

# CATEGORY 1

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SUBJECT: Fulfills 1998 annual reporting requirement of  
10CFR50.46(a)(d)(ii) for Nine Mile Point, Units 1 & 2. One new  
error reported in latest approved ECCS analysis, applying to  
non-jet pump plants & involves coding error in CORCL module.

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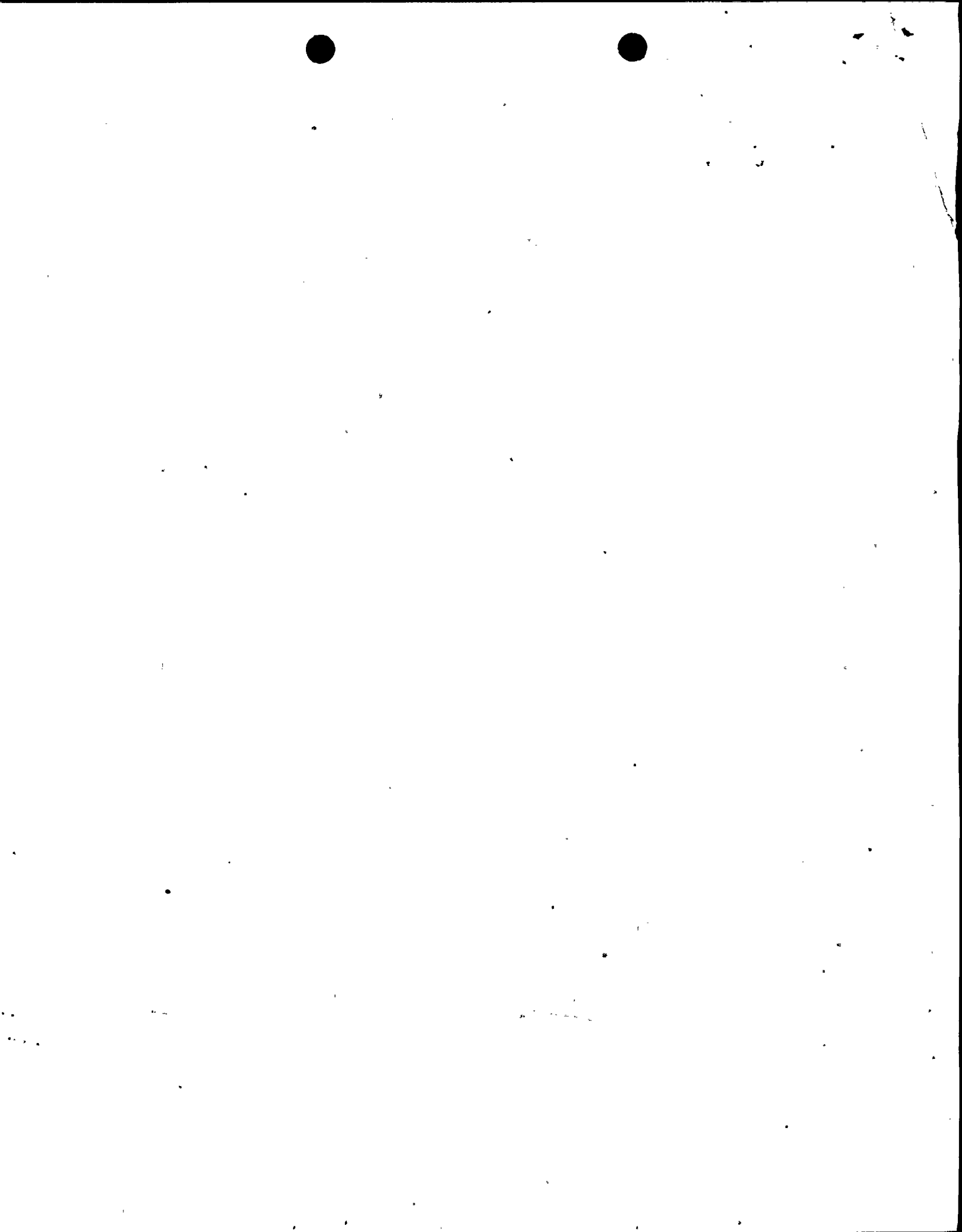
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December 22, 1998  
NMP1L 1395

U. S. Nuclear Regulatory Commission  
Attn: Document Control Desk  
Washington, DC 20555

RE:                   Nine Mile Point Unit 1  
                          Docket No. 50-220  
                                    DPR-63          

Nine Mile Point Unit 2  
Docket No. 50-410  
          NPF-69          

**Subject:       10CFR50.46 (a)(3)(ii) Report**

Gentlemen:

10CFR50.46(a)(3)(i) requires each licensee to estimate the effect of any change to or error in an acceptable emergency core cooling system (ECCS) evaluation model or in the application of such a model to determine if the change or error is significant. 10CFR50.46(a)(3)(ii) requires that for each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the licensee report the nature of the change or error and its estimated effect on the limiting ECCS analysis at least annually.

This letter is to fulfill the 1998 annual reporting requirement of 10CFR50.46(a)(3)(ii) for Nine Mile Point Units 1 and 2 (NMP1 and NMP2).

**NMPL ANALYSIS**

Since submittal of the 1997 annual report on August 29, 1997 (NMP1L 1247), the fuel vendor, General Electric (GE), has reported one new error in the latest approved ECCS analysis and has reevaluated the effect on Peak Clad Temperature (PCT) of a previously reported error.

The new error reported by GE applies to non-jet pump plants and involves a coding error in the CORCL module of the SAFER code. The error causes the diameter of the large central water rod of the assembly to be not passed as needed into one of the initialization routines. This error affects the cross-section distribution of water droplets within subchannels, causing the calculation of PCTs that are up to 40°F too high.

GE has also provided Niagara Mohawk Power Corporation (NMPC) a plant-specific evaluation of an error previously reported in the 1997 annual report, which was due to not

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modeling the reactor pressure vessel bottom head drain in the latest approved ECCS analysis. Based on GE's plant-specific evaluation, this error results in less than a 0.1°F increase in PCT instead of the less than 3°F increase that was reported in 1997. Therefore, NMPC has determined that the maximum increase in PCT is less than 0.1°F. The sum of the absolute values of the effect on PCT due to changes or errors identified since the latest approved ECCS analysis is 43.1°F. This summation includes the 40°F and 0.1°F errors discussed above, along with a 3°F error that was reported in 1993.

### NMP2 ANALYSIS

Since the 1997 annual report, GE has reported no new changes or errors in the latest approved ECCS analysis. The maximum increase in PCT due to the only change or error that has been identified since the latest approved ECCS analysis is less than 10°F. The sum of the absolute values of the effect on PCT due to changes or errors identified since the latest approved ECCS analysis is, therefore, less than 10°F.

### CONCLUSION

The latest ECCS analysis indicates a PCT of 2197°F for NMP1 and 1280°F for NMP2. After adjusting for the increase of less than 0.1°F for NMP1 and less than 10°F for NMP2, the estimated maximum PCT remains less than 2200°F for both units, which satisfies the acceptance criterion stated in 10CFR50.46(b)(1). For NMP1, the up to 40°F change in PCT due to the coding error described above and the 3°F error reported in 1993 both caused the code to overpredict PCT. Therefore, these errors did not adversely affect the estimated maximum PCT for NMP1. Additionally, for both NMP1 and NMP2, the cumulative effect on PCT of changes and errors identified since the latest approved ECCS analysis is less than 50°F and is, therefore, not significant according to the criterion stated in 10CFR50.46(a)(3)(i).

Very truly yours,



Carl D. Terry  
Vice President

Nuclear Safety Assessment and Support

CDT/IAA/sc

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