UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 30 TO FACILITY OPERATING LICENSE NO. DPR-63

## NIAGARA MOHAWK POWER CORPORATION

## NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

## DOCKET NO. 50-220

### Introduction

By letter dated July 11, 1977, counsel for the Niagara Mohawk Power Corporation (NMPC) submitted the following: (1) an Application for Amendment to Operating License; (2) Proposed Changes to Technical Specifications (Appendix A to License No. DPR-63); and (3) Supporting Information.

The Technical Specification change was requested because NMPC had not been able to perform one of the tasks called for (see below) during the Spring, 1977, outage. The proposed change would formalize delaying the task until the next refueling outage. Since the plant had returned to operation, the choices for the NRC were either to approve the Technical Specification change or require an unscheduled shutdown for the purpose of performing the task. The task was to remove and inspect stainless steel stress corrosion surveillance specimens installed in the water phase. The need to perform the task was judged of insufficient importance to warrant an unscheduled shutdown.

### Discussion

The basic conclusion thereby was reached that the proposed Technical Specification change was acceptable to the NRC. Thus, NMPC will develop a method which will permit the water phase stress corrosion samples to be removed during the 1979 refueling outage. It was reported that the water phase samples were inaccessible during the 1977 refueling outage and were not removed. That constituted a violation of the Technical Specification which required that the samples, along with those in the steam and steam/ water phases be removed and examined at each outage.

We agree with the statement in the Supporting Information to the Proposed Technical Specification Changes which contends that the delay in the water phase sample examination will be unimportant relative to the program goals. It is safe to draw that conclusion for two reasons. First, in the six years of the stress-corrosion cracking surveillance program, none of the sensitized stainless steel samples have shown any signs of crack initiation. Second, several stainless steel components in the Nine Mile Point Nuclear Station, over the same period of time, have developed stress corrosion cracks. Thus, although the surveillance samples have contributed nothing

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to the prediction or control of stress-corrosion failure, operating experience has been used to develop a plan of inservice inspection and make necessary repairs or replacements. Time has shown that virtually nothing useful has been gained from the surveillance program while service experience has provided the information that the program originally set out to obtain.

The program was started in 1969 at the request of the NRC (or what then was the Division of Reactor Licensing, AEC). The material employed was stainless steel provided by the General Electric Company, judged representative of the type 304 used in the Nine Mile plant construction. Three material conditions were included: (1) annealed sheet (reference condition); (2) furnace sensitized sheet; (3) furnace sensitized forging. Nine specimens were taken from each source (a total of 27 specimens) and equally distributed in three specimen holders. The holders were loaded in the reactor pressure vessel in the steam, steam/water and water phases. Thus each of the three material conditions was exposed to the three different operating conditions with triplicate representation.

The specimens were rectangular bars, 3 in. long, 3/8 in. wide and 60-mils thick. They were assembled in holders designed to stress them individually in three-point bending.

During the 1970 refueling outage, nine of the original 27 specimens were removed and examined both non-destructively and destructively. The nine represented the three material conditions and exposure to the three operating conditions. No evidence of cracking could be detected. Subsequently, at each refueling outage except, as noted earlier, for the 1977 outage, the remaining specimens have been examined non-destructively without any evidence of cracking having been discovered. In the same period of time, stainless steel components in the core spray system, exposed to the same operating conditions, have showed through-wall cracking by the action of stress corrosion mechanisms.

Noting that the surveillance specimen fixtures impose a constant deflection (initially pre-set to achieve a pre-determined elastic outer fiber stress), the most likely reason for the continued crack-free behavior is that the initial stress has relaxed below the threshold needed for crack initiation. Moreover, the nature of stress relaxation under constant deflection is a roughly exponential decay. Therefore, if there is merit in the relaxation hypothesis, then there has been insufficient stress to make a fair test of the program for the past several years. Under such conditions, continuation of the Nine Mile Point stress corrosion surveillance program would be fruitless.

The program was requested by the AEC in reaction to the observation of extensive cracking in the core spray lines at the Oyster Creek (New Jersey) nuclear power station. The Oyster Creek plant core spray lines included type 304 stainless steel which cracked because it was sensitized.

NMPC was asked to institute a stress corrosion cracking surveillance program because the Nine Mile plant was the sister to Oyster Creek. Cracking of the core spray line stainless at Nine Mile subsequently has provided the perspective. Nine Mile's stainless was not sensitized as much as that at Oyster Creek but enough to have cracked in less than ten years of operation.

The NMPC stress corrosion surveillance program shall be brought to a conclusion in the following manner. All samples shall be removed from the reactor during the 1979 refueling outage. NMPC should engage an independent, technically competent, organization to examine the samples destructively and/or non-destructively, as judged necessary by the independent organization, to determine if any stress corrosion cracking has occurred. Those individuals who have examined the samples in the past (at Nine Mile and at GE) should participate in the 1979 task to provide continuity. A final report shall be formally submitted which provides the findings of this examination.

## Environmental Considerations

We have determined that this amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that this amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

## Conclusion

We have concluded that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 26, 1979