UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO DENIAL OF AMENDMENT 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 September 1, 1995, the Niagara Mohawk Power Corporation, the licensee, requested permission to revise the pressure/temperature (P/T) limits in the Nine Mile Point Unit 1 (NMP-1) Technical Specifications (TSs), Section 3.2.2. The licensee proposed to replace existing Figures 3.2.2.a,b,c,d and e and associated Tables 3.2.2.a,b,c,d and e. These figures and tables define the limits for minimum reactor pressure vessel (RPV) temperature for pressurization and account for neutron damage at exposures up to 18 effective full power years (EFPY). They will be replaced with new figures and tables that are also applicable for up to 18 EFPY. Editorial changes have been proposed to Specification 3.2.2.c to clarify that Figure 3.2.2.e also applies to in-service leak tests. The Commission had previously issued a proposed finding that the amendment did not involve a significant hazards consideration and there was no public comment on such findings (59 FR 51620).

On November 21, 1994, the NRC staff sent a Request for Additional Information (RAI) to the licensee. The RAI was clarified in a telephone conference on December 1, 1994. On December 5, 1994, the licensee submitted a proprietary copy of the "Plant Specific Charpy Shift Model for Nine Mile Point Unit 1." The remainder of the requested information was sent on December 20, 1994. The December 20, submittal included the "Response to NRC Additional Information Request Related to Proposed Changes to Nine Mile Point Unit 1 (NMP-1) Pressure-Temperature Limits;" a non-proprietary version of the "Plant Specific Charpy Shift Model for Nine Mile Point Unit 1;" and a waiver of Copyright Restrictions for the non-proprietary report.

2.0 BACKGROUND

The new P/T limits were calculated using a proposed Charpy shift model that shows the mean value of the change in reference temperature caused by irradiation (ΔRT_{NDT}) as a function of the square root of fluence, with no chemistry dependence. The licensee established criteria for defining the data set from the NRC Power Reactor Embrittlement Data Base (PR-EDB). Application of the model to the limiting NMP-1 beltline plate decreases the P/T heatup

Enclosure

curves by 26 °F, and the cooldown and hydrotest P/T curves by 41 °F. As a result, in-service leak testing could be conducted below 212 °F, and the duration of testing could be reduced by approximately 8 hours. Flexibility in outage scheduling would also increase.

The staff evaluates the P/T limits based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Generic Letters (GLs) 88-11 and 92-01; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P/T limits for the reactor vessel must be at least as conservative as those obtained using Appendix G to Section III of the American Society of Mechanical Engineers (ASME) Code. GL 88-11 requires that licensees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation by calculating the adjusted reference temperature (ART) of reactor vessel materials. The ART is defined as the sum of the initial nil-ductility transition reference temperature (RT_{NDT}) of the material, the increase in RT_{NDT} caused by neutron irradiation (ΔRT_{NDT}), and a margin to account for uncertainties in the prediction method. ΔRT_{NDT} is calculated from the product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the vessel materials. GL 92-01 requires that licensees submit reactor vessel materials data, which the staff will use in the review of the P/T limits submittals.

SRP 5.3.2 provides guidance on calculation of the P/T limits using linear elastic fracture mechanics methodology specified in Appendix G to Section III of the ASME Code. The linear elastic fracture mechanics methodology postulates sharp surface defects that are normal to the direction of maximum stress and have a depth of one-fourth of the reactor vessel beltline thickness (1/4T) and a length of 1-1/2 the beltine thickness. The critical locations in the vessel for this methodology are the 1/4T and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

3.0 EVALUATION

The licensee used its proposed Charpy shift model to calculate ART values for the limiting material, upper shell plate G-307-4, of 101.6 °F and 85.8 °F at 1/4T and 3/4T, respectively for 18 EFPY. The licensee's values for ART at 1/4T and 3/4T are non-conservative when compared to the methodology in the RG which recommends the use of credible plant specific surveillance data to determine the effect of neutron irradiation embrittlement on the vessel. The licensee chose not to follow the RG procedure, and instead used generic data from BWR reactor vessel surveillance programs. The proposed Charpy shift model was developed for BWR radiation environments and consists of 40 data points from 22 reactor vessel surveillance programs, including three NMP-1 data points. The maximum fluence included in the model was 2E18 n/cm^2, with one exception (Point Beach 1, with a fluence of 5.24E18, was included). The proposed Charpy shift model is generic, not vessel specific, and has a large

standard deviation of 32 °F. It is dependent on fluence and independent of copper and nickel. The staff denies the proposed changes to the TSs for the following reasons:

A) The licensee proposes to use the RG 1.99, Rev. 2 standard deviation as opposed to the larger deviation of the proposed model.

The standard deviation (σ_Δ) for the model is 32 °F. The licensee proposed to use the RG value; however, this method would be non-conservative since the standard deviation for the proposed model is much larger than the RG 1.99, Rev. 2 value of 17 °F. In addition, the use of a smaller standard deviation than the value derived from the analysis of the data would not account for the variability in the model. To account for this variability, the model should use the standard deviation from the analysis of the data.

B) The data used in the analysis do not support the licensee's conclusions.

The staff generated graphs of ΔRT_{NDT} vs. fluence as shown in attached Figures 1, 2, and 3. The graphs represent all of the data used in the licensee's regression analysis. Figure 1 represents low (0.10%-0.15%) copper values, Figure 2 intermediate (0.15%-0.20%) values, and Figure 3 high (0.20% and above) values. With a few exceptions, the data do not support the licensee's contention that at fast neutron fluences as high as 2E18 n/cm^2, there is no significant copper functional dependence in the Charpy shift model. In contrast, the data show that for high ranges of percent copper, higher copper contents result in higher values of ΔRT_{NDT}.

C) The licensee's contention that copper only plays a minor role in neutron radiation embrittlement conflicts with the NRC staff's position.

The licensee also reports that Atom Probe-Field Ion Microscopy (APFIM) studies have shown definitively that the concept of pure copper precipitates in RPV steels is incorrect. These studies are also reported to have shown that clusters of elements like Ni, Mn, Si and Cu actually form, but copper only plays a minor role. This is in direct contrast to long recognized processes that lead to embrittlement in RPVs. Copper-rich precipitates are among the most prevalent mechanisms that lead to embrittlement of RPV steels that are subject to long-term service at elevated temperatures. A model that was developed to determine the relative importance of point defect clusters and copper-rich precipitates in hardening and embrittlement of RPV steels is described in NUREG/CR-6231, "A Comparison of the Relative Importance of Copper Precipitates and Point Defect Clusters in Reactor Pressure Vessel Embrittlement," prepared by Oak Ridge National Laboratory. The study concluded that point defect clusters (PDCs) could provide a major source of matrix hardening at low irradiation temperatures; however, copper-rich precipitates (CRPs) appear to be more important at high temperatures. The CRP component of embrittlement increased over high

temperature ranges for relatively high copper contents. A study describing nickel and copper interaction with regard to steel radiation sensitivity is described in NUREG/CR-2948, "Significance of Nickel and Copper Content to Radiation Sensitivity and Postirradiation Heat Treatment Recovery of Reactor Vessel Steel" prepared by Materials Engineering Associates, Inc. Experimental observations showed that high nickel content can be detrimental to the radiation resistance of steel, but the effect is clearly dependent on steel copper content...

D) The proposed model does not bound the copper content of the limiting material.

Since copper dependence is a significant parameter in determining embrittlement, data used in a proposed model should bound the copper content for the most limiting material in the reactor vessel beltine. The limiting material is plate G-307-4 with Cu = 0.27%. The highest Cu value used in the licensee's regression analysis is 0.24%. Therefore, the proposed model does not bound the copper content for the limiting material.

E) A small number of data points was used to develop the proposed model.

In comparison to the number of data points used to develop the RG methodology and equations, a relatively small number of data points was used to develop the model. Statistically, 40 data points may not be sufficient to adequately characterize a Charpy shift model.

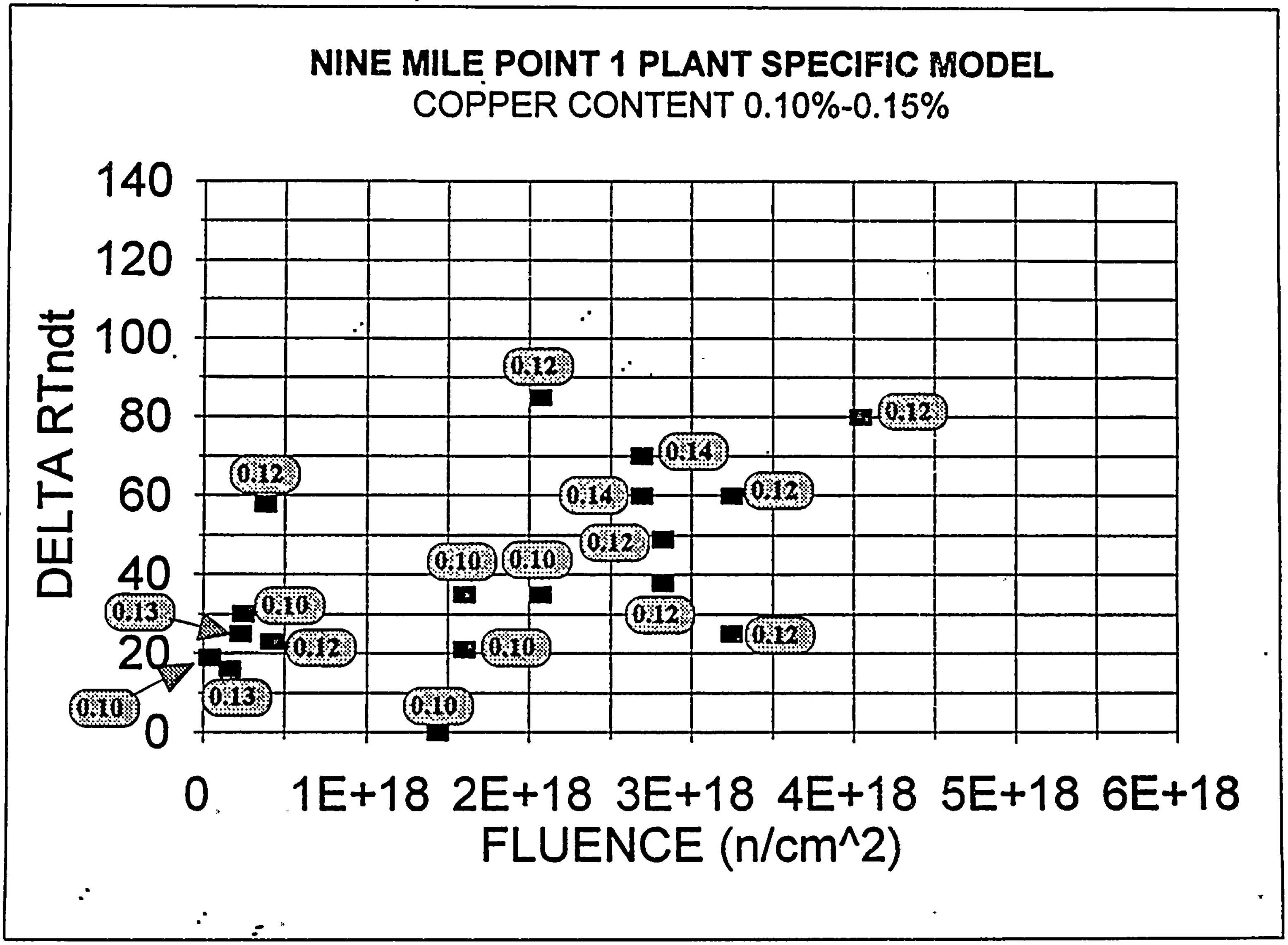
4.0 CONCLUSION

Considering the data and analyses submitted by the licensee, the staff finds insufficient technical basis to support the proposed changes to the Nine Mile Point 1 Technical Specifications.

The staff concludes that the large standard deviation (σ_{Δ}) for the proposed model would result in a larger margin term and greater uncertainty in the value for the adjusted reference temperature. The use of the smaller RG 1.99, Rev. 2 standard deviation as opposed to the value derived from the analysis of the data would not account for the variability in the model. To account for this variability, the model should use the standard deviation from the analysis of the data, and not the lower RG value as proposed by the licensee.

With respect to copper dependence, the data used in the regression analysis do not support the licensee's contention that at fast neutron fluences as high as 2E18 n/cm^2, there is no significant copper functional dependence in the Charpy shift model. In contrast, the data show that for high, intermediate, and low ranges of percent copper, higher copper contents result in higher values of ΔRT_{NDT}.

For the reasons described in this Safety Evaluation, the request for revision of the NMP-1 P/T limits is denied.


NINE MILE POINT 1 PLANT SPECIFIC MODEL
COPPER CONTENT 0.10%-0.15%
DELTA RTndt
FLUENCE (n/cm^2)
0
20
40
60
80
100
120
140
1E+18
2E+18
3E+18
4E+18
5E+18
6E+18
0.12
0.12
0.14
0.12
0.14
0.12
0.10
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0.13
0.10

FIGURE 1

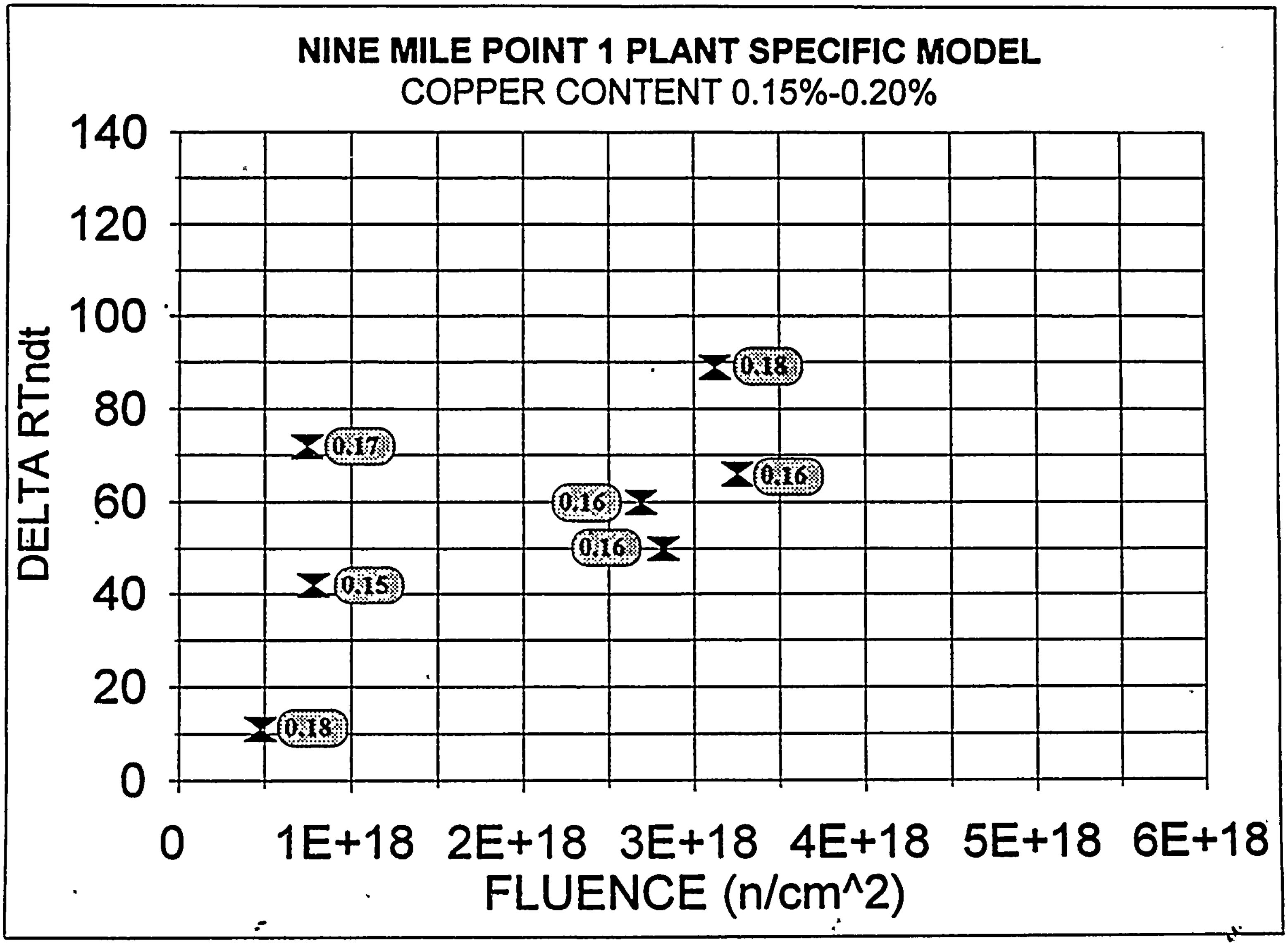
NINE MILE POINT 1 PLANT SPECIFIC MODEL
COPPER CONTENT 0.15%-0.20%
140
120
100
80
60
40
20
0
DELTA RTndt
0
1E+18
2E+18
3E+18
4E+18
5E+18
6E+18
FLUENCE (n/cm^2)
0.18
0.17
0.16
0.16
0.16
0.15
0.18

FIGURE 2

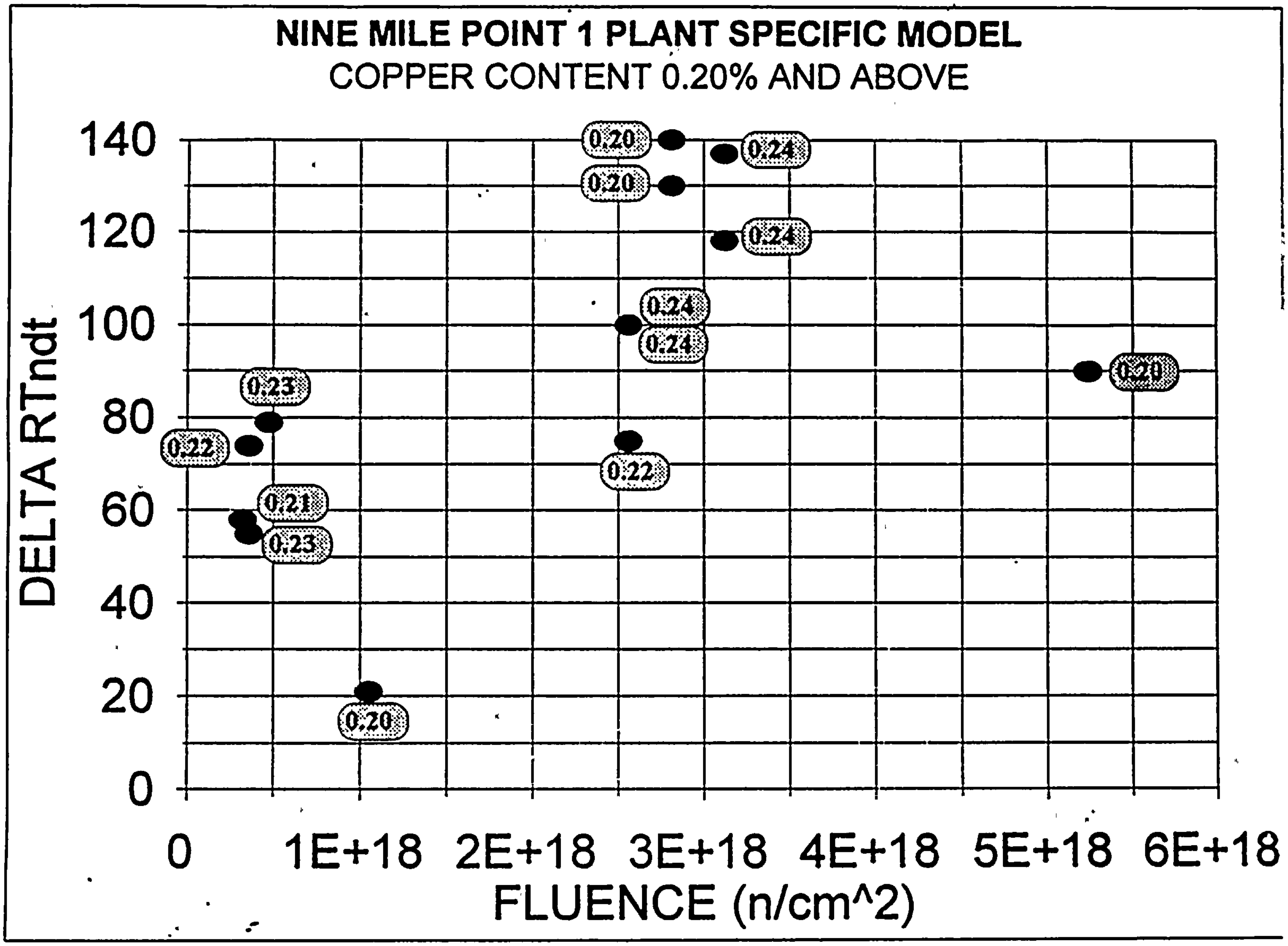
NINE MILE POINT 1 PLANT SPECIFIC MODEL
COPPER CONTENT 0.20% AND ABOVE
DELTA RTndt
FLUENCE (n/cm^2)
0
20
40
60
80
100
120
140
0
1E+18
2E+18
3E+18
4E+18
5E+18
6E+18
0.20
0.20
0.24
0.24
0.24
0.24
0.20
0.23
0.22
0.22
0.21
0.23
0.20

FIGURE 3

5.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988.

2. NUREG-0800, Standard Review Plan, Section 5.3.2: Pressure-Temperature Limits.

3. Code of Federal Regulations, Title 10, Part 50, Appendix G, Fracture Toughness Requirements.

4. Generic Letter 88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations, July 12, 1988.

5. ASME Boiler and Pressure Vessel Code, Section III, Appendix G for Nuclear Power Plant Components, Division 1, "Protection Against Nonductile Failure."

6. Dr. M. P. Manahan, Final Report Nine Mile Point Unit 1 Surveillance Capsule Program, January, 1991.

7. September 1, 1994, Letter from C. D. Terry (NMP1L 0858) to USNRC Document Control Desk, Subject: Licensee Amendment Request Revised RCS Pressure/Temperature Limits.

8. December 5, 1994, Letter from C. D. Terry (NMP1L 0882) to USNRC Document Control Desk, Subject: Proposed License Amendment - New Pressure-Temperature Limit Curves, Response to Request for Additional Information.

9. December 20, 1994, Letter from B. Ralph Sylvia (NMP1L 0888) to USNRC Document Control Desk, Subject: Proposed License Amendment - New Pressure-Temperature Limits Curves, Response to Request for Additional Information.

10. January 4, 1991, Report from M. P. Manahan to Niagara Mohawk Power Corporation, Subject: Nine Mile Point Unit 1 Surveillance Capsule Program.

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Date: September 22, 1995