probability or consequences of an accident previously evaluated.

With the sleeve dimensions, materials, and connecting joints designed to the applicable American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), the proposed sleeving repair becomes an in-kind substitution for the steam generator tube being repaired. The design criteria for the sleeves conform to the stress limits and safety margins of Code Section III. Safety factors of 3 (normal operation) and 1.5 (accident conditions) were applied to the sleeve design. Mechanical testing using Code stress allowables also has been performed in support of the sleeve design. Based on the results of vendor test and analysis programs, the sleeves fulfill their intended function as leak tight structural members and meet or exceed all design criteria.

Evaluation of the steam generator tubes and proposed sleeves indicates no detrimental effects on the sleeve or sleeve-tube assembly from reactor coolant system flow, reactor or steam generator coolant chemistry, or thermal or pressure conditions (including transients) that may be experienced by the Maine Yankee plant. Corrosion testing of sleeve-tube assemblies indicates no evidence of sleeve or steam generator tube corrosion considered detrimental under anticipated service

conditions.

Installation of the proposed sleeves will be controlled via Combustion Engineering's proprietary equipment and process. The process has been used 24 separate times since 1984 to install approximately 4100 steam generator sleeves in nuclear facilities worldwide. The Maine Yankee steam generator design has been reviewed and found compatible with the sleeve installation equipment and process. Installation of the proposed sleeves will have no significant effect on either plant configuration or operation.

The licensee therefore concludes that implementation of the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

As discussed above, the structural integrity, thermal characteristics, and material properties of the proposed sleeves are compatible with Maine Yankee's steam generators. Therefore, the functions of the steam generators will not be significantly affected by installation of the proposed sleeves. In addition, the proposed sleeves do not

interact with any other plant systems. Finally, the continued integrity of installed sleeves is periodically verified by the steam generator inspections required by plant Technical Specifications.

The licensee therefore concludes that implementation of the proposed change will not create a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment would not involve a significant reduction in a

margin of safety.

Repair of degraded steam generator tubes via the use of the proposed sleeves has been confirmed to restore the structural integrity of faulted tubes under normal operating and postulated accident conditions. The design safety factors used for the sleeves are consistent with ASME Code safety factors required in the design of Maine Yankee's steam generators. The repair limit for the proposed sleeves is consistent with that established for Maine Yankee's steam generators. The design of the sleeve-to-tube joint has been verified by testing to preclude significant leakage during normal and postulated accident conditions. Use of the previously identified design safety factors design verification testing assures that margin to safety with respect to installation of the proposed sleeves is not significantly different from the original steam generator tubes.

The licensee therefore concludes that implementation of the proposed change would not involve a significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, Maine 04578.

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NRC Project Director: Walter R. Butler.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of amendment request: December 23, 1994.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) 2.1.2, "Fuel Cladding Integrity," 3.6.2/4.6.2, "Protective Instrumentation," and associated Bases to extend the calibration frequency of the reactor recirculation flow transmitters from once per quarter to once per operating cycle and for the square rooters and summers from once per quarter to once per year. The proposed amendment would revise the flow biased average power range monitor (APRM) scram and rod block, recirculation flow comparator, and flow unit upscale setpoints and the associated Bases of TSs 2.1.2, 2.2.2, and 3.6.2/4.6.2.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes extend the calibration interval for the recirculation flow square rooters, summers and transmitters and revise the setpoints for the recirculation flow upscale and comparator rod block trips. The associated analytical limits for APRM flow biased scram and rod block increase by 2% and 8% respectively. Setpoints are for plant protective functions (i.e., scram and rod block) which respond to an accident or transient. The scram and rod block function responds to mitigate the consequences of an accident or transient. Therefore, a change to the setpoints cannot increase the probability of these accidents or transients. Likewise. changes to surveillance intervals for the protective functions which respond to an accident or transient cannot increase the probability. In fact, the proposed increase in the surveillance intervals reduce the probability of an inadvertent scram by reducing the duration that the plant is in the one-half scram condition.

The new surveillance intervals, setpoints and allowable setpoint deviations are calculated using the approved GE [General Electric Company] setpoint methodology documented in NEDC-31336. The methodology in NEDC-31336 provides assurance that safety system actuation (i.e., reactor scram or control rod withdrawal block) will occur prior to the associated system parameters (neutron flux and recirculation flow) exceeding their analytical limits. Based upon re-evaluation of NMP1 [Nine Mile Point Nuclear Station Unit No. 1] accidents and transients, it has been shown that the fuel thermal limits are not significantly impacted. Therefore, the consequences of an accident or transient has not significantly increased.

Thus, plant response to previously analyzed accidents remains within previously determined limits. Therefore, the operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.