GE Nuclear Energy

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

February 17, 1993 Docket No. STN 52-001

Chet Poslusny, Senior Project Manager
Standa. dization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of the Nuclear Reactor Regulation

Subject: **Submittal Supporting Accelerated ABWR Review Schedule - Chapter 17 COL Action Items**

Dear Chet:

Enclosed are SSAR markups addressing COL Action Items 17.1.1-1, 17.2, 17.3.1-1, 17.3.5-1 and 17.3.9-1. It should be noted that the change reflected in these markups are in addition to those proposed in my January 19, 1993 letter which addresses Open Item 17.3.5-1.

Please provide a copy of this tranmittal to Tim Polich.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Norman Fletcher (DOE)
Cal Tang (GE)

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17.0 INTRODUCTION

Section 17.1 of this Standard Safety Analysis Report describes the Quality Assurance (QA) Program which is implemented by GE for the ABWR project. It is based upon the standard GE QA Program documented in the GE Nuclear Energy topical report NEDO-11209-04A (Reference 1) and the additional information in this chapter describing and clarifying GE's interfaces and responsibilities with its technical associates on the ABWR. These technical associates are major international corporations who are licensees of GE's technology and have extensive independent experience in the design and construction of nuclear power stations.

The standard program is used throughout GE Nuclear Energy on all other nuclear power plant work and has been accepted by the Nuclear Regulatory Commission. It is in compliance with Title 10, Code of Federal Regulations, Part 50, Appendix B; ANSI/ASME N45.2; ANSI/ASME N45.2-series standards; and NRC Regulatory Guides with some NRC-accepted GE Nuclear Energy alternate positions.

The QA Program described in this chapter meets Regulatory Guide 1.28, Revision 3 and is organized to show its relationship to Reference 1, ANSI/ ASME NQA-1-1983 and NQA-1a-1983, and GE's interfaces with its technical associates. The terms and definitions of supplement S-1 of NQA-1a-1983 apply. Table 17.0-1 summarizes ABWR compliance with the quality related Regulatory Guides.

¶ The COL applicant/holder is responsible to prepare and implement a quality assurance program for the construction phase of Section 17.1 and the operations phase of Section 17.2 that also meets the requirements of ANSI/ASME NQA-1-1983 and NQA-1a-1983. See Subsection 17.0.1.1 for COL license information.

17.0.1 COL License Information

17.0.1.1 QA Programs for Construction and Operation

The COL applicant/holder shall prepare and implement a quality assurance program for the construction phase of Section 17.1 and the operations phase of Section 17.2. They will meet the requirements of ANSI/ASME NQA-1-1983 and NQA-1a-1983. (See Section 17.0)

17.3 RELIABILITY ASSURANCE PROGRAM DURING DESIGN PHASE

This section presents the ABWR Design Reliability Assurance Program (D-RAP).

17.3.1 Introduction

The ABWR Design Reliability Assurance Program (D-RAP) is a program that will be performed by GE Nuclear Energy (GE-NE) during detailed design and specific equipment selection phases to assure that the important ABWR reliability assumptions of the probabilistic risk assessment (PRA) will be considered throughout the plant life. The plant owner/operator will also have an operational RAP (O-RAP) that tracks equipment reliability to demonstrate that the plant is being operated and maintained consistent with PRA assumptions so that overall risk is not unknowingly degraded. The PRA evaluates the plant response to initiating events to assure that plant damage has a very low probability and risk to the public is very low. Input to the PRA includes details of the plant design and assumptions about the reliability of the plant risk-significant structures, systems and components (SSCs) throughout plant life.

The D-RAP will include the design evaluation of the ABWR. It will identify relevant aspects of plant operation, maintenance, and performance monitoring of important plant SSCs for owner/operator consideration in assuring safety of the equipment and limited risk to the public. The policy and implementation procedures will be specified by the owner/operator.

Also included in this explanation of the D-RAP is a descriptive example of how the D-RAP will apply to one potentially important plant system, the standby liquid control system (SLCS). The SLCS example shows how the principles of D-RAP will be applied to other systems identified by the PRA as being significant with respect to risk.

17.3.2 Scope

The ABWR D-RAP will include the future design evaluation of the ABWR, and it will identify relevant aspects of plant operation, maintenance, and performance monitoring of plant risk-significant SSCs. The PRA for the ABWR and other industry sources will be used to identify and prioritize those SSCs that are important to prevent or mitigate plant transients or other events that could present a risk to the public.

17.3.3 Purpose

The purpose of the D-RAP is to assure that the plant safety as estimated by the probabilistic risk analysis (PRA) is maintained as the detailed design evolves through the implementation and procurement phases and that pertinent information is provided in the design documentation to the future owner/operator so that equipment reliability, as it affects plant safety, can be maintained through operation and maintenance during the entire plant life.

17.3.4 Objective

The objective of the D-RAP is to identify those plant SSCs that are significant contributors to risk, as shown by the PRA or other sources, and to assure that, during the implementation phase, the plant design continues to utilize risk- significant SSCs whose reliability is commensurate with the PRA assumptions. The D-RAP will also identify key assumptions regarding any operation, maintenance and monitoring activities that the owner/operator should consider in developing its O-RAP to assure that such SSCs can be expected to operate throughout plant life with reliability consistent with that assumed in the PRA.

A major factor in plant reliability assurance is risk-focused maintenance, by which maintenance resources are focused on those SSCs that enable the ABWR systems to fulfill their essential safety functions and on SSCs whose failure may directly initiate challenges to safety systems. All plant modes are considered, including equipment directly relied upon in Emergency Operating Procedures (EOPs). Such a focus of maintenance will help to maintain an acceptably low level of risk, consistent with the PRA.

17.3.5 GE-NE Organization for D-RAP

The relevant portion of the GE-NE organization chart for a future ABWR D-RAP is shown in Figure 17.3-1. The

The COL applicant will specify the policy and implementation procedures for using the D-RAP information. See subsection 17.3.13.1 for COL license information.

Managers of the Nuclear Services and Projects Department and of the Nuclear Operations Department report to the Vice President and General Manager of GE Nuclear Energy. Two sections involved with an ABWR D-RAP are the Advanced Reactor Programs Section and the Engineering Services Section.

Authority for the management of an ABWR program is centered with the Advanced Reactor Programs Manager. Day-to-day details of an ABWR program are directed by the Project Manager, who reports to the Advanced Reactor Programs Manager. The Project Manager and his staff coordinate both the GE-NE support for the Project and the work of external organizations, such as the Architect Engineer.

Responsibility for the design of key equipment, components and subsystems is shared by the several units in the Advanced Reactor Programs Section together with external organizations, including the Architect Engineer. Reporting directly to each engineering functional manager will be performing engineers, including system designers and component designers. Design support will also be provided by other design sections within GE-NE and the Nuclear Services and Projects Department. Responsibility for ABWR safety analysis and PRA studies is under the Systems Integration and Performance Engineering Unit.

The Manager, System Integration and Performance Engineering, will be assigned the responsibility of managing and integrating the D-RAP Program. He will have direct access to the ABWR Project Manager and will keep him abreast of D-RAP critical items, program needs and status. He has organizational freedom to:

(1) Identify D-RAP problems.
(2) Initiate, recommend or provide solution to problems through designated organizations.
(3) Verify implementation of solution.
(4) Function as an integral part of the final design process.

Reliability analyses, including the PRA, are performed by the Reliability Engineering Services Unit in the Reliability and Analysis Services Subsection of the Engineering Services Section (Figure 17.3-1). Thus, the PRA input to the D-RAP and many of the ABWR reliability analyses will be performed in this organization, within the Nuclear Operations Department. Responsibility for reliability review of designed ABWR systems and components also falls on the Reliability Engineering Services Unit, under direction from the Systems Integration and Performance Engineering Unit.

17.3.6 SSC Identification /Prioritization

The PRA prepared for the ABWR will be the primary source for identifying risk- significant SSCs that should be given special consideration during the detailed design and procurement phases and/or considered for inclusion in the O-RAP. The method by which the PRA is used to identify risk-significant SSCs is described in Chapter 19. It is also possible that some risk-significant SSCs will be identified from sources other than the PRA, such as nuclear plant operating experience, other industrial experience, and relevant component failure data bases.

17.3.7 Design Considerations

The reliability of risk-significant SSCs, which are identified by the PRA, will be evaluated at the detailed design stage by appropriate design reviews and reliability analyses. Current data bases will be used to identify appropriate values for failure rates of equipment as designed, and these failure rates will be compared with those used in the PRA. Normally the failure rates will be similar, but in some cases they may differ because of recent design or data base changes. Whenever failure rates of designed equipment are significantly greater than those assumed in the PRA, an evaluation will be performed to determine if the equipment is acceptable or if it must be redesigned to achieve a lower failure rate.

For those risk-significant SSCs, as indicated by PRA or other sources, component redesign (including selection of a different component) will be considered as a way to reduce the CDF contribution. (If the system unavailability or the CDF is acceptably low, less effort will be expended toward redesign.) If there are practical ways to redesign a risk-significant SSC, it will be redesigned and the change in system fault tree results will be calculated. Following the redesign phase, dominant SSC failure modes will be identified so that protection against such failure modes can be accomplished by appropriate activities during plant life. The design considerations that will go into determining an acceptable, reliable design and the

¶ The COL applicant completing its detailed design and equipment selection during the design phase, must submit its specific D-RAP organization for NRC review. See Subsection 17.3.13.2 for COL license information.

COL 17.3.9-1

SSCs that must be considered for O-RAP activities are shown in Figure 17.3-2.

GE-NE will identify in the PRA or other design documents to the plant owner/operator the risk-significant SSCs and the associated reliability assumptions, including any pertinent bases and uncertainties considered in the PRA. GE-NE will also provide information for the plant owner/operator to incorporate into the O-RAP to help assure that PRA results will be achieved over the life of the plant. This information can be used by the owner/operator for establishing appropriate reliability targets and the associated maintenance practices for achieving them.

17.3.8 Defining Failure Modes

The determination of dominant failure modes of risk-significant SSCs will include historical information, analytical models and existing requirements. Many BWR systems and components have compiled a significant historical record, so an evaluation of that record comprises Assessment Path A in Figure 17.3-3. Details of Path A are shown in Figure 17.3-4.

For those SSCs for which there is not an adequate historical basis to identify critical failure modes, an analytical approach is necessary, shown as Assessment Path B in Figure 17.3-3. The details of Path B are given in Figure 17.3-5. The failure modes identified in Paths A and B are then reviewed with respect to the existing maintenance activities in the industry and the maintenance requirements, Assessment Path C in Figure 17.3-3. Detailed steps in Path C are outlined in Figure 17.3-6.

17.3.9 Operational Reliability Assurance Activities

Once the dominant failure modes are determined for risk-significant SSCs, an assessment is required to determine suggested O-RAP activities that will assure acceptable performance during plant life. Such activities may consist of periodic surveillance inspections or tests, monitoring of SSC performance, and/or periodic preventive maintenance (Ref. 1). An example of a decision tree that would be applicable to these activities is shown in Figure 17.3-7. As indicated, some SSCs may require a combination of activities to assure that their performance is consistent with that assumed in the PRA.

Periodic testing of SSCs may include startup of standby systems, surveillance testing of instrument circuits to assure that they will respond to appropriate signals, and inspection of passive SSCs (such as tanks and pipes) to show that they are available to perform as designed. Performance monitoring, including condition monitoring, can consist of measurement of output (such as pump flow rate or heat exchanger temperatures), measurement of magnitude of an important variable (such as vibration or temperature), and testing for abnormal conditions (such as oil degradation or local hot spots).

Periodic preventive maintenance is an activity performed at regular intervals to preclude problems that could occur before the next PM interval. This could be regular oil changes, replacement of seals and gaskets, or refurbishment of equipment subject to wear or age related degradation.

Planned maintenance activities will be integrated with the regular operating plans so that they do not disrupt normal operation. Maintenance that will be performed more frequently than refueling outages must be planned so as to not disrupt operation or be likely to cause reactor scram, ESF actuation, or abnormal transients. Maintenance planned for performance during refueling outages must be conducted in such a way that it will have little or no impact on plant safety, on outage length or on other maintenance work.

17.3.10 Owner/Operator's Reliability Assurance Program

The O-RAP that will be prepared and implemented by the ABWR owner/operator will make use of the information provided by GE-NE. This information will help the owner/operator determine activities that should be included in the O-RAP. Examples of elements that might be included in an O-RAP are:

1. <u>Reliability Performance Monitoring</u>: Measurement of the performance of equipment to determine that it is accomplishing its goals and/or that it will continue to operate with low probability of failure.

2. <u>Reliability Methodology</u>: Methods by which the plant owner/operator can compare plant data to the SSC data in the PRA.

(a) The COL applicant will provide a complete O-RAP to be reviewed by the NRC. See Subsection 17.3.13.3 for COL license information.

Risk-Significant Those SSCs which are identified as contributing significantly to the system unavailability.

SSCs Structures, systems and components identified as being important to the plant operation and safety.

17.3.~~12~~ 14 Reference

(1) E. V. Lofgren, et. al., *A Process for Risk-Focused Maintenance*, SAIC, NUREG/CR-5695, March 1991.

17.3.13 COL License Information

17.3.13.1 Policy and Implementation Procedures for D-RAP

The COL applicant will specify the policy and implementation procedures for using D-RAP information. (See Subsection 17.3.1)

17.3.13.2 D-RAP Organization

The COL applicant completing its detailed design and equipment selection during the design phase, must submit its specific D-RAP organization for NRC review. (See Subsection 17.3.5)

17.3.13.3 Provision for O-RAP

The COL applicant will provide a complete O-RAP to be reviewed by the NRC. (See Subsection 17.3.9)