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

May 16, 1967

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON SM-1A PRESSURE VESSEL ANNEALING

Dear Dr. Seaborg:

At its eighty-fifth meeting, May 11-13, 1967, the Advisory Committee on Reactor Safeguards reviewed the U. S. Army's proposal to anneal in place the reactor pressure vessel of the SM-1A facility at Fort Greely, Alaska. The Committee had the benefit of discussion with representatives of the Army, NUS Corporation, the Naval Research Laboratory, and the AEC Regulatory Staff, and of the documents listed. An ACRS Subcommittee reviewed this program in a meeting held on April 19, 1967, in Bethesda, Maryland. The Committee previously reviewed a proposal to replace the SM-1A pressure vessel and commented on this in a letter to you dated October 15, 1965.

The annealing is to be accomplished by operating the reactor at 575°F and 1350 psi, compared with normal operating conditions of 430°F and 1200 psi, at a power level of approximately 1 to 2% of the normal 20 MWt. The increased temperature and pressure are both within the plant design limits. Primary and secondary pressure relief valves will satisfy Code requirements.

Army representatives stated that detailed procedures are being developed for the annealing operation, and that these would be reviewed by the Army Reactor Systems Health and Safety Review Committee. The annealing will be performed by the SM-1A operations staff, under the command of the Officer-in-Charge. In addition, the Nuclear Power Field Office (NPFO), Fort Belvoir, Virginia, will provide on-site technical support, on all shifts, with NPFO and NUS personnel.

Additional amplifiers will be installed in the reactor safety system to provide the increase in sensitivity necessary for establishing the trip level at 6% of normal power. Army representatives stated that the revised system would be thoroughly tested, including overload measurements at simulated ion-chamber currents no less than ten times the expected maximum current for postulated accident conditions.

Additional irradiated surveillance specimens of the pressure-vessel material have been obtained to supplement those originally provided. The Committee recommends that the surveillance program be carefully reviewed by the Army and the AEC Regulatory Staff, to ensure that the state of the vessel can be monitored, not only to establish the partial radiation-damage recovery by the annealing, but throughout the life of the vessel, so that future operation can be planned conservatively. The Committee would like to be kept informed of the data obtained from the surveillance program.

In view of the foregoing, the ACRS believes that the annealing operation can be carried out as proposed without undue risk to the health and safety of the public.

Sincerely yours,

/s/ N. J. Palladino

N. J. Palladino
Chairman

References:

1. ED-6604, "SM-1A Reactor Pressure Vessel Lifetime", U.S. Army Engineer Reactors Group, dated 1 September 1966.
2. Letter dated 4 January 1967 from Army Engineers Safety Office to AEC Director of Regulation with attachments: NUS-306, Supplement No. 1, and Errata Sheet.
3. Letter dated 15 March 1967 from Army Engineers Safety Office to AEC Director of Regulation with attachment: Supplement No. 2.
4. "Answers to DRL Questions", undated, received April 5, 1967.
5. "Answers to DRL Questions and Modifications to NUS-306 Concerning Reactor Vessel Head Stress Analysis", dated April 8, 1967.
6. Supplement No. 3, dated April 1967.
7. ARCHS 67-5, ARCHS Safety Evaluation Report dated April 1967.
8. "Answers to Questions, ACRS Subcommittee, Concerning Annealing of the SM-1A Reactor Vessel", dated May 3, 1967.