ATTACHMENT FOR REQUEST 230.1

NOV 08 1977

Central Files - Topical Reports

General Electric Company
ATTN: Dr. G. G. Sherwood, Manager
Safety and Licensing
175 Curtner Avenue
San Jose, California 95114

Gentlemen:

SUBJECT: REVIEW OF GENERAL ELECTRIC TOPICAL REPORT NEDE-20944-P, "BWR/4 AND BWR/5 FUEL DESIGN" (NEDO-20944 NON-PROPRIETARY VERSION)

We have completed our review of Chapter 4, "Thermal and Hydraulic Design" of the General Electric Topical Report NEDE-20944-P. Our evaluation of the rest of the report (Chapters 2 and 3) was sent to you on September 30, 1977. We have concluded that Chapter 4 does not contain sufficient information to stand as a reference document. Further discussion is included in the enclosed evaluation.

This evaluation of Chapter 4 is not intended to serve as our final evaluation. We expect General Electric to utilize the evaluation to modify Chapter 4 so that a final evaluation accepting NEDE-20944-P can be written. Please contact us if you desire any discussion or clarification of the enclosed evaluation.

Sincerely,

Original Signed by
O. D. Parr

Olan D. Parr, Chief
Light Water Reactors Branch No. 3
Division of Project Management

Enclosure:
Topical Report Evaluation

cc w/enclosure:

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ENCLOSURE

Review of Chapter 4 of Topical Report NEDE-20944

Report Number: NEDE-20944
Report Title: BWR/4 and BWR/5 Fuel Design
Report Date: September, 1976
Originating Organization: General Electric Company
Reviewed By: Analysis Branch

The General Electric Company (GE) has submitted for review a licensing topical report, NEDE-20944, "BWR/4 and BWR/5 Fuel Design". The Reactor Analysis Section of the Analysis Branch has reviewed the thermal-hydraulics (Chapter 4) aspects of this report. Our evaluation of this report follows:

Summary of Topical Report (Chapter 4)

Chapter 4 of NEDE-20944 addresses selected subsections of section 4.4 as given in "Standard Format of Safety Analysis Report for Nuclear Plants", Revision 2. These subsections are:

- 4.4.1 Design Basis
    - 4.4.1.1 Safety Design Bases
    - 4.4.1.2 Power Generation Design Bases
    - 4.4.1.3 Requirements for Steady-State Conditions
    - 4.4.1.4 Requirements for Transient Conditions
    - 4.4.1.5 Summary of Design Bases
- 4.4.2 Description of Thermal-Hydraulic Design of the Reactor Core
    - 4.4.2.1 Summary Comparison
    - 4.4.2.2 Critical Power Ratio
    - 4.4.2.3 Linear Heat Generation Rate (LHGR)
    - 4.4.2.4 Void Fraction Distribution
    - 4.4.2.5 Core Coolant Flow Distribution and Orificing Pattern
    - 4.4.2.6 Core Pressure Drop and Hydraulic Loads
    - 4.4.2.7 Correlation and Physical Data
    - 4.4.2.8 Thermal Effects of Operational Transients

4.4.2.9 Uncertainties in Estimates

4.4.2.10 Flux Tilt Considerations

4.4.4 Evaluation

4.4.4.1 Critical Power

4.4.4.2 Core Hydraulics

4.4.4.3 Influence of Power Distributions

4.4.4.4 Core Thermal Response

4.4.4.5 Analytical Methods

4.4.4.6 Thermal-Hydraulic Stability Analysis

4.4.5 References

GE intends to include these subsections in FSAR's through reference to the topical report, i.e., these sections would be replaced with references to the report.

The text of Chapter 4 is almost identical to the text in section 4.4 of the Hatch-2 and Shoreham FSAR's. The principal difference between chapter 4 of the topical report and section 4.4 of these FSAR's is that plant specific information has been omitted from the topical report.

Summary of Staff Evaluation

Standardization of plant design is desirable and is to be encouraged; however, chapter 4 of this topical report represents only superficial standardization. The bulk of the text which normally appears in section 4.4 of FSAR's is included in the report with the intent of inclusion by reference in FSAR's, but most of the required information such as safety limits, pump characteristics and power-flow maps has been omitted because it's plant specific.

Specifically, the following information is missing from the report (subsection numbers refer to FSAR subsection numbers):

1.) Subsection 4.4.1 - Numerical values for MCPR and LHGR limits.

2.) Subsection 4.4.2 - Hydraulic loads, normally discussed in 4.4.2.6.

3.) Subsection 4.4.3 - Description of the thermal and hydraulic design of the reactor coolant system.

4.4.3.1 Plant Configuration Data

4.4.3.2 Operating Restrictions on Pumps

4.4.3.3 Power-Flow Operating Map

4.4.3.5 Following Characteristics

4.) Subsection 4.4.4.6.7 Analysis Results (Stability)

5.) Subsection 4.4.5 Testing and Verification

6.) Subsection 4.4.6 Instrumentation Requirements (Includes vibration and loose parts monitoring equipment).

7.) Table with plant configuration data.

8.) Figure with pump curve

9.) Figure showing decay ratio vs. power.

10.) Power-flow map

11.) Total system stability model diagram

12.) Figures (44 total) showing effects of step changes in pressure, rod position water level and load demand. These represent part of the stability analysis.

In addition, some of the information included in chapter 4 of the report is typically plant specific and cannot be referenced generically. This includes:

1. Table 4.4-1. "Thermal and Hydraulic Design Characteristics of the Reactor Core".

2. Table 4.4-2. "Void Distribution".