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Subject: **NEDO-33201, Revision 1, "ESBWR Probabilistic Risk Assessment," Section 1**

Enclosure 1 contains the subject partial ESBWR Probabilistic Risk Assessment (PRA) document (Revision 1). This completes GE's submittal of individual sections of the ESBWR PRA. We will provide the NRC with the complete complied document in the near future.

If you have any questions about the information provided here, please let me know.

Sincerely,

Kathy Sedney for

David H. Hinds
Manager, ESBWR

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Enclosure:

1. MFN 06-296 – NEDO-33201, Revision 1, "ESBWR Probabilistic Risk Assessment:"
    - Section 1 – Introduction

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ENCLOSURE 1

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NEDO-33201, Revision 1, "ESBWR Probabilistic Risk Assessment"

- **Section 1 – Introduction**

1 INTRODUCTION

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1 INTRODUCTION

1.1 PURPOSE

The purpose of this analysis is to describe the methodology and the results of the ESBWR probabilistic risk assessment (PRA) and severe accidents.

The PRA has been performed in an iterative manner with the ESBWR design development to evaluate and improve the safety aspects of the ESBWR design.

The overall objectives of the ESBWR PRA are:

- Provide an integrated and systematic assessment of the ESBWR design in response to transient and accident events (including severe accidents)
- Assess the capability of the ESBWR design to meet, with sufficient margin, the NRC safety goals for new plant designs
- Identify design and analysis areas where further investigation and/or improvement is needed to meet the safety goals
- Assess the sensitivity of the ESBWR risk profile to human interactions
- Identify the importance of individual systems and components to the ESBWR risk profile
- Develop an analytic tool for use in investigating alternatives in design and operational strategies to optimize ESBWR plant safety.

The specific objectives of the ESBWR design PRA and severe accident evaluations are as follows:

(1) Core damage frequency less than 10^{-4} per year.

(2) Large release frequency less than 10^{-6} per year with a Conditional Containment Failure Probability of 0.1 or less.

(3) Frequency of radiation dose of 25 rem at the site boundary below 10^{-6} per year.

(4) Overall risks lower than the conventional BWRs.

(5) Identify non-safety functions requiring enhanced regulatory oversight.

(6) Assess PRA findings not meeting the first 3 goals (e.g. single failures that would prevent meeting any of these goals).

The ESBWR design PRA uses the current information available from the ESBWR plant design, Technical Specifications, and procedures. Component failure data and initiating event frequencies are based on generic industry data with consideration of the ESBWR design. Given the state of the ESBWR plant design, the PRA analyses contain conservative elements (e.g., pre-initiator and post-initiator human error probabilities; maintenance unavailabilities; component failure rates; flood and fire initiation, propagation and effects; ground level release with no evacuation warning assumed in consequence analysis). As such, the actual ESBWR risk profile is judged to be lower than the current quantitative results described in this report.

1.2 SCOPE

The ESBWR PRA is a full scope (Level 1, Level 2, and Level 3) PRA, that covers both internal and external events, full power and shutdown. Where applicable, ASME-RA-S-2002 capability category 3 attributes are included in the analysis. Some of these attributes are not achievable at the certification stage of a nuclear power plant. For example, many aspects of assessing human actions cannot be analyzed in absence of a physical, operating plant and operation staff. In these cases, a bounding approach is taken to encompass potential sites, configurations, and operating organizations. In addition, any analyses requiring site-specific characteristics are treated in a bounding manner.

1.3 PRA OVERVIEW

1.3.1 Internal Events

Methodology used in the ESBWR Level 1 PRA is a linked fault tree approach.

Fault trees have been developed and evaluated for the major ESBWR front line and support systems to determine the unavailability on demand of emergency core cooling and decay heat removal systems. Transient and loss-of-coolant accident events have been consolidated into major accident event sequences that are described by the accident event trees. These event trees are used to calculate the frequency of core damage sequences by directly linking the fault trees and solving for the minimal cutsets.

Outcomes of the event trees are transferred to containment event trees for further treatment to determine frequencies of radioactive releases to the environment.

Results of the containment event tree analyses provide the necessary input to model and assess fission product transport through the drywell and containment; calculate fission product release fractions associated with containment release paths; and determine potential consequences associated with each fission product release category.

The characteristics of the internal events PRA are as follows:

- Initiating Events

  Transients, Loss of Preferred Power, Loss of Coolant Accidents, and special initiator categories are identified based on review of industry PRAs and guidance documents. These are modified based on specifics of the ESBWR design and expected operation.

  Initiating event frequencies are estimated based on generic industry data for operating BWRs.

- Accident Sequence Analysis

  Accident sequence event tree structures and end states are defined for each initiator category based on review of industry PRAs and guidance documents. These are modified based on specifics of the ESBWR design and expected operation.

  Event tree nodal inputs are system fault tree logic or nodal point estimates, as appropriate.

  Functional success criteria are based on analysis of ESBWR design and expected operation.

- Systems Analysis

  System fault trees are developed based on standard industry techniques and reflect the ESBWR system design. Systemic success criteria are based on analysis of the ESBWR design and expected operation.

- Human Reliability Analysis

  Pre-initiator and post-initiator human error probabilities are defined based on the ESBWR design and expected operation. The human error probabilities used in the model are screening values based on the time available to perform the various actions.

- Data Analysis

  Component failure probabilities are estimated based on generic industry data.

- Containment Performance Analysis

  Severe accident phenomena are explicitly addressed and are quantitatively treated. The Risk Oriented Accident Analysis Methodology is used to assess the containment response to severe accident phenomena. A linked fault tree approach is used to address the containment systems and the ability to prevent overpressurization from loss of decay heat removal.

  In order to support the consequence analysis, multiple radionuclide release categories are modeled.

- Consequence Analysis

  Source terms are defined based on ESBWR thermal hydraulic analysis. Bounding consequence analyses are performed, showing that the ESBWR design meets NRC safety goals with sufficient margin.

1.3.2 External Events

The external events portion of the PRA explicitly analyzes core damage accidents initiated during power and shutdown operation for the following hazards:

- Internal floods
- Internal fires
- High winds
- Seismic events

The external events analyses are bounding assessments that are meant to show significant design margin for these hazards. The frequencies of initiating events are based on generic industry data, and are applied in a bounding manner. The fault trees and event trees developed for the internal events evaluations are used in the external events analyses to the maximum extent possible, using logic flags that account for the common failures induced by the external hazard events.

The ESBWR seismic assessment is a seismic margins analysis. The analysis demonstrates the ESBWR plant and equipment can withstand an earthquake with a magnitude at least two times the safe shutdown earthquake.

1.3.3 Shutdown Risk

The shutdown and transition risk analysis includes an assessment of the internal event initiated core damage accidents occurring during shutdown operations. Initiator categories and frequencies are based on a review of industry studies, generic industry data, and consideration of ESBWR design and operation. In addition, a typical refueling outage time line is assumed.

1.4 PRA DOCUMENTATION ORGANIZATION

The ESBWR PRA is documented as follows:

Level 1 Analysis

- Initiating Event Analysis (Section 2)
- Accident Sequence Analysis (Section 3)
- Systems Analysis (Section 4)
- Data Analysis (Section 5)
- Human Reliability Analysis (Section 6)
- Core Damage Frequency Results (Section 7)
- CDF Uncertainty and Sensitivity Analysis (Section 11)

Level 2 Analysis

- Containment Systems Performance (Section 8)
- Source Term (Section 9)
- Severe Accident Phenomena (Section 21)

Level 3 Analysis

- Consequence Analysis (Section 10)

External Event Analysis

- Internal Fires Analysis (Section 12)
- Internal Flooding Analysis (Section 13)
- High Wind Analysis (Section 14)
- Seismic Margins Analysis (Section 15)

Low Power/Shutdown Analysis

- Shutdown Analysis (Section 16)

Results and Insights

- Integrated Results (Section 17)
- Insights Affecting ESBWR Design (Section 18)

Fault trees, event trees, and thermal hydraulic analyses are documented in supporting appendices.

In addition, the following other aspects of the risk assessment are documented:

- PRA-Based Reliability, Availability and Maintenance Analysis (Section 19)
- Regulatory Treatment of Non-Safety Systems (Section 20)