Westinghouse Electric Corporation

Water Reactor Divisions

Box 355
Pittsburgh Pennsylvania 15230

NS-EPR-2951
August 24, 1984

Docket No. STN-50-601

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Westinghouse Advanced Pressurized Water Reactor (WAPWR), RESAR-SP/90, PDA-Level PRA Scope/Schedule

Attention: K. T. Eccleston, Project Manager, SSPB
A. C. Thadani, Branch Chief, RRAB

Dear Mr. Denton:

In response to an NRC request made at a Westinghouse/NRC meeting held August 3, 1984, enclosed is an outline of the proposed scope and schedule for the Probabilistic Risk Assessment (PRA) to be submitted as part of the application for Preliminary Design Approval (PDA) of the WAPWR Nuclear Power Block design (i.e., RESAR-SP/90).

As discussed in this meeting, it is important to have an unambiguous mutual understanding and agreement as to what constitutes an appropriate level of detail for the PRA for a preliminary design well in advance of this submittal.

Westinghouse appreciates the NRC's desire to be involved in the WAPWR PRA during its development so as to expedite the review of the resulting documentation in an attempt to meet Westinghouse's desire to obtain the SER/PDA, referencable by a construction permit applicant, in early 1986.

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Your prompt response to this letter is requested. Please contact Douglas G. Bevard (412/374-5597) of my staff should you require further information.

Very truly yours,

E. P. Rahe, Jr., Manager
Nuclear Safety Department

MDB/kk
Enclosure

cc: C. O. Thomas
F. R. Miraglia, Jr.
R. Bernero
D. Eisenhut
R. Silver
T. P. Speis

ENCLOSURE

Westinghouse Advanced (WAPWR) PDA PRA Scope/Schedule

The WAPWR Probabilistic Risk Assessment (PRA) will address the intended NRC criteria in NUREG-1070 and will be completed and submitted in two phases. The report of the first phase to be delivered to the NRC in June, 1985 will cover the internal initiating events analysis and will address the requirements for the PDA phase. The external events analysis of the second phase will be completed at a later date when the data required to perform it will be more mature. A comprehensive PRA study will be carried out for both phases. The internal initiating events report will contain the following sections:

1. Internal Initiating Event Analysis
2. Accident Sequence Modeling
3. Plant Systems Analysis
4. Core Melt Quantification
5. Core and Containment Analysis
6. Consequence Analysis
7. Uncertainty Analysis
8. Plant Risk Analysis

To facilitate the NRC review of the final reports, the following meetings with the NRC PRA staff at the Westinghouse Nuclear Center are suggested:

1st Meeting: November, 1984

Objectives:

1. Review the decision making processes involved in selection of design options and alternatives.
2. Discuss the methodology and tools being used and to demonstrate the codes.

2nd Meeting: February, 1985

Objective:

1. Discuss the preliminary numerical results and the analysis of dominant risk contributors.

A detailed PRA PDA-level report outline follows.

WAPWR PRA PDA-Level Report Outline

1.0 INTERNAL INITIATING EVENT ANALYSIS

1.1 Internal Initiating Event Categorization
1.2 Internal Initiating Event Quantification

2.0 ACCIDENT SEQUENCE MODELING

2.1 Event Tree Guidelines

2.1.1 Event Tree Guidelines
2.1.2 Event Tree Node Definitions
2.1.3 Event Tree Success Criteria Definitions
2.1.4 Core Melt State Definitions
2.1.5 Node Success Criteria Definitions
2.1.6 Consequential Failure Model
2.1.7 Support State Model (See Note 1)

2.2 Event Tree Modeling

2.2.1 Transients
2.2.2 Loss of Offsite Power
2.2.3 Steam Generator Tube Rupture
2.2.4 Steamline Break
2.2.5 Small LOCA
2.2.6 Large LOCA
2.2.7 ATWS
2.2.8 Interfacing Systems LOCA
2.2.9 Vessel Failure
2.2.10 Total Loss of Auxiliary Cooling

3.0 PLANT SYSTEMS ANALYSIS

3.1 AC Power
3.2 Integrated Protection System
3.3 Service Water/Component Cooling Water Systems
3.4 Emergency Core Cooling System
3.5 Containment Sprays
3.6 Containment Fans
3.7 Emergency Feedwater System
3.8 Emergency Seal Cooling
3.9 Other Fault Trees
3.10 Fault Tree Guidelines

3.10.1 Fault Tree Model Guidelines
3.10.2 Test and Maintenance Model
3.10.3 Human Error Model for Fault Trees
3.10.4 Common Cause Model (See Note 2)
3.10.5 Data Bank (See Note 3)

3.11 Screening Model for Operator Actions in Event Trees (See Note 4)

4.0 CORE MELT QUANTIFICATION

- 4.1 Quantification of Event Tree Nodes
- 4.2 Quantification of Core Melt
- 4.3 Analysis of Core Melt Contributors
- 4.4 Uncertainty Analysis in Core Melt

5.0 CORE AND CONTAINMENT ANALYSIS

- 5.1 Sequence Categorization
- 5.2 Core Melt and Containment Analysis
- 5.3 Containment Event Tree
- 5.4 Fission Product Source Term Analysis
- 5.5 Uncertainty Analysis

6.0 CONSEQUENCE ANALYSIS

- 6.1 Data Preparation
- 6.2 Consequence Analysis
- 6.3 Sensitivity Analysis
- 6.4 Uncertainty Analysis
- 6.5 Liquid Pathways Analysis

7.0 UNCERTAINTY ANALYSIS (See Note 5)

8.0 PLANT RISK ANALYSIS (See Note 6)

The second phase report will contain the following sections:

1. Seismic Events Analysis

2. Fire Events Analysis

3. Other External Events Analysis (See Note 7)

4. Sabotage Analysis (See Note 8)

5. Plant Risk From External Events

6. Plant Risk From Internal and External Events (See Note 9)

NOTES:

1. Dependencies on major support state systems such as AC-power, SWS, CCWS, etc. will be treated by the support state modeling approach. This approach is already used in the Millstone Unit 3 Probabilistic Safety Study.

2. Common cause will be included at least for active components, such as pumps, MOV's, etc. other system specific major common cause sources, such as strainers, human tasks, . . will also be considered whenever applicable. The common cause data will be taken from Atwood's work.

3. The most current generic data base maintained by Westinghouse will be used.

4. A screening model for operator actions will be used. It will be based on generic THERP modeling and Westinghouse emergency procedures.

5. Propagation of uncertainty in core melt, core and containment analysis, and consequence analysis will be done using the dominant accident sequences. For this purpose the results of Sections 4.4, 5.5 and 6.4 will be used.

6. Matrix approach in risk assembly process will be used for constructing the point estimate risk curves. This approach is the same as that of Millstone Unit 3 Probabilistic Safety Study.

7. Other external initiating events analysis may include relevant events such as wind, aircraft, etc.

8. Sabotage analysis will be done as a stand alone analysis and its results will be kept separate from the total plant risk analysis. It is envisioned that at most conditional probabilities for a set of sabotage scenarios will be quantified.

9. Uncertainties in external events will be addressed in each analysis section and for the total plant risk, based on uncertainty analysis on the dominant accident sequences.