1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

This section provides a description of safety related technical information relevant to this application. Combustion Engineering, Inc., (C-E), is conducting research and development programs relating to the requirements of this section.

Waterford 3 reactor incorporates a 16 x 16 fuel assembly design with five guide tubes. This design provides an increase in conservatism for loss of-coolant accident (LOCA) considerations with a minimum change from previous C-E fuel designs. Previous designs have undergone extensive testing, and operating experience is now being acquired.

The three test programs described in Subsections 1.5.1, 1.5.2, and 1.5.3 are considered necessary to confirm the adequacy of the 16 x 16 fuel assembly design.

→(DRN 01-758, R11-A)

CENPD-84(1), CENPD-143(2), CENPD-184(3), CENPD-299(4) present descriptions of development programs aimed at verifying the Nuclear Steam Supply System (NSSS) design and the anticipated performance characteristics and at confirming the design margins. Other programs that apply to this plant are identified in Subsection 1.5.4 through 1.5.8.

←(DRN 01-758, R11-A)

1.5.1 FRETTING AND VIBRATIONS TESTS OF FUEL ASSEMBLIES

Extensive autoclave vibration and dynamic flow tests have been performed to characterize fuel rod and spacer grid fretting corrosion in C-E fuel assemblies. The results of these tests are discussed in more detail in Subsection 4.2.3.2.4.

Tests have been completed using a full sized 16 x 16 fuel assembly. This assembly is similar to the 16 x 16 five guide tube design used on the Waterford 3 reactor. This assembly was subjected to flow testing under conditions of temperature, water chemistry, pressure, and flow velocities in excess of normal reactor conditions. Further information is provided in Subsections 4.2.3.1.1, 4.2.3.1.2, and 4.2.4.4.

1.5.2 DEPARTURE FROM NUCLEATE BOILING (DNB) TESTING

→(DRN 01-758, R11-A; EC-13881, R304)

Extensive heat transfer testing has been completed with electrically heated rod bundles representative of the C-E 16 x 16 and 14 x 14 fuel assemblies. The program for each assembly geometry included tests to determine the effects on DNB of the control element assembly (CEA) guide tube, bundle heated length, and grid spacing, and lateral and axial power distributions. Each test yielded DNB data over a wide range of conditions of interest for pressurized water reactor (PWR) design. Those data were used with the TORC subchannel analysis code to develop and to verify the CE-1 DNB correlation for predicting DNB in fuel assemblies with standard spacer grids. The CE-1 correlation (for standard assemblies) and the ABB-NV correlation (for NGF assemblies), which are discussed in more detail in Subsection 4.4.4.1, are used in computing margin to DNB for Waterford 3.

←(DRN 01-758, R11-A; EC-13881, R304)

1.5.3 FUEL ASSEMBLY STRUCTURAL TESTS

The fuel assembly structural testing program was designed to verify the structural adequacy of the fuel assembly design under normal handling, normal operation, seismic excitation, and LOCA loadings. The test program provides the structural characteristics employed in the fuel assembly structural analyses.

A series of tests were conducted on a 14 x 14 fuel assembly to determine the combined axial and lateral load deflection characteristics of the fuel assembly. Axial compression tests and axial drop tests were performed. Measurements were made of axial loads, axial deflections, lateral deflections of all spacer grids, and strains in the guide tubes and fuel rods.

A series of structural tests on the 16 x 16 fuel assembly design was also conducted. The fuel assembly was subjected to both static and dynamic tests so as to determine basic structural characteristics. In addition, several 16 x 16 spacer grids were subjected to impact tests to determine dynamic load deflection characteristics and damage limits. These tests are also discussed in Subsection 4.2.3.1.3.

1.5.4 FUEL ASSEMBLY FLOW MIXING TESTS

The objective of the fuel assembly flow mixing program was to obtain information on the magnitude of coolant mixing in C-E fuel assemblies. Several series of tests have been completed, and the data from these tests provide a sound basis for the treatment of coolant mixing in design thermal margin calculations.

The first series of single phase flow mixing tests was run in 1966 with a prototype C-E PWR fuel assembly. The average level of coolant mixing was determined using dye injection and sampling equipment.

A second series of single phase mixing tests was conducted in 1968 with a model representing a portion of a 14 x 14 CEA-type fuel assembly. Those tests, which also used dye injection and sampling techniques, are described in Reference 1.

More recently, tests were conducted in which coolant temperatures were measured in the subchannels of electrically heated rod bundles representative of the 14 x 14 or 16 x 16 fuel assemblies with standard spacer grids.

As discussed in Subsection 4.4.4.1, those data provide confirmation that the results from the previous dye sampling experiments are applicable for the fuel assembly design used in Waterford 3.

1.5.5 REACTOR FLOW MODEL TESTING AND EVALUATION

The objective of the reactor flow model test programs is to obtain information on:

a) Flow and pressure distributions in various regions of the reactor

b) Pressure loss coefficients

c) Hydraulic loads on certain vessel internal components

This information is used for establishing or verifying design hydraulic parameters.

Flow model testing, which began in 1966, was designed to obtain those reactor hydraulic design data not amenable to direct calculation. Scale model testing possesses the advantages, relative to actual reactor tests, of:

- Providing the information early in the design stage
- Being more suitable for extensive instrumentation
- Being flexible so that proposed design modifications can be investigated

The reactor flow models used by C-E are generally 1/5 true scale models. In the first four C-E flow model programs, a closed-core design was used. The closed core simulates the reactor fuel assemblies with individual closed wall tubes containing orifices to provide the correct axial hydraulic resistance. Conclusions from the tests on the first four model configurations are summarized in Reference 1. A 1/5 scale flow model, representative of Waterford 3, was tested in 1976. This model has an open core design. Further discussion of the C-E flow model test programs is provided in Subsection 4.4.4.2.1.

1.5.6 FUEL ASSEMBLY FLOW TESTS

The objectives of the fuel assembly flow test program included assessment of the effect of postulated flow maldistributions on thermal behavior and margin.

The program originated in 1967 with fuel assembly flow distribution testing. Both flow visualization and flow pattern measurements were generated on an overscale model of the lower portion of an early C-E design fuel assembly.

A second test series was conducted for the CEA type fuel assembly. The second test series was designed to:

- Determine the effect of flow obstructions on flow distribution within the fuel assembly
- Determine the magnitude of the effect of the disturbed flow patterns on the thermal margin within a CEA type fuel assembly

The information from these tests, described further in Reference 1, has established the effect of flow obstructions within the fuel assembly. Additional information on the effects of postulated fuel coolant channel flow blockages is presented in Subsection 4.2.3.2.16.

1.5.7 CONTROL ELEMENT DRIVE MECHANISM (CEDM) TESTS

Performance testing of the magnetic jack CEDM is described in Subsections 3.9.4.4.1 and 4 2.4.4 and in Reference 1. The program has confirmed the operability of the drive assembly in normal and misaligned conditions as well as the load carrying capability and life characteristics.

1.5.8 DNB IMPROVEMENT

The DNB improvement program was initiated by C-E in order to obtain empirical information on the departure from nucleate boiling (DNB) phenomenon and on other thermal and hydraulic characteristics of C-E fuel assemblies. Testing has been performed with electrically heated rod bundles that correspond dimensionally to fuel rod configurations under in-reactor temperature pressure and flow conditions to obtain data on DNB, pressure drop, and coolant channel exit temperatures. These data were employed to verify that the C-E thermal hydraulic design methods conservatively predict DNB.

The DNB improvement program is described in References 1, 2, 3 and 4. It is a continuing program providing improvements in the accuracy of C-E thermal and hydraulic computer programs for predicting local coolant conditions and pressure drops and confirming the applicability of currently used DNB correlations to the C-E fuel design. Additional information on the program and results applicable to Waterford 3 are presented in Subsection 4.4.4.1.

SECTI0N 1.5: REFERENCES

1. "Safety Related Research and Development for Combustion Engineering Pressurized Water Reactors, Program Summaries," CENPD-87 and CENPD-87, Rev. 01 (Nonproprietary) (Proprietary) transmitted to DL by letter, Mr. F.M. Stern to Mr. R.C. DeYoung, Jr., March 18, 1973.

2. "Safety Related Research and Development for C-E Pressurized Water Reactors 1974 Program Summaries," Combustion Engineering Topical Report, CENPD-143 (Proprietary) and CENPD-143, Rev. 01, (Nonproprietary), May, 1974.

3. "Safety Related Research and Development for Combustion Engineering Pressurized Water Reactors - 1974 Program Summaries," CENPD-184-P (Proprietary) and CENPD-184 (Nonproprietary) May, 1975.

4. "Safety Related Research and Development for Combustion Engineering Pressurized Water Reactors, 1975 Program Summaries," CENPD-229-P (Proprietary) and CENPD-229 (Nonproprietary), June, 1976.