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June 13, 1996

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

Gentlemen:

Subject: Three Mile Island Nuclear Generating Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Response to Request for Additional Information - Core Reload Methodology

NRC letter dated May 10, 1996 (6710-96-3190) requested additional information regarding GPU Nuclear Topical Report TR-087, "TMI-1 Core Thermal-Hydraulic Methodology using the VIPRE-01 Computer Code." This Topical Report was submitted by GPU Nuclear on April 25, 1995 for NRC review and approval for in-house GPU Nuclear core reload design.

The attachment provides an itemized response to each of the NRC questions. If any additional information is required, please contact Mr. David J. Distel, GPU Nuclear Regulatory Affairs at (201) 316-7955.

Sincerely,

J. Knubel
Vice President and Director, TMI

Attachment
DJD/plp

c: Administrator, Region I
NRC TMI Senior Resident Inspector
NRC Senior Project Manager, TMI

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ATTACHMENT

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) ON TR-087

QUESTION 1.0

What training has been done to ensure that you have a good working knowledge of VIPRE-01, i.e. able to set up the input, to understand and interpret the output results, to understand the applications and limitations of the code, and to perform analyses in compliance with the application procedure.

RESPONSE

GPU Nuclear has been a member of the working group for the EPRI VIPRE-01 development project since the inception of the program in early 1980. As an actively participating member, GPU Nuclear attended EPRI VIPRE-01 workshops, and has presented and discussed results of VIPRE-01 analyses at regular EPRI-sponsored working group meetings. As given in Section 5.0 of the VIPRE-01 Topical Report (TR-087), the TMI-1 VIPRE-01 model has been qualified through an extensive data base analysis by comparing the VIPRE-01 predictions with critical heat flux test (CHF) data. The applicability of the TMI-1 VIPRE-01 model and GPU Nuclear's capability in performing licensing analyses have been further demonstrated through benchmark analysis to the TMI-1 Cycle 10 vendor analysis results, such as core thermal hydraulic (T-H) design and setpoint analyses. These benchmark analyses are included in the reload methodology Topical Report, TR-092P (Reference 1), which was submitted to NRC for approval on February 27, 1996. The above analysis activities over the last 15 years have provided us with an extensive training basis and an in-depth working knowledge of VIPRE-01 regarding setting up input data, interpreting output, applications in reload design, and code limitations.

GPU Nuclear is developing TMI-1 Fuel Standards (application procedures) conforming to the recommendations given in the NRC Generic Letter (GL) 83-11. The Fuel Standards describe how to prepare VIPRE-01 input decks and to interpret output data for various applications in reload design including application limitations for the licensing analyses. These Standards will be referenced in GPU Nuclear licensing analyses and will become the training basis for VIPRE-01 applications in reload design.

QUESTION 2.0

2a. **What is the source of the data used to benchmark the VIPRE code analysis?**

2b. **Justify why the use of 211 data points is sufficient.**

2c. **What statistical methods have been included as a result of the size of the data set?**

RESPONSE

2a. As mentioned in Section 5.1 of the VIPRE-01 Topical Report, TR-087, the source data is the CHF test data for the B&W Fuel Company's (herein after referred to as Framatome Cogema Fuels (FCF)) 15X15 fuel assemblies with the intermediate zircaloy grid design taken from Reference 2. The test was performed utilizing bundles with a 12-foot-long 5x5 rod array with a non-uniform axial flux shape. Three different test geometries were utilized to represent FCF15x15 fuel with zircaloy grids: (1) unit subchannel rod geometry (5x5 all rod array) without an instrument guide tube, (2) same geometry as (1) except an instrument guide tube was placed at the center of the 5x5 array, and (3) an inter-assembly geometry where four fuel assemblies intersect.

2b. As described in Reference 2, the BWC CHF correlation was originally derived for application to 17x17 Mark C fuel based on 601 data points. FCF obtained an additional 211 data points based on a 15x15 Mark BZ fuel test geometry to support the applicability of the BWC correlation to the 15x15 fuel assembly.

Therefore, the use of 211 data points represents the total number of the available CHF test data points directly applicable to TMI-1 fuel (15x15 array with 6 intermediate zircaloy grids) and core thermal-hydraulic conditions (coolant flow, inlet temperature, pressure as given in Table 5.1 of TR-087). The incorporation of the additional data points would result in use of data which is not directly applicable to TMI-1. Furthermore, as described in 2c below, the sample size of 211 is considered adequate for the statistical handling of CHF data.

FCF analyzed the 211 data points by using the BWC correlation and LYNX2 code and its applicability was approved by NRC (Reference 2). In TR-087, GPU Nuclear analyzed the same 211 data points as above by utilizing VIPRE-01/BWC combination instead of LYNX2/BWC coupling in Reference 2.

2c. The statistical method used for the correction of the finite sampling size, as given in Section 5.1.3 and 5.1.4 of TR-087, was the one-sided tolerance theory by Owen (Reference 3).

By utilizing this method the design DNBR was determined based on the criterion for a 95% population protection with a 95% confidence (or tolerance) level. Based on the sample size of 211 and with given statistics of the CHF test data (i.e., a standard deviation, σ, of 0.069696 and a measured-to-predicted ratio of 0.98206 as shown in Table 1), the 95% tolerance factor with a 95% population protection is determined to be 1.8327. The 95/95 design DNBR based on this tolerance factor is 1.1705. However, the design DNBR for the TMI-1 Technical Specifications is conservatively chosen to be 1.18.

QUESTION 3.0

3a. **Describe how your QA program complies with the requirements of Appendix B of 10CFR50.**

3b. **Do you have provisions for evaluating updates and implementing applicable updates related to VIPRE?**

3c. **What are the planned provisions for informing the code developer of any problems or errors discovered while using the VIPRE?**

RESPONSE

3a. GPU Nuclear Operational Quality Assurance Plan (GPU Nuclear-OQA Plan), 1000-PLN-7200.01, which has been reviewed and approved by NRC, describes how GPU Nuclear complies with the requirements of Appendix B of 10CFR50. Independent reload analyses and calculations are within the scope of the GPU Nuclear OQA Plan. As such, these activities are performed in accordance with the engineering procedures controlled by the GPU Nuclear OQA Plan.

3b. GPU Nuclear Procedure, Computer Program Control, 5000-ADM-7340.01, specifies requirements for evaluating and implementing computer programs. The GPU Nuclear program owner is formally notified of updates by the EPRI VIPRE-01 Users Maintenance Group. These updates are evaluated by the program owner in accordance with Procedures 5000-ADM-7340.01, Sections 4.2.4, 4.3, and 5.2, which prescribe requirements to incorporate code error corrections and revisions, and perform a new verification and validation for significant revisions to computer codes. This procedure is implemented and controlled in accordance with the GPU Nuclear OQA Plan.

3c. For any problems or errors identified with the use of a program, the GPU Nuclear program owner is required to evaluate the nature of problems/errors, and notify the code developer and the affected users via the EPRI VIPRE-01 Users Maintenance Group of which GPU Nuclear is a member. The GPU Nuclear program owner also revises the affected documents in accordance with requirements set forth in the GPU Nuclear Procedure, Computer Program Control, 5000-ADM-7340.01, Section 4.4, Error Reporting.

QUESTION 4.0

What type of fuel was used in the qualification of the VIPRE-01/BWC? Explain why that particular fuel type was selected.

RESPONSE

As mentioned in the responses to Questions 2a and 2b, the fuel type used for the qualification is the FCF Mark BZ fuel (15x15 rod array) with a zircaloy intermediate grid design. The main reason for this fuel type selection is that TMI-1 employs this type of fuel, i.e., 15x15 rod array with the zircaloy grid design. The fuel geometry (rod diameter, rod and assembly pitch, flow area and spacer grids) used for the CHF tests is identical

with the fuel assemblies in the TMI-1 core. Also, the range of TMI-1 operating parameters is well within the variation of the thermal-hydraulic parameters (pressure, inlet enthalpy, flow rate and heat flux) for the test.

QUESTION 5.0

Table 5.1 gives a range of validity for the DNB correlation. Please explain how your methodology assures that the correlation will not be used outside of the acceptable range.

<u>RESPONSE</u>

Table 5.1 of TR-087 provides the applicable range of system local conditions (pressure, flow rate, and quality) for VIPRE-01/BWC. As mentioned in the response to Question 1.0, a Fuel Standard is being developed in which limitations and restrictions are explicitly stated for the use of VIPRE-01/BWC in reload design. In addition, other VIPRE-related Fuel Standards will contain a section which specifies limitations and restrictions. These limitations and restrictions will be identified as verification items in the analysis verification process.

QUESTION 6.0

Please add the 95th percentile line on Figure 5.1, "Measured CHF vs. VIPRE-01/BWC Predicted CHF."

<u>RESPONSE</u>

Figure 1 shows the CHF data against the 95/95 criterion line.

REFERENCES:

Reference 1. GPU Nuclear Topical Report No. TR-092P, Rev. 0, "TMI-1 Reload Design and Setpoint Methodology," December 8, 1995.

Reference 2. B&W Topical Report No. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," April 1985

Reference 3. D. B. Owen, "Factors for One-Sided Tolerance Limits and for Variable Sampling Plans," SCR-607, Sandia Corporation, Albuquerque, New Mexico, March 1963

Table 1. BWC CHF Test Data Statistics with the VIPRE-01/BWC Predictions

		MEASURED-TO-PREDICTED RATIO	
TEST GEOMETRY	NO. OF DATA POINTS	MEAN VALUES	STD DEVIATIONS
NORMAL MATRIX	70	0.98246	0.081255
GUIDE TUBE	44	0.98162	0.073445
ASSEMBLY INTERSECTION	97	0.98494	0.057800
ALL	**211**	**0.98206**	**0.069696**

FIGURE 1. BWC CHF DATA AGAINST THE 95/95 PROTECTION LINE