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

June 16, 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON NINE MILE POINT NUCLEAR STATION

Dear Dr. Seaborg:

At its 122nd meeting, June 11-13, 1970, the Advisory Committee on Reactor Safeguards reviewed the program proposed by the Niagara Mohawk Power Corporation for restoration to service of the Nine Mile Point Station following the discovery during March, 1970, of cracks and leakage in a "safe end" (stainless steel extension of the reactor vessel nozzle). The program was also considered at Subcommittee meetings on May 5, 1970, and June 1 and 2, 1970. During its review, the Committee had the benefit of discussions with representatives of the applicant, the General Electric Company, the AEC Regulatory Staff, and their consultants, and of the documents listed. The Committee previously reported to you on this project on April 17, 1969.

Normal procedures for most reactor pressure vessels have been to join the austenitic stainless steel safe ends to the nozzles prior to the stress relieving heat treatment. This heat treatment sensitizes the safe ends, which makes the steel less resistant to certain types of corrosion. Sensitized austenitic stainless steels in this condition have given reasonably satisfactory service over many reactor years of operation.

The applicant and the General Electric Company have conducted an extensive investigation of the cracking and its causes. An independent stress analysis of the as-built piping has revealed that stresses in the cracked safe end, and one other safe end, were excessive. It is believed that this excessive stress, possibly in combination with a high concentration of

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The applicant and the General Electric Company have conducted an extensive investigation of the cracking and its causes. An independent stress analysis of the as-built piping has revealed that stresses in the cracked safe end, and one other safe end, were excessive. It is believed that this excessive stress, possibly in combination with a high concentration of oxygen in the non-flowing fluid in the pipe concerned, caused the intergranular cracking of the furnace-sensitized stainless steel safe end. Both of the overstressed safe ends have been removed and replaced with new ones made of unsensitized material. The thermal sleeves have been slotted to avoid the possibility of gas bubbles at the high points. The piping supports have been rearranged, and the entire primary system re-analyzed, for both hot and cold conditions, to give assurance that stresses will remain within allowable limits.

One other safe end made of sensitized material has been removed and examined, found not to contain cracks, and has been replaced with a new one of unsensitized material. All other safe ends made of sensitized material have been non-destructively tested. The minor defects found will be ground out before the reactor is operated again.

The applicant stated that expansions of primary piping will be measured during a hot functional test to be conducted prior to restarting the reactor, to check the pipe supports and the seismic restraints.

The applicant has proposed an augmented surveillance program for the sensitized safe ends remaining in the primary system, including non-destructive testing at least once a year and re-checking piping expansions for several full thermal cycles. The Regulatory Staff should assure itself that the details of the proposed program are appropriate.

The applicant is studying improved leak-detection methods. The Committee believes that detection and location of small leaks is an essential part of the surveillance program. The applicant should expeditiously install such leak-detection devices as seem likely to give improved sensitivity or speed of leak detection. The Committee recommends that at least one leak-detection system in addition to the proposed sump accumulation rate and dew point systems be installed within a few months and wishes to be kept informed of progress in this regard.

The Committee believes that the Regulatory Staff should assure itself that the biological shield surrounding the reactor vessel can withstand the pressure that could be developed by loss of integrity of a safe end or nozzle, or that failure of the shield would have no intolerable consequences.

The ACRS believes that, if due regard is given to the recommendations above and in its previous report to you of April 17, 1969, there is reasonable assurance that the Nine Mile Point Nuclear Station can be operated at power levels up to 1538 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

Joseph M. Hendrie
Chairman

References

1. Niagara Mohawk Power Corporation report, "Reactor Primary System Investigation at Nine Mile Point Nuclear Station," dated May 1, 1970.

2. Niagara Mohawk Power Corporation report, "Reactor Primary System Investigation at Nine Mile Point Nuclear Station, Report No. 2," dated May 11, 1970.

3. Niagara Mohawk Power Corporation report, "Program for Restoration to Service Based on Reports of Primary System Investigation Nine Mile Point Nuclear Station," dated May 11, 1970.