SAXTON NUCLEAR EXPERIMENTAL CORPORATION

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Amendment No. 2 to Change Reports 15 and 16

1. On November 13, 1968, Applicant submitted Change Reports 15 and 16 describing the installation of a water seal on the pressurizer safety valves and the installation of a spilled coolant recirculation system.

2. On July 8, 1969, Applicant submitted one amendment each to Change Reports 15 and 16 proposing pressure tests to demonstrate the integrity of the above installations. The proposed pressure tests involved pressurization of the reactor coolant system, and both amendments presented arguments to support the adequacy of the tests and conformance with the applicable Codes.

3. By letter dated October 7, 1969, Division of Reactor Licensing advised Applicant of apparent inconsistencies between the supporting arguments and the Technical Specifications, and requested clarifying information be provided.

4. Applicant hereby submits Amendment No. 2 to Change Reports 15 and 16 providing the clarifying information requested by Division of Reactor Licensing.

SAXTON NUCLEAR EXPERIMENTAL CORPORATION

By C. R. Montgomery
C. R. Montgomery, President

3784

December 2, 1969

On July 8, 1969, SNEC submitted one amendment each to Change Reports 15 and 16 proposing pressure tests to demonstrate the integrity of the pressurizer safety valve water seal installation and the spilled coolant recirculation system installation. The proposed pressure tests involved pressurization of the reactor coolant system and, therefore, were limited by the pressure capability of the reactor coolant system.

In determining the maximum pressure capability of the reactor coolant system the following conditions were considered:

1. Pressure retaining components in the reactor coolant system include austenitic stainless steel piping and fittings, and low alloy steels.

2. Paragraph N.3.a.(6) of the Technical Specifications specifies that main coolant pressure shall not exceed 500 psig until the temperature of the main coolant is at least 60°F above the nil ductility transition temperature of the reactor vessel. For the reactor vessel the nil ductility transition (NDT) temperature considerations are:

Initial NDT temperature	30°
NDT temperature shift due to the effects of radiation at the end of predicted Core III life	240°
Fracture transition for elastic loading (FTE) temperature	60°
	330°F

Thus, NDT temperature considerations fix the minimum reactor vessel temperature for the test at approximately 330°F.

3. The maximum pressure capability of the reactor coolant system is set by the centrifugally cast austenitic stainless steel piping in the system. The piping and fittings were designed, fabricated and erected in accordance with USAS B31.1 1955 Edition, and nuclear piping Cases N-9 and N-10 respectively. The material property data curves are the same for both components; therefore the allowable stresses as set forth in Cases N-9 and N-10 are identical. The allowable stress for austenitic stainless steel in both the ASME Boiler and Pressure Vessel Code and USAS B31.1 recognizes the work hardening characteristics of these materials by permitting the allowable stress at elevated temperature to reach 90% of the minimum 0.2% offset yield strength (S_a) at the specific design temperature. A footnote to Tables A-1 and A-2 of USAS B31.1 cautions that allowable stress is 90% of yield strength, and for some loading conditions undesirable plastic deformation could occur. Section III of the ASME Boiler and Pressure Vessel Code provides a more quantitative warning by limiting any test pressure to the lesser of 1.25x design pressure or that which produces a stress equal to 90% of the material yield at the test temperature. The austenitic stainless steels in