ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
ON REACTOR VESSEL PRESSURE-TEMPERATURE LIMITS AND SURVEILLANCE PROGRAM

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT UNIT 1
DOCKET NO. 50-220

INTRODUCTION

By letter dated June 16, 1988, Niagara Mohawk Power Corporation (the licensee) provided an evaluation of the impact of a potential material mix-up in the reactor vessel surveillance capsule test material on the current pressure-temperature limits for Nine Mile Point Unit 1 (NMP-1). The current pressure-temperature limits were approved by the staff for 13 effective full power years (EFPY) in Amendment No. 95 dated March 15, 1988. The bases for the current pressure-temperature limits were the test results from the NMP-1 surveillance program provided in a licensee's letter dated August 15, 1985. The potential material mix-up in the NMP-1 reactor vessel surveillance program raises uncertainties in the bases of the current pressure-temperature limits. In the June 16, 1988 letter the licensee contends that the current pressure-temperature limits remain valid because they are conservative even assuming a material mix-up.

Fracture toughness properties of ferritic steel in reactor vessels decrease as a result of neutron irradiation. The reactor is required to be operated at a pressure and a temperature such that there is sufficient fracture toughness present in the reactor vessel material. Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G to 10 CFR Part 50. Changes in the fracture toughness of the reactor vessel materials are required to be monitored in a surveillance program whereby material specimens are exposed in surveillance capsules inserted in the reactor vessel and withdrawn periodically for fracture toughness testing. The reactor vessel material surveillance program must comply with Appendix H to 10 CFR Part 50.

Ferritic steel has a nil-ductility transition temperature below which the fracture behavior of this material changes from ductile to brittle. Neutron irradiation increases the nil-ductility transition temperature. This ductile to brittle behavior is described by the reference temperature RT_{NDT} in Paragraph NB-2331 of Section III of the ASME Code. The pressure-temperature limits depend on the initial RT_{NDT} for the limiting materials in the beltline and the increase in RT_{NDT} resulting from neutron irradiation. The methodology in Regulatory Guide 1.99, Revision 2 is used by the staff in evaluating the effect of radiation embrittlement of reactor vessels.

PRESSURE-TEMPERATURE LIMITS

The staff has approved pressure-temperature limits for NMP-1 for 13 EFPY based on NMP-1 surveillance capsule data and Regulatory Guide 1.99, Revision 2

evaluation procedures. Subsequent to the staff's review and approval the licensee indicated that because the surveillance data indicated that the increase in RT_{NDT} was almost three times the predicted value based on Regulatory Guide 1.99, Revision 2, the possibility of a material mix-up in the surveillance capsule was being investigated. The potential material mix-up in the NMP-1 reactor vessel surveillance program raised uncertainties in the bases of the current pressure-temperature limits.

The staff evaluated the conservatism of the current NMP-1 pressure-temperature limits considering the potential material mix-up. The limiting material in the NMP-1 beltline is the plate material. The staff made three conservative assumptions as follows:

(1) The NMP-1 surveillance data are not credible because of the potential material mix-up. Thus, Regulatory Position 1.1 in Regulatory Guide 1.99, Revision 2 was used in estimating the increase in RT_{NDT}.

(2) There is no information available on the vessel material chemistry. Thus, the default values of 0.35% copper and 1.0% nickel were used according to Regulatory Guide 1.99, Revision 2 in estimating the increase in RT_{NDT}.

(3) The initial RT_{NDT} was based on the largest measured value of all NMP-1 beltline plates in the beltline as provided in a licensee's letter dated August 15, 1985. The largest unirradiated "Charpy" temperature at the 50 ft-lb energy level is 52°F for Plate G-8-1. Thus, the initial RT_{NDT} according to NB-2331(a)(3) of Section III of the ASME Code is 52°F minus 60°F, or -8°F.

The following table summarizes the increase in RT_{NDT}, margin as discussed in Regulatory Guide 1.99, Revision 2, and initial RT_{NDT} for 10 and 20 EFPY based on the licensee's calculations using NMP-1 surveillance data and the staff's calculations using the above assumptions. The adjusted RT_{NDT} is the summation of the increase in RT_{NDT}, margin, and initial RT_{NDT}.

	10 EFPY		20 EFPY	
	Licensee	Staff	Licensee	Staff
Increase in RT_{NDT}	130	89	183	125
Margin	0	34	0	34
Initial RT_{NDT}	-4	-8	-4	-8
Adjusted RT_{NDT}	126	115	179	151

Note: All values are temperatures in degrees F. The estimated fluences at 10 and 20 EFPY are 0.616×10^{18} and 1.232×10^{18} n/cm^2, respectively.

The licensee used the NMP-1 surveillance data following the methodology of Regulatory Guide 1.99, Revision 2, except that a margin of zero was assumed. Also, the licensee used the unirradiated "Charpy" temperature at the 30 ft-lb energy level of an intermediate shell plate as the initial RT_{NDT}. The

licensee's estimate of initial RT_{NDT} is not consistent with Appendix G to 10 CFR Part 50.

However, the table shows that the adjusted RT_{NDT} values used by the licensee in developing the current pressure-temperature limits are larger than that calculated by the staff using conservative assumptions. Since a larger adjusted RT_{NDT} results in more conservative pressure-temperature limits, the licensee's current pressure-temperature limits are acceptable.

APPENDIX H COMPLIANCE

Appendix H to 10 CFR Part 50 requires that the reactor vessel surveillance program conducted prior to the first capsule withdrawal meet the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. The NMP-1 reactor vessel surveillance program was based on ASTM E 185-66 according to NMP-1 Technical Specifications (Bases for 3.2.2 and 4.2.2). ASTM E 185-66 recommends that surveillance samples be obtained from the heat of the base metal having the highest initial nil-ductility transition temperature. In the licensee's letter of January 31, 1978, the licensee indicated that based on existing staff guidance all of the beltline plate materials had the same initial nil-ductility transition temperature. Therefore the licensee could select any beltline material as surveillance material to comply with ASTM E 185-66. However, because of the potential material mix-up identified in the licensee's letter of June 16, 1988, the material in the surveillance program needs to be identified.

REACTOR VESSEL SURVEILLANCE PROGRAM

The licensee is currently evaluating the potential material mix-up in the reactor vessel surveillance program. The licensee should determine the chemistry and initial RT_{NDT} of all the plates in the beltline region and determine which is the limiting material based on current guidance. The licensee should provide justification that the test material in the surveillance capsules has been positively identified. Then, the licensee should provide a plan for continuing the surveillance program and revising the pressure-temperature limits in accordance with Regulatory Guide 1.99, Revision 2. The revised pressure-temperature limits should account for the most limiting material in the reactor vessel beltline.

CONCLUSIONS

The staff has evaluated the impact of the potential material mix-up in the reactor vessel surveillance program for NMP-1 and concurs with the licensee's contention that the current pressure-temperature limits are conservative and thus acceptable. However, the licensee should determine the identity of the test material in the surveillance program and submit revised pressure-temperature limits for staff review and approval before the current pressure-temperature limits expire at the end of 13 EFPY.

September 14, 1988

Docket 50-220

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Mr. Charles V. Mangan
Senior Vice President
Niagara Mohawk Power Corporation
301 Plainfield Road
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Dear Mr. Mangan:

SUBJECT: SAFETY EVALUATION OF NINE MILE POINT UNIT 1 REACTOR VESSEL PRESSURE-TEMPERATURE LIMITS AND SURVEILLANCE PROGRAM (TAC 68701)

By letter dated June 16, 1988, Niagara Mohawk provided an evaluation of the effect of a potential mix-up of the reactor vessel surveillance capsule test material on the pressure-temperature limits for Nine Mile Point, Unit 1. The staff has reviewed that submittal and agrees that the current pressure-temperature limits are conservative and are therefore acceptable. However, Niagara Mohawk should determine the identity of the test material in the surveillance program and submit revised pressure-temperature limits for staff review and approval in a timely manner before the current pressure-temperature limits expire at the end of 13 effective full power years. The details of the staff's review are discussed in the enclosed safety evaluation.

If you have any questions concerning the enclosed safety evaluation, please contact the Project Manager, Mary Haughey (301) 492-1439.

Sincerely,

original signed by

Mary F. Haughey, Project Manager
Project Directorate I-1
Division of Reactor Projects I/II

Enclosure:
Safety Evaluation

cc: See next page

PDI-1:LA	PDI-1:PM	PDI-1:D
CVogan	MHaughey:vr	RCapra
9/13/88	9/13/88	9/13/88

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