GE Nuclear Energy

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

February 23, 1993

Docket No. STN 52-001

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

Subject: **Submittal Supporting Accelerated ABWR Review Schedule - Resubmittal of Open Item 15.3-1 and COL Action Items 17.1.1-1 and 17.2-1**

Dear Chet:

Enclosed are SSAR markups revising our responses to Open item 15.3-1 and COL Action Items 17.1.1-1 and 17.2-1 originally transmitted in my letters dated February 16 and 17, 1993.

Please provide a copy of this tranmittal to George Thomas and Tim Polich.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Norman Fletcher (DOE)
Bob Huang (GE)
Phil Novak (GE)

JF93-36

OI 15.3-1 REVISED

worst possible location, and the plant is operated with the mislocated bundle. This event is categorized as a limiting fault based on the following data:

Expected Frequency: 0.002 events/operating cycle.

This number is based upon past experience.

15.4.7.2 Sequence of Events and Systems Operation

15.4.7.2.1 Sequence of Events

The postulated sequence of events for the misplaced bundle accident (MBA) is presented in Table 15.4-6.

15.4.7.2.2 Systems Operation

A fuel loading error, undetected by in-core instrumentation following fueling operations, may result in an undetected reduction in thermal margin during power operations. For the analysis reported herein, no credit for detection is taken and, therefore, no corrective operator action or automatic protection system functioning is assumed to occur.

15.4.7.3 Core and System Performance

This event is presented in Subsection S.2.5.4 of Reference 1.

Mislocated bundle analyses are not performed for reload cores because, based on analysis of data available from past reloads, the probability that a mislocated fuel bundle loading error will result in a CPR less than the safety limit is sufficient small (see Reference S.2-58 of Reference 1).

For ABWR initial core, the mismatch of exposures and integrated bundle power between mislocated bundles are less severe than the equilibrium cycle. Therefore, the consequence of a postulated misplaced bundle accident for the initial core is less severe than that for the equilibrium cycle. Consequently, the conclusion drawn from the reload core analysis as previously presented is applicable to the ABWR initial core. Hence, no specific analysis is required.

INSERT A

15.4.7.4 Barrier Performance

An evaluation of the barrier performance is not made for this event, because it is a mild and highly localized event. No perceptible change in the core pressure is observed.

15.4.7.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released for the fuel.

15.4.8 Rod Ejection Accident

15.4.8.1 Identification of Causes and Frequency Classification

The rod ejection accident is caused by a major break on the FMCRD housing, outer tube or associated CRD pipe lines. Due to a break of this type, the reactor pressure exerted on the CRD spud pushes down the hollow piston and the ballnut with a large force. The shaft screw and the motor are forced to unwind. A passive brake mechanism is installed in the FMCRD system to prevent the control rod from moving. The design of the brake is presented in Section 4.6.1. The probability of the initial causes, i.e., a CRD pipe line break or housing break, is considered low enough to warrant its being categorized as a limiting fault. Even if this accident does happen, the brake prevents the control rod from ejection. Should the brake fail, the check valve will serve as a backup brake to prevent the rod ejection.

15.4.8.2 Sequence of Events and Systems Operation

If a major break occurs on the FMCRD housing, the reactor pressure will provide forces that could cause the shaft screw to unwind. The FMCRD brake mechanism prevents the rod from moving. Therefore, no rod ejection can occur.

The COL applicant will provide an analysis to confirm that the consequences of a fuel misorientation event meet all requirements approved by the NRC. See subsection 15.4.10.1 for COL license information.

INSERT A

The other potential type of bundle loading error that can occur is the mis-oriented fuel bundle (MOFB). In this case, the bundle is in the correct location but is rotated by 90 or 180 degrees. In reactors where the water gaps are non-uniform around the bundle or where the rod enrichment distribution is not quadrant-symmetrical, rotation can cause increases in local rod power through increased moderation. In ABWR lattice, the rotation results in non-uniform water gaps and produces similar increases in local rod power.

The initiator for operating a reactor with a MOFB is an operator placing the bundle into the core in a mis-oriented position. The next step in the accident progression is failure to detect the MOFB. A verification process is recommended to detect a MOFB. This verification procedure requires two core scans. One scan is with an underwater TV camera positioned close enough to read the bundle serial numbers on top of the lifting bail (first attribute) and to check the orientation of the bosses (second attribute). The other scan is with a TV camera positioned sufficiently above the core to allow viewing one complete 4 bundle cell for the following four attributes: boss on lifting bail, channel fasteners, channel buttons, and "cells look alike". Two independent reviewers (checkers A and B) are recommended to verify video tapes from the above procedures.

A generic model was developed based on the recommended verification procedure to quantify the probability of operating a reactor with a MOFB. An event tree was constructed to find this probability using human error rates from NUREG/CR-1278. The results show that the probability of operating the reactor with a MOFB is 8.5×10^{-5} per cycle (5.7×10^{-5}/year with a 18 month fuel cycle). This probability of operation with a MOFB is lower than the probability of a large break LOCA (i.e., 10^{-4} per year).

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control rod from the hollow piston of the FMCRD. If the control rod is stuck, the separation-detection devices will detect the separation of the control rod and hollow piston from the ballnut of the FMCRD and rod block interlock will prevent further rod withdrawal. The operator will be alarmed for this separation.

There is no basis for the control rod drop event to occur.

15.4.9.3.2 Identification of Operator Actions

No operator actions are required to preclude this event. However, the operator will be notified by the separation-detection alarm if separation is detected.

15.4.9.4 Core and System Performance

The performance of the separation-detection devices and the rod block interlocks virtually preclude the cause of a rod drop accident.

15.4.9.5 Barrier Performance

An evaluation of the barrier performance is not made for this accident since there is no circumstance for which this event could occur.

15.4.9.6 Radiological Consequences

The radiological analysis is not required.

15.4.~~10~~ 11 References

1. *General Electric Standard Application for Reactor Fuel--United States Supplement*, NEDE-24011-P-A-US, (Latest approved revision).

2. C. J. Paone and J. A. Woolley, *Rod Drop Accident Analysis for Large Boiling Water Reactors*, Licensing Topical Report, March 1972 (NEDO-10527, Supplements 1 and 2).

15.4.10 COL License Information

15.4.10.1 Fuel Misorientation Event Analysis

COL applicants will provide an analysis to confirm that the consequences of a fuel misorientation event meet all requirements approved by the NRC. (See Subsection 15.4.7.3).

COL 17.1.1-1
& COL 17.2-1

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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 and the quality related Regulatory Guides listed in Table 17.0-1. (See Section 17.0)

(REVISION)