ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

February 10, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON EDWIN I. HATCH NUCLEAR POWER PLANT UNIT 1

Dear Dr. Schlesinger:

At its 142nd meeting, February 3-5, 1972, the Advisory Committee on Reactor Safeguards completed a review of the condition of the reactor pressure vessel for the Edwin I. Hatch Nuclear Power Plant Unit 1. This matter was reviewed by a Subcommittee on February 2, 1972. During its review the Committee had the benefit of discussions with representatives and consultants of the Georgia Power Company (owner of the Hatch Plant), Southern Services, Inc. (the architect-engineer), General Electric Company (supplier of the nuclear system), and Combustion-Engineering, Inc. (manufacturer of the vessel), and the AEC Regulatory Staff. The Committee also had the benefit of the document listed.

The Hatch reactor pressure vessel was manufactured to the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components. It met all requirements of Section III, including radiographic examination of the nozzle welds that are the subject of this report, and was delivered to the site as a code-stamped Section III vessel.

In conformity with the ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Reactor Coolant Systems, the vessel was ultrasonically tested at the site to provide a baseline reference for inservice surveillance over the life of the plant. The ultrasonic tests showed indications of discontinuities around two of the ten approximately 12-inch inside-diameter inlet nozzles of the water recirculation system. The indications appeared to be near the interface between the nozzle-attachment weld and the vessel wall, at mid-wall thickness, and extending circumferentially around the nozzle for a distance of approximately 37 inches in one nozzle and 12 inches in the other. The orientation of the indications is approximately normal to the vessel wall (like a ribbon wrapped around the nozzle) but their character cannot be determined by existing nondestructive techniques and their widths can be expressed only as an upper limit, which was estimated to be 3/4 inch, over a limited distance, in one nozzle and less in the other. Two other independent ultrasonic examinations confirmed, in general, the length and orientation of the indications, but placed much lower upper limits on the width. An independent radiographic examination in the field failed to show indications, which is in agreement with the shop findings during fabrication.

The vessel thus meets the ASME fabrication code, Section III, but the field inspection by a method not required by Section III has revealed linear indications which, depending on their character, might have required repair if they had been found prior to certification.

The applicant has made fracture mechanics analyses assuming, as an extreme case, a full-circumference crack 3/4 inch wide. The analyses show that the calculated stresses in the region are low and that there should be no significant crack growth over the life of the plant. The analyses assume material properties at the lower limit of acceptability.

Notwithstanding the applicant's analyses, with which the Committee does not disagree, the Committee believes it is unacceptable to put this vessel into service with linear indications of incompletely defined character and dimensions. The ACRS therefore believes the vessel should be repaired, unless it can be shown by physical examination of samples obtained from the vessel that the discontinuities present and the relevant physical properties of the metal are within the limits set by Section III of the ASME Code. Any changes in the reactor vessel resulting from sampling should be evaluated analytically to establish the integrity and design life of the vessel will not be significantly impaired. The sampling program, acceptance criteria for discontinuities and metal properties, and analyses of effects of the sampling program on the vessel should be developed in conjunction with and be satisfactory to the Regulatory Staff. The Committee wishes to be kept informed.

Sincerely yours,

/s/ C. P. Siess

C. P. Siess
Chairman

Reference:

Georgia Power Company letter dated January 25, 1972, w/Summary of the Detection and Evaluation of Ultrasonic Indications for the Edwin I. Hatch Unit 1 Reactor Pressure Vessel