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PHILADELPHIA ELECTRIC COMPANY, PEACH BOTTOM ATOMIC POWER STATION - (ANNEX F - DESIGN AND FABRICATION OF REACTOR PRESSURE VESSEL AND STEAM GENERATOR

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In reviewing this report submitted by the applicant, the information presented agrees substantially with the suggested data outlined in the AEC Licensing Guide for the design and manufacture of system pressure vessels, both with respect to contents and details.

The documentation, as submitted, of vessel specifications, design criteria, stress analysis, quality control during manufacture and inspection and testing attests to the compliance with the requirements and rules of the ASME Code for Unfired Pressure Vessels Section III, in the case of the reactor vessel, and ASME Boiler and Pressure Vessel Code Section I for the steam generator.

Stress analysis of reactor vessel has been performed by Franklin Institute, which is recognized as qualified to responsibly analyze stresses under transient conditions in such vessels. Although not required by either Sections VIII or I of the code, the recent issuance of Section III rules (Nuclear Vessels) requires such analyses and certification by a registered professional engineer. The procedure, as followed by the vessel manufacturer, may therefore be interpreted as conformance with current acceptable practice. A similar procedure has been followed by Baldwin-Lima-Hamilton in analyzing the areas of critical stresses in the steam generator.

Safety valve pressure settings and rupture disc burst pressures for this reactor have been conservatively selected inasmuch as pressure ratings are less than that permitted by code rules. The application demonstrates that adequate consideration has been employed to account for the permissible tolerance on pressure settings as indicated in the code rules. Of noteworthy significance is the 34% margin provided, in this

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case, between operating pressure and maximum pressure attained when safety valves are blowing. Not only is this margin adequate but it also exceeds the generally allowed margin when compared with other applications.

Quality control procedures, as reported, fabrication surveillance inspection followed in the course of manufacture of the vessels, and the number of tests conducted (including strain gage tests of the reactor vessel) indicate that reasonable steps were taken to ensure compliance with the intent of vessel specifications.

On the basis of the review of the design criteria, the reported summary of stress analyses, the manufacturing effort and procedures followed to meet specifications, it is concluded that these pressure vessels may be expected to conform with the requirement of the vessel specifications and to provide reasonable assurance of their serviceability for the intended use.

Since the vessels are intended to provide service for a period of 30 years, the above conclusions are predicated upon the assumption that Division of Reactor Licensing will include in the Technical Specifications sections pertaining to the vessels the following requirements:

(a) Provisions or means to verify the pressure and temperature limitations (as assumed in the design calculations) at critical areas of the vessel.

(b) Safety valve and rupture disc pressure setting and venting capacities during blowdown, which provides overpressure protection of the vessel for the stated operating conditions defined by the vessel specification.

(c) Means employed to verify or monitor the limits of heating and cooling rates upon which the transient stress and fatigue analysis was based and the accounting of the number of significant transient cycles (as postulated in the analysis or otherwise if operational procedures change during service) to which the vessel may be subjected during its entire 30-year service life.

(d) Material surveillance program to monitor and predict the estimated shift in NDT temperature of the reactor vessel material, including the method which will be employed to measure and verify the integrated neutron flux estimates, from dosimetry measurements and the correlation and interpretation of Charpy V-notch impact tests of the

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surveillance specimens applied to determine the NDT temperature of vessel material during any time of its service life.

The irradiation damage surveillance program proposed should compare favorably with the recommended practice for "Surveillance Test on Structural Material in Nuclear Reactors", ASTM-E-185 Specification, which is currently being studied by the Division of Safety Standard's staff.

(e) The extent (planned or contemplated) of periodic examination or inspection of the interior of reactor vessel and steam generators, insofar as practical, considered after a significant interval of service to detect any unusual deterioration which may compromise the structural integrity of the vessels.

(f) The reporting requirements which apply to the reactor vessel and the steam generators during the service life period of 30 years.

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