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Director
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U S Nuclear Regulatory Commission
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Error in COBRA IIIC/MIT Model Setup

An error has been identified and corrected in the COBRA IIIC/MIT model setup. This error affects all calculations done previously for Prairie Island using an 1/8 core COBRA IIIC/MIT model. This information is being supplied for your information, since it affects documents previously submitted to the NRC.

The error was identified during benchmark calculations to compare VIPRE-01 to COBRA IIIC/MIT. Identical model setups with VIPRE produced MDNBR results which were approximately 3% - 5% lower than those from COBRA IIIC/MIT. Upon investigation we found that, in COBRA IIIC/MIT when using the simplified channel input, i.e. J=2 on card type T2, for lumped channels, the wetted perimeter is calculated based on assuming a can around the assembly, i.e. a BWR assembly. The code does not identify this in the manual or on the output. This larger wetted perimeter increases the frictional pressure loss in the lumped channels and hence increased the flow diverted into the hot channel. The increased flow increases the MDNBR by approximately an equivalent fraction. This error was not detected by the NSP code setup review or by the independent consultant's review. The error was corrected by specifying the channel area, heated and wetted perimeter directly on card T2 for the lumped channels, as is done for the hot subchannels. Additionally, the entire code setup has been thoroughly reviewed again, by NSP, to ensure accuracy.

The effect of the error on documents submitted to the NRC is as follows:

NSPNAD-8102P Rev.1 and Rev.2, "Reload Safety Evalatuation Methods for Application to PI Units"

The single channel and the 1/8 assembly COBRA calculations are not affected by this error. The 1/8 core COBRA calculations reported in the document are approximately 3% - 5% too high. These calculations were not repeated since in all cases the corrected values will be lower and hence more conservative than the reported value. There is no effect on the sensitivity studies in Appendix C since the same relative error can be applied to each case.

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NSPNAD-8406, "Prairie Island Unit 1 and 2 Safety Evaluation with Increased Enthalpy Rise Factor."

There are four sections to this report; transient analysis, rod bow penalty, safety limit curves, and overtemperature WT setpoint verification. All of these items have been recalculated for the current operating cycles and will be reported in a revision to this report.

The error also affected the Final Reload Design Reports for Prairie Island Units 1 and 2. These reports are not routinely submitted to the NRC. All cycles have been reviewed and no violations of the applicable acceptance criteria occurred. The effect of the error on Unit 2 Cycle 9 is shown in the attached Table 1. Other cycles show similar results. The initial MDNBR decreased by 5.1%. The corrected MDNBR for the limiting Class II and III transient, the slow rod withdrawal, decreases by 6.6% to 1.41 (including rod bow penalty). This is still well above the acceptable MDNBR of 1.3. The locked rotor transient, a Class IV event, showed an increase in the number of failed fuel rods to 12.3%. This is still well below the acceptance criteria of 20%. The large steam line break transient, a Class IV event, was not rerun due to the large safety margin which currently exists, i.e. MDNBR=3.7 vs. a fuel failure criteria of 20%.

We would be glad to answer any questions you may have on this matter.

David Musolf
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DMM/TMP/bd

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Attachment

TABLE 1

Prairie Island Unit 2 Cycle 9
Comparison of RSE Results

Transient Parameter	NSPNAD-8404P		Acceptance Criteria
	Original	Corrected	
Base Case, BOC, HFP			
MDNBR	1.95	1.85	> 1.3
Slow Rod Withdrawal			
$MDNBR_{NB}$	1.56	1.46	> 1.3
$MDNBR_{B}$	1.51	1.41	> 1.3
Locked Rotor			
Fuel Failure	9.1%	12.3%	< 20%
Large MSL Break			
MDNBR	3.7	-	