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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 126 TO FACILITY OPERATING LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION UNIT NO. 1

DOCKET NO. 50-220

1.0 INTRODUCTION

By letter dated July 27, 1988, as supplemented May 21, 1991, Niagara Mohawk Power Corporation (the licensee) requested an amendment to the Facility Operating License No. DPR-63 for Nine Mile Point Nuclear Station Unit No. 1 (NMP-1). The proposed amendment would extend the expiration date of the facility operating license from April 11, 2005, to August 22, 2009.

2.0 DISCUSSION

Title 10 CFR 50.51 specifies that each license will be issued for a fixed period of time not to exceed 40 years from the date of issuance. The currently licensed term for NMP-1 is 40 years, commencing with the issuance of the construction permit on April 11, 1965. Therefore, the current expiration date is April 11, 2005. However, due to construction time, the effective facility operating license term is about 35.7 years. Consistent with Section 50.51 of the Commission's regulations, the licensee has requested an extension of the facility operating license so that the facility operating license would end 40 years after the date of issuance (August 22, 1969) of Provisional Operating License No. DPR-17. The Provisional Operating License was superseded in its entirety by full-term Facility Operating License No. DPR-63 which was issued on December 26, 1974.

The licensee's request for extension of the facility operating license is based on the fact that a 40-year service life was considered during the design and construction of the plant. Although this does not mean that some components will not wear out during the plant lifetime, design features were incorporated which maximize the inspectability of structures, systems, and equipment. Surveillance and maintenance practices which are implemented in accordance with the ASME Code and the facility Technical Specifications provide assurance that any unexpected degradation in plant equipment will be identified and corrected.

3.0 EVALUATION

The NRC staff has evaluated the safety issues associated with issuance of the proposed license amendment which would allow 4.3 additional years of plant

operation. The issues addressed consist of additional radiation exposure to the licensee's operating staff, impacts on the offsite population, nonradiological impacts, and the general aging of plant structures and equipment. The impact of additional radiation exposure to the operating staff, the impact on the offsite general population, and the nonradiological impacts are addressed in the NRC staff Environmental Assessment dated August 7, 1991.

3.1 Mechanical Equipment

The components of the reactor coolant pressure boundary were designed, built and tested in accordance with the ASME Boiler and Pressure Vessel Code Section I Power Boilers, 1962 Edition and the ASA B31.1-1955 Piping Code, plus the Nuclear Code Cases in effect in December 1963 (when the vessel was purchased). Furthermore, the vessel manufacturer (Combustion Engineering) was directed by the purchase specification of the buyer (General Electric) to use Section VIII of the Code for Unfired Pressure Vessels where Section I Power Boilers did not cover specific details.

The initial inservice inspection program was described in Technical Specification 4.2.6. The program was written in accordance with Regulatory Guide 1.51 (1973) which endorsed the Winter 1972 Edition of ASME Code, Section XI. By letter dated May 2, 1980, as revised July 30, 1982, the licensee submitted a proposed inservice inspection program description and request for relief. The program was approved by the NRC staff on September 19, 1983. The First Ten-Year interval ended in June 1986. The Second Ten-Year Interval Inservice Inspection Program Plan was initially submitted on December 16, 1985. However, on April 9, 1991, the licensee withdrew the December 16, 1985, submittal and committed to submit a revised comprehensive Inservice Program Plan for the second ten-year interval by March 31, 1992.

In 1982, intergranular stress corrosion cracking (IGSCC) was first identified in the large diameter recirculation piping at NMP-1. In a refueling outage which extended from March 1982 to June 1983, the licensee replaced the NMP-1 recirculation piping with low carbon type 316 stainless steel piping which is resistant to IGSCC. Inspections of similar piping at other boiling water reactors (BWRs) disclosed additional instances of IGSCC in large diameter stainless steel pipes. Since the NRC staff considered this a generic problem, Generic Letter 84-11 was issued to require a reinspection program at all BWRs. This program involved welds in stainless steel pipes greater than 4 inches in diameter, in systems that are part of or connected to the reactor coolant pressure boundary. If IGSCC was discovered, repair, analysis, and additional surveillance were required to ensure the continued integrity of the affected pipe.

Generic Letter 88-01, issued on January 25, 1988, superseded Generic Letter 84-11, and included a copy of NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping." NUREG-0313, Revision 2, described methods acceptable to the staff to control the susceptibility of BWR ASME Boiler and Pressure Vessel Code Class

1, 2, and 3 pressure boundary piping and safe ends to intergranular stress corrosion cracking. For piping that does not conform to the staff positions, varying degrees of inservice inspection are required to ensure structural integrity of the pressure boundary piping system, pursuant to 10 CFR 50. 55a(g)(6)(ii). The licensee responded to GL 88-01 by letter dated July 28, 1988, as supplemented on August 25, 1989, September 6, 1989, and November 3, 1989. The NRC staff has reviewed these responses and forwarded its safety evaluation to the licensee by letter dated May 15, 1990. The NRC staff concluded that the licensee's proposed IGSCC inspection and mitigation program will provide reasonable assurance of maintaining the long term structural integrity of austenitic stainless steel piping at NMP-1. License Amendment No. 107, which incorporates the requirements of GL 88-01, was issued on July 7, 1989.

Torus wall thinning, as a result of corrosion, was first identified in 1975 by the licensee. By letter dated January 12, 1989, the licensee committed to measure the wall thickness of the torus material every six months for trending purposes and comparison to the minimum allowable values required by the ASME Code. The NRC staff has reviewed the measurement data and concluded that the licensee may operate NMP-1 in its current condition for the remainder of the current fuel cycle provided surveillance of the torus continues at six-month intervals. By letter dated November 22, 1989, the licensee committed to strengthen the torus during the next (1992) refueling outage. However, by letter dated May 14, 1991, the licensee submitted a plant specific analysis of the torus dynamic loadings. The purpose of this plant specific analysis is to demonstrate that the torus will remain acceptable for use without modifications until at least 2007. The NRC staff is in the process of reviewing the May 14, 1991, submittal and will prepare a safety evaluation upon completion of that review. This review is expected to be completed by December 1991.

The NRC requires licensees to develop and implement an inservice testing (IST) program for demonstrating the continued operability of NMP-1 pumps and valves in response to the requirements of 10 CFR 50.55a. By letter dated March 28, 1989, the licensee submitted its second ten-year interval IST program plan for demonstrating the continued operability of the NMP-1 pumps and valves. By letter and enclosed safety evaluation dated March 7, 1991, the NRC staff approved the licensee's IST program.

From our evaluation, we conclude that compliance with the codes, standards, and regulatory requirements to which the mechanical equipment for NMP-1 was originally analyzed, constructed, tested and inspected, including the inservice inspection programs in compliance with Section XI of the ASME Boiler and Pressure Vessel Code and the other augmented inspections of austenitic stainless steel piping, provide adequate assurance that the structural integrity of components important to safety will be maintained during the additional periods authorized by this amendment. Additionally, the testing program for pumps and valves will enable early detection of degradation which could affect their operability. Any significant degradation by an active mechanism would be discovered and the mechanical equipment or component restored to an acceptable condition. Therefore, the age of the mechanical equipment or components should not be a concern in the proposed extension of the facility operating license.

3.2 Electrical Equipment - and Environmental Qualification

Aging analyses and environmental qualifications have been performed for all electrical equipment important to safety in accordance with the environmental qualification requirements of 10 CFR 50.49. This program included identification of qualified lifetimes for the required equipment and incorporation of maintenance requirements into appropriate plant procedures to maintain the qualification of the required equipment for the life of the plant. This qualification program provides assurance that electrical equipment important to safety will function as required if called upon to mitigate design basis events, regardless of the term of the facility operating license.

3.3 Structures

The design and construction of structures and supports was in accordance with various codes and standards applicable at the time of plant construction. The design bases, fabrication, construction, and quality assurance criteria for the plant were reviewed and found acceptable by the NRC staff. Industrial experience with such structures and supports confirms that a service life in excess of 40 years can be anticipated.

The Updated Final Safety Analysis Report (UFSAR) states that the reactor vessel was fabricated, inspected, and tested in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section I, Power Boilers, 1962 Edition and Addenda plus the Nuclear Code Cases applicable on December 11, 1963.

The integrity and performance capability of the ferritic materials in the reactor vessel is assured because the fracture toughness is monitored with a surveillance program in conformance, to the extent practical, with the requirements of Appendix H, 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," and ASTM 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." The vessel material and surveillance sample withdrawal schedule is specified in Technical Specification 4.2.2b. The ferritic materials must meet the fracture toughness properties of Section III of the ASME Boiler and Pressure Vessel Code and Appendix G, 10 CFR Part 50, "Fracture Toughness Requirements."

A comprehensive vessel material surveillance test program is maintained in accordance with 10 CFR Part 50, Appendix H. Samples of vessel base materials, weld materials, heat affected zone materials, and standard materials are located within the core region to permit periodic monitoring of exposure and material properties relative to control samples.

The NRC staff concludes that there are no special considerations regarding degradation of structures due to the proposed operating lifetime extension. The structural integrity of the reactor vessel is assured because it was originally designed assuming a 40-year lifetime; it is monitored, inspected, and tested to detect degradation processes at an early stage of development; and it is operated in accordance with procedures to assure that its design conditions are not exceeded.

4.0 SUMMARY

The NRC staff concluded in the Environmental Assessment that the annual radiological effects during the additional years of operation that would be authorized by the proposed license amendment are not more than were previously estimated in the Final Environmental Statement, and are acceptable. The NRC staff concludes from its considerations of the design, operation, testing, and monitoring of the mechanical equipment, electrical equipment, structures, and the reactor vessel that an extension of the facility operating license for NMP-1 to a 40-year service life is consistent with the UFSAR, SER, and submittals made by the licensee, and that there is reasonable assurance that the plant will be able to continue to operate safely for the additional period authorized by this amendment. We also conclude that the plant is operated in compliance with the Commission's regulations, and that issues associated with plant degradation have been adequately addressed.

Based on the above, the NRC staff concludes that extension of the facility operating license for NMP-1 to allow a 40-year service life from issuance of the initial operating license is consistent with the Final Environmental Statement and Safety Evaluation Report for the plant and that the Commission's previous findings are not changed.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on August 15, 1991 (56 FR 40645). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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