decision to build the plant has been made and an acceptable site has been selected. No further action is taken on such issues unless a design feature for the ALWR is used which, upon further review, requires the not applicable classification to be changed.

Six categories of not applicable issues have been identified for use in the categorization of issues in the ALWR Program. These categories are described below and the issues assigned to each category are identified in Tables A-1a through A-1f.

- (1) Superseded - Issues that are redundant to or interrelated with other issues (Table A-1a).
- (2) Licensing - Issues that address NRC training programs, policy developments, organizational changes, studies, code development and maintenance, and research activities not directly affecting plant design (Table A-1b).
- (3) Not a Current Standardized Design - Issues that are applicable only to reactor types and design features not included in current standardized designs (Table A-1c).
- (4) Operating Plants Only - Issues that are applicable only to operating plants (Table A-1d).
- (5) Not Applicable to Plant Design - Issues that do not affect plant design, such as: plant operation and operating procedures, management of operations, accident management and emergency preparedness, operator training and qualifications, inspection and maintenance, and operating experience reporting (Table A-1e).

- (6) Regulatory Impact - Issues that address optimization of current regulatory requirements but do not require resolution because the current requirements are acceptable (Table A-1f).

Table A-1 Not applicable issues

Code	Category	No. corroborated	No. forecast	Total issues
1	Superseded issues	109	4	113
2	Licensing issues	87	10	97
3	Not current standardized design issues	38	1	39
4	Operating plants only issues	68	31	18
5	Not applicable to plant design issues	68	31	99
6	Regulatory impact issues	9	11	20
	Total	<u>319</u>	<u>67</u>	<u>386</u>

Table A-1a Code 1 - Superseded issues

Issue No.	Title
A-32	Missile Effects
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents
B-4	ECCS Reliability
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA
B-16	Protection Against Piping Failures in Fluid Systems Outside Containment
B-17	Criteria for Safety-Related Operator Action (F)
B-18	Vortex Suppression Requirements for Containment Sumps
B-24	Seismic Qualification of Electrical and Mechanical Components
B-34	Occupational Radiation Exposure Reduction
B-45	Need for Power--Energy Conservation
B-51	Assessment of Inelastic Techniques for Equipment and Components
B-52	Fuel Assembly Seismic and LOCA Response
B-57	Station Blackout
B-67	Effluent and Process Monitoring Instrumentation
B-69	ECCS Leakage Ex-Containment
B-71	Incident Response
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel
C-3	Insulation Usage Within Containment
C-13	Non-Random Failures
I.A.2.6(1)	Long-Term Upgrading of Training and Qualifications
I.A.2.6(3)	Long-Term Upgrading of Training and Qualifications
I.A.3.3	Establish Requirements for Operator Fitness

(F) = forecast issue

Table A-1a Code 1 - Superseded issues (continued)

Issue No.	Title
I.A.4.2(1)	Training Simulator Improvements--Long Term
I.A.4.2(4)	Training Simulator Improvements--Long Term
I.B.1.1(1)	Management for Operations--Long-Term Improvements
I.B.1.1(2)	Management for Operations--Long-Term Improvements
I.B.1.1(3)	Management for Operations--Long-Term Improvements
I.B.1.1(4)	Management for Operations--Long-Term Improvements
I.B.1.1(6)	Management for Operations--Long-Term Improvements
I.B.1.1(7)	Management for Operations--Long-Term Improvements
I.C.9	Long-Term Plan for Upgrading Procedures
I.D.3	Control Room Design--Safety System Monitoring
I.D.4	Control Room Design Standard
I.D.5(5)	Control Room Design--Improved Instrumentation Research
II.A.2	Site Evaluation of Existing Facilities
II.C.3	Systems Interaction
II.E.2.1	Reliance on Emergency Core Cooling System
II.E.3.2	Decay Heat Removal--Systems Reliability
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements
II.E.3.5	Decay Heat Removal Regulatory Guide
II.H.3	Evaluate and Feedback Information Obtained From TMI
II.J.3.1	Organization and Staffing to Oversee Design and Construction
II.J.3.2	Management for Design and Construction--Issue Regulatory Guide
II.K.2(8)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(12)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(18)	Commission Orders on B&W Plants to Mitigate Accidents

Table A-1a Code 1 - Superseded issues (continued)

Issue No.	Title
II.K.3(4)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(6)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(8)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(23)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(26)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(32)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(33)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(34)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(35)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(36)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(37)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(38)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(39)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(40)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(41)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(42)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(43)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(47)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(48)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(49)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(50)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(51)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(52)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(53)	Final Recommendations of B&O Task Force to Mitigate Accidents

Table A-1a Code 1 - Superseded issues (continued)

Issue No.	Title
II.K.3(54)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(55)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(56)	Final Recommendations of B&O Task Force to Mitigate Accidents
III.D.2.2(2)	Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis
III.D.2.2(3)	Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis
III.D.2.2(4)	Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis
5	Design Check and Audit of Balance of Plant Equipment
8	Inadvertent Actuation of Safety Injection in PWRs
9	Reevaluation of Reactor Coolant Pump Trip Criteria
11	Turbine Disc Cracking
16	BWR Main Steam Isolation Valve Leakage Control System
18	Steamline Break With Consequential Small-Break LOCA
19	Safety Implications of Non-Safety Instrument and Control Power Supply Bus
26	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power
27	Manual Versus Automated Actions
28	Pressurized Thermal Shock
31	Automatic Emergency Core Cooling System Switch to Recirculation
32	Flow Blockage in Essential Equipment Caused by Corbicula
33	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control Power System
37	Steam Generator Overfill and Combined Primary and Secondary Blowdown
39	Potential for Unacceptable Interaction Between the CRD System and Nonessential Control Air System

Table A-1a Code 1 - Superseded issues (continued)

Issue No.	Title
42	Combination Primary/Secondary System LOCA
44	Failure of Saltwater Cooling System
46	Loss of 125-Volt DC Bus
52	SSW Flow Blockage by Blue Mussels
54	Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980
56	Abnormal Transient Operating Guidelines as Applied to Steam Generator Overfill Events
60	Lamellar Tearing of Reactor System Structural Supports
62	Reactor Systems Bolting Applications (F)
65	Component Cooling Water System Failures
67.3.1	Steam Generator Staff Actions--Steam Generator Overfill
67.3.2	Steam Generator Staff Actions--Pressurized Thermal Shock
67.3.3	Steam Generator Staff Actions--Improved Accident Monitoring
67.3.4	Steam Generator Staff Actions--Reactor Vessel Inventory Measurement
67.4.1	Steam Generator Staff Actions--Reactor Coolant Pump Trip
67.4.2	Steam Generator Staff Actions--Control Room Design Review
67.4.3	Steam Generator Staff Actions--Emergency Operating Procedures
67.6.0	Steam Generator Staff Actions--Organizational Response
67.9.0	Steam Generator Staff Actions--Reactor Coolant System Pressure Control
97	PWR Reactor Cavity Uncontrolled Exposure
102	Human Error in Events Involving Wrong Unit or Wrong Train
109	Reactor Vessel Closure Failure (F)
113	Hydraulic Snubbers (F)

(F) = forecast issue

Table A-1b Code 2 - Licensing issues

Issue No.	Title
A-19	Digital Computer Protection System (F)
A-20	Impacts of the Coal Fuel Cycle
A-27	Reload Application
B-3	Event Categorization
B-7	Secondary Accident Consequence Modeling
B-11	Subcompartment Standard Problems
B-13	Marviken Test Data Evaluations
B-15	CONTEMPT Computer Code Maintenance
B-20	Standard Problem Analysis
B-21	Core Physics
B-23	LMFBR Fuel
B-25	Piping Benchmark Problems
B-27	Implementation and Use of Subsection NF
B-30	Design-Basis Floods and Probability
B-33	Dose Assessment Methodology
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Nuclear Power Plants
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments
B-62	Reexamination of Technical Basis for Establishing SLs and LSSSs
B-72	Development of Models for Assessing Health Effects and Life Shortening From Uranium and Coal Fuel Cycles
C-15	NUREG Report for Liquids Tank Failure Analysis
I.A.2.4	NRR Participation in Inspector Training
I.A.3.5	Licensing of Personnel--Statement of Understanding With INPO and DOE

(F) = forecast issue

Table A-1b Code 2 - Licensing issues (continued)

Issue No.	Title
I.A.4.3	Feasibility Study for Procurement of Training Simulator
I.A.4.4	Feasibility Study to Evaluate Potential Value of NRC Engineering Computer
I.B.1.3(1)	Loss of Safety Function
I.B.1.3(2)	Loss of Safety Function
I.B.1.3(3)	Loss of Safety Function
I.B.2.1(1)	Revision of IE Inspection Program
I.B.2.1(2)	Revision of IE Inspection Program
I.B.2.1(3)	Revision of IE Inspection Program
I.B.2.1(4)	Revision of IE Inspection Program
I.B.2.1(5)	Revision of IE Inspection Program
I.B.2.1(6)	Revision of IE Inspection Program
I.B.2.1(7)	Revision of IE Inspection Program
I.B.2.2	Resident Inspectors at Operating Reactors
I.B.2.3	Inspections at Operating Reactors--Regional Evaluations
I.B.2.4	Overview of Licensee Performance
I.D.5(3)	Control Room Design--Improved Instrumentation Research (F)
I.D.6	Control Room Design Technology Transfer Conference
I.E.1	Establish Office for Analysis and Evaluation of Operational Data
I.E.2	Program Office--Operational Data Evaluation
I.E.3	Operational Safety Data Analysis
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs
I.E.5	Nuclear Plant Reliability Data System
I.E.6	Reporting Requirements for Operating Experience

(F) = forecast issue

Table A-1b Code 2 - Licensing issues (continued)

Issue No.	Title
I.E.7	Information for Analysis and Dissemination of Operating Experience--Foreign Sources
I.E.8	Human Error Rate Analysis
II.B.5(1)	Safety Review Consideration--Research on Phenomena Associated With Degraded Core (F)
II.B.5(2)	Safety Review Consideration--Research on Phenomena Associated With Degraded Core (F)
II.B.5(3)	Safety Review Consideration--Research on Phenomena Associated With Degraded Core (F)
II.E.2.2(1)	Research on Small-Break LOCAs and Anomalous Transients (F)
II.E.2.2(3)	Research on Small-Break LOCAs and Anomalous Transients (F)
II.E.2.2(4)	Research on Small-Break LOCAs and Anomalous Transients (F)
II.E.2.2(6)	Research on Small-Break LOCAs and Anomalous Transients (F)
II.H.4(1)	Determine Impact of TMI on Socioeconomic and Real Property Values
II.H.4(2)	Determine Impact of TMI on Socioeconomic and Real Property Values
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections
II.J.1.2	Modify Existing Vendor Inspection Programs
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers
II.J.2.1	Reorient Construction Inspection Program
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program
II.J.2.3	Assign Resident Inspectors to All Construction Sites
III.C.1(1)	Public Information--Provide to News Media and Public
III.C.1(2)	Public Information--Provide to News Media and Public
III.C.1(3)	Public Information--Provide to News Media and Public

(F) = forecast issue

Table A-1b Code 2 - Licensing issues (continued)

Issue No.	Title
III.C.2(1)	Public Information--Provide Training
III.C.2(2)	Public Information--Provide Training
III.D.2.4(2)	Offsite Dose Measurements
III.D.2.6	Independent Radiological Measurements
III.D.3.2(1)	Health Physics Improvements
III.D.3.2(2)	Health Physics Improvements
III.D.3.2(3)	Health Physics Improvements
III.D.3.2(4)	Health Physics Improvements
III.D.3.5(1)	Radiation Worker Exposure Data Base
III.D.3.5(2)	Radiation Worker Exposure Data Base
III.D.3.5(3)	Radiation Worker Exposure Data Base
IV.A.1	Legislative Authority in Enforcement Process
IV.A.2	Revise Enforcement Policy
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensee
II.D.1(1)	NRC Staff Training
II.D.1(2)	NRC Staff Training
II.D.1(3)	NRC Staff Training
II.D.1(4)	NRC Staff Training
IV.E.1	Expand Research on Quantification of Safety Decision-Making
IV.E.2	Plan for Early Resolution of Safety Issues
IV.E.3	Plan for Resolving Issues at the Construction Permit Stage
IV.E.4	Resolve Generic Issues by Rulemaking
IV.G.1	Develop a Public Agenda for Rulemaking
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules

Table A-1b Code 2 - Licensing issues (continued)

Issue No.	Title
IV.G.3	Improve Rulemaking Procedures
IV.G.4	Study Alternatives for Improved Rulemaking Process
IV.H	NRC Participation in the Radiation Policy Council
67.5.1	Steam Generator Staff Actions--Reassessment of Radiological Consequences
67.5.2	Steam Generator Staff Actions--Reevaluation of Steam Generator Tube Rupture Design Basis
67.10.0	Steam Generator Staff Actions--Supplemental Tube Inspections
119.5	Piping Review Committee Recommendations--Leak Detection Requirements (F)

(F) = forecast issue

Table A-1c Code 3 - Not current standardized design issues

Issue No.	Title
A-5	B&W Steam Generator Tube Integrity
A-6	Mark I Short-Term Program
A-7	Mark I Long-Term Program
A-8	Mark II Containment Pool Dynamic Loads--Long-Term Program
B-54	Ice Condenser Containments
B-55	Improved Reliability of Target Rock Safety-Relief Valves
II.E.5.1	Design Evaluation of B&W Reactors
II.E.5.2	B&W Reactor Transient Response Task Force
II.K.1(2)	Measures to Mitigate Small-Break LOCAs and Feedwater Accidents--IE Bulletins
II.K.1(7)	Measures to Mitigate Small-Break LOCAs and Feedwater Accidents--IE Bulletins
II.K.1(8)	Measures to Mitigate Small-Break LOCAs and Feedwater Accidents--IE Bulletins
II.K.1(18)	Measures to Mitigate Small-Break LOCAs and Feedwater Accidents--IE Bulletins
II.K.1(19)	Measures to Mitigate Small-Break LOCAs and Feedwater Accidents--IE Bulletins
II.K.1(20)	Measures to Mitigate Small-Break LOCAs and Feedwater Accidents--IE Bulletins
II.K.1(21)	Measures to Mitigate Small-Break LOCAs and Feedwater Accidents--IE Bulletins
II.K.2(1)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(2)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(3)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(4)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(5)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(6)	Commission Orders on B&W Plants to Mitigate Accidents

Table A-1c Code 3 - Not current standardized design issues (continued)

Issue No.	Title
II.K.2(7)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(9)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(10)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(11)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(13)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(14)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(15)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(20)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.2(21)	Commission Orders on B&W Plants to Mitigate Accidents
II.K.3(7)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(14)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(29)	Final Recommendations of B&O Task Force to Mitigate Accidents
61	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments
69	Makeup Nozzle Cracking in B&W Plants
80	Pipe Break Effects on Control Rod Drive Hydraulic Lines in Drywells in BWR Mark I and II Containments
87	Failure of HPCI Steam Line Without Isolation
100	OTSG Level
115	Reliability of Westinghouse Solid-State Protection System (F)

(F) = forecast issue

Table A-1d Code 4 - Operating plants only issues

Issue No.	Title
A-46	Seismic Qualification of Equipment in Operating Plants
B-56	Diesel Reliability (F)
II.B.6	Risk Reduction for Operating Reactors at Sites With High Population
II.C.1	Interim Reliability Evaluation Program (IREP)
II.C.2	Continuation of IREP
II.K.3(20)	Final Recommendations of B&O Task Force to Mitigate Accidents
IV.E.5	Assess Currently Operating Reactors
24	Automatic Emergency Core Cooling System Switch to Recirculation (F)
38	Potential Recirculation System Failure as a Consequence of Injection of Containment Paint Flakes or Other Fine Debris (F)
48	LC0 for Class 1E Vital Instrument Buses in Operating Reactors
49	Interlocks and LC0s for Redundant Class 1E Tie Breakers (F)
68	Postulated Loss of Auxiliary Feedwater System Resulting From Turbine-Driven Auxiliary Feedwater Pump Steam Supply Line Rupture (F)
70	PORV and Block Valve Reliability (F)
72	Control Rod Drive Tube Support Pin Failures (F)
74	Reactor Coolant Activity Limits for Operating Reactor (F)
77	Flooding of Safety Equipment Compartments by Backflow Through Floor Drains
95	Loss of Effective Volume for Containment Recirculation Spray (F)
106	Piping and Use of Highly Combustible Gases in Vital Areas (F)

(F) = forecast issue

Table A-1e Code 5 - Not applicable to plant design issues

Issue No.	Title
B-2	Forecasting Electricity Demands
B-28	Radionuclide/Sediment Transport Program
B-38	Reconnaissance Level Investigations
B-43	Value of Aerial Photographs for Site Evaluation
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants
B-61	Allowable ECCS Equipment Outage Periods
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection
I.A.1.1	Operational Safety--Shift Technical Advisor
I.A.1.2	Operational Safety--Shift Technical Administrative Duties
I.A.1.3	Operational Safety--Shift Manning
I.A.1.4	Operational Safety--Long-Term Upgrading
I.A.2.1(1)	Immediate Upgrading of Operator and Senior Operator Training
I.A.2.1(2)	Immediate Upgrading of Operator and Senior Operator Training and Qualifications
I.A.2.1(3)	Immediate Upgrading of Operator and Senior Operator Training and Qualifications
I.A.2.2	Training and Qualification Requirements for Operations Personnel
I.A.2.3	Administration of Training Programs
I.A.2.5	Training and Qualification of Operating Personnel Plant Drills
I.A.2.6(2)	Long-Term Upgrading of Training and Qualifications
I.A.2.6(4)	Long-Term Upgrading of Training and Qualifications
I.A.2.6(5)	Long-Term Upgrading of Training and Qualifications
I.A.2.6(6)	Long-Term Upgrading of Training and Qualifications
I.A.2.7	Accreditation of Training Institutions
I.A.3.1	Revise Scope and Criteria for Licensing Exams

Table A-1e Code 5 - Not applicable to plant design issues (continued)

Issue No.	Title
I.A.3.2	Operator Licensing Program Changes
I.A.3.4	Licensing of Additional Operations Personnel
I.A.4.1(1)	Training Simulator Improvement--Initial
I.A.4.1(2)	Training Simulator Improvement--Initial
I.A.4.2(2)	Training Simulator Improvements--Initial
I.A.4.2(3)	Long-Term Training Simulator Upgrade
I.B.1.1(5)	Management for Operations--Long-Term Improvements
I.B.1.2(1)	Management for Operations--Evaluation of NTOL Applicants
I.B.1.2(2)	Management for Operations--Evaluation of NTOL Applicants
I.B.1.2(3)	Management for Operations--Evaluation of NTOL Applicants
I.C.1(1)	Short-Term Accident Analysis Procedures Revision
I.C.1(2)	Short-Term Accident Analysis Procedures Revision
I.C.1(3)	Short-Term Accident Analysis Procedures Revision
I.C.1(4)	Short-Term Accident Analysis Procedures Revision
I.C.2	Shift and Relief Turnover Procedures
I.C.3	Shift Supervisor Responsibilities
I.C.4	Operating Procedures--Control Room Access
I.C.5	Procedures for Feedback of Operating Experience
I.C.6	Procedures for Verification of Correct Performance of Operating Activities
I.C.7	NSSS Vendor Review of Operating Procedures
I.C.8	Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants
I.G.1	Scope of Test Program--Preoperational and Low-Power Testing
I.G.2	Scope of Test Program--Preoperational and Low-Power Testing
II.A.1	Siting Policy Reformation

Table A-1e Code 5 - Not applicable to plant design issues (continued)

Issue No.	Title
II.B.4	Safety Review Consideration--Training to Mitigate Core Damage
II.C.4	Risk Assessment--Reliability Engineering (F)
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure
II.K.3(3)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(17)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(30)	Final Recommendations of B&O Task Force to Mitigate Accidents
II.K.3(31)	Final Recommendations of B&O Task Force to Mitigate Accidents
III.A.1.1(1)	Upgrade Emergency Preparedness
III.A.1.1(2)	Upgrade Emergency Preparedness
III.A.1.3(1)	Maintain Supplies of Thyroid Blocking Agent
III.A.1.3(2)	Maintain Supplies of Thyroid Blocking Agent
III.A.2.1(1)	Amendment to 10 CFR 50 and Appendix E
III.A.2.1(2)	Amendment to 10 CFR 50 and Appendix E
III.A.2.1(3)	Amendment to 10 CFR 50 and Appendix E
III.A.2.1(4)	Amendment to 10 CFR 50 and Appendix E
III.A.2.2	Development of Guidance and Criteria
III.D.3.1	Radiation Protection Plans (F)
III.D.3.3(3)	Inplant Radiation Monitoring
III.D.3.3(4)	Inplant Radiation Monitoring
10	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges
88	Earthquakes and Emergency Planning
90	Technical Specifications for Anticipatory Trips

(F) = forecast issue

Table A-1e Code 5 - Not applicable to plant design issues (continued)

Issue No.	Title
B-2	Forecasting Electricity Demands
B-28	Radionuclide/Sediment Transport Program
B-38	Reconnaissance Level Investigations
B-43	Value of Aerial Photographs for Site Evaluation
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants
B-61	Allowable ECCS Equipment Outage Periods
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection
I.A.1.1	Operational Safety--Shift Technical Advisor
I.A.1.2	Operational Safety--Shift Supervisor Administrative Duties
I.A.1.3	Operational Safety--Shift Manning
I.A.1.4	Operational Safety--Long-Term Upgrading
I.A.2.1(1)	Immediate Upgrading of Operator and Senior Operator Training
I.A.2.1(2)	Immediate Upgrading of Operator and Senior Operator Training and Qualifications
I.A.2.1(3)	Immediate Upgrading of Operator and Senior Operator Training and Qualifications
I.A.2.2	Training and Qualification Requirements for Operations Personnel
I.A.2.3	Administration of Training Programs
I.A.2.5	Training and Qualification of Operating Personnel Plant Drills
I.A.2.6(2)	Long-Term Upgrading of Training and Qualifications
I.A.2.6(4)	Long-Term Upgrading of Training and Qualifications
I.A.2.6(5)	Long-Term Upgrading of Training and Qualifications
I.A.2.6(6)	Long-Term Upgrading of Training and Qualifications
I.A.2.7	Accreditation of Training Institutions

(F) = forecast issue

Table A-1e Code 5 - Not applicable to plant design issues (continued)

Issue No.	Title
I.A.3.1	Revise Scope and Criteria for Licensing Exams
I.A.3.2	Operator Licensing Program Changes
I.A.3.4	Licensing of Additional Operations Personnel
I.A.4.1(1)	Training Simulator Improvement--Initial
I.A.4.1(2)	Training Simulator Improvement--Initial
I.A.4.2(2)	Training Simulator Improvements--Long Term
I.A.4.2(3)	Long-Term Training Simulator Upgrade
I.B.1.1(5)	Management for Operations--Long-Term Improvements
I.B.1.2(1)	Management for Operations--Evaluation of NTOL Applicants
I.B.1.2(2)	Management for Operations--Evaluation of NTOL Applicants
I.B.1.2(3)	Management for Operations--Evaluation of NTOL Applicants
I.C.1(1)	Short-Term Accident Analysis Procedures Revision
I.C.1(2)	Short-Term Accident Analysis Procedures Revision
I.C.1(3)	Short-Term Accident Analysis Procedures Revision
I.C.1(4)	Short-Term Accident Analysis Procedures Revision
I.C.2	Shift and Relief Turnover Procedures
I.C.3	Shift Supervisor Responsibilities
I.C.4	Operating Procedures--Control Room Access
I.C.5	Procedures for Feedback of Operating Experience
I.C.6	Procedures for Verification of Correct Performance of Operating Activities
I.C.7	NSSS Vendor Review of Operating Procedures
I.C.8	Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants
I.G.1	Scope of Test Program--Preoperational and Low-Power Testing

(F) = forecast issue

Table A-1e Code 5 - Not applicable to plant design issues (continued)

Issue No.	Title
I.G.2	Scope of Test Program--Preoperational and Low-Power Testing
II.A.1	Siting Policy Reformation
II.B.4	Safety Review Consideration--Training to Mitigate Core Damage
II.C.4	Risk Assessment--Reliability Engineering (F)
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure
II.K.3(3)	Final Recommendations of B&O Task Force to Mitigate Accidents
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III.A.1.3(2)	Maintain Supplies of Thyroid Blocking Agent
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III.A.2.1(2)	Amendment to 10 CFR 50 and Appendix E
III.A.2.1(3)	Amendment to 10 CFR 50 and Appendix E
III.A.2.1(4)	Amendment to 10 CFR 50 and Appendix E
III.A.2.2	Development of Guidance and Criteria
III.D.3.1	Radiation Protection Plans (F)
III.D.3.3(3)	Inplant Radiation Monitoring
III.D.3.3(4)	Inplant Radiation Monitoring
10	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges

(F) = forecast issue

Table A-1e Code 5 - Not applicable to plant design issues (continued)

Issue No.	Title
88	Earthquakes and Emergency Planning
90	Technical Specifications for Anticipatory Trips
HF-01.1.1	Staffing and Qualifications--Policy Statement on Engineering Expertise on Shift and Evaluate Effectiveness of Policy Statement (F)
HF-01.1.2	Staffing and Qualifications--Revise and Evaluate Changes to Regulatory Guide 1.8 (F)
HF-01-1.3	Staffing and Qualifications--Develop a Means to Evaluate Acceptability of Nuclear Power Plant Personnel Qualifications Program (F)
HF-01.1.4	Staffing and Qualifications--Review and Evaluate Industry Programs
HF-01-2.1	Training--Evaluate Industry Training (F)
HF-01.2.2	Training--Evaluate INPO Accreditation Program (F)
HF-01.2.3	Training--Revise Standard Review Plan Section 13.2.3 (F)
HF-01.3.1	Licensing Examinations--Develop Job Knowledge Catalogue (F)
HF-01.3.2	Licensing Examinations--Develop Licensing Examinations Handbook (F)
HF-01.3.3	Licensing Examinations--Develop Criteria for Nuclear Power Plant Simulators (F)
HF-01.3.4	Licensing Examinations--Training Requirements Package (Review 10 CFR 55 and Regulatory Guides 1.149 and 1.8) (F)
HF-01.3.5	Licensing Examinations--Develop Computerized Exam System (F)
HF-01.4.1	Procedures--Inspection Module for Upgrading Procedures (F)
HF-01.4.2	Procedures--Engineering Operating Procedure Effectiveness Evaluation (F)
HF-01.4.3	Procedures--Criteria for Safety Related Operator Actions (GI #B-17) (F)
HF-01.4.4	Procedures--Guidelines for Upgrading Other Procedures (F)

(F) = forecast issue

Table A-1e Code 5 - Not applicable to plant design issues (continued)

Issue No.	Title
HF-01.6.1	Management and Organization--Development of Regulatory Position on Management and Organization (F)
HF-01.6.2	Management and Organization--Evaluate Criteria for SALP Review (F)
HF-01.6.3	Management and Organization--Revise Standard Review Plan 13.1 (F)
HF-02.I.1	Survey of Current Maintenance Practices (F)
HF-02.I.2	Maintenance and Performance Indicators (F)
HF-02.I.3	Monitor Industry Activities (F)
HF-02.I.4	Participate in Standards Group (F)
HF-02.I.5	Maintenance and Surveillance Program Integration (F)
HF-02.I.6	Analysis of Japanese/U.S. Nuclear Power Plant Maintenance Programs (F)
HF-02.I.7	Maintenance Personnel Qualifications (F)
HF-02.I.8	Human Factors in In-Service Inspections (F)
HF-02.I.9	Human Error in Events Involving Wrong Unit or Wrong Train (F)
HF-02.II.1	Phase II Tasks to Be Determined After Resolution of Phase I (F)

(F) = forecast issue

Table A-1f Code 6 - Regulatory impact issues

Issue No.	Title
A-23	Containment Leak Testing
B-31	Dam Failure Model (F)
B-37	Chemical Discharge to Receiving Waters (F)
B-50	Post-Operating-Basis-Earthquake Inspection
B-53	Load Break Switch
B-59	Review of (N-1) Loop Operation in BWRs and PWRs
C-4	Statistical Methods for ECCS Analysis (F)
C-5	Decay Heat Update (F)
C-6	LOCA Heat Sources (F)
C-8	Main Steam Line Leakage Control System (F)
59	Technical Specification Requirements for Plant Shutdown When Equipment for Safe Shutdown Is Degraded or Inoperable (F)
67.2.1	Steam Generator Staff Actions--Integrity of Steam Generator Tube Sleeves
67.8.0	Steam Generator Staff Actions--Denting Criteria
104	Reduction of Boron Dilution Requirements (F)
108	BWR Suppression Pool Limits
112	Westinghouse Reactor Protection System Surveillance Frequencies and Out-of-Service Times (F)
119.1	Piping Review Committee Recommendations--Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads (F)
119.2	Piping Review Committee Recommendations--Piping Damping Valves (F)
119.3	Piping Review Committee Recommendations--Decoupling of the Design-Basis Earthquake (DBE) From the Safe Shutdown Earthquake (F)
119.4	Piping Review Committee Recommendations--BWR Materials (F)

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This report summarizes the NRC/EPRI coordinated effort to develop design requirements for a standardized advanced light water reactor (ALWR) and the procedures for screening and applying new generic safety issues to this program. The end-product will be an NRC-approved ALWR Requirements Document for use by the nuclear industry in generating designs of LWRs to be constructed for operation in the 1990s and beyond.

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