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L47-91(01-02)-LP
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ILLINOIS POWER

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JSP-0002-91
January 2, 1991

Docket No. 50-461

Nu..ear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Clinton Power Station
Clarification Regarding Previously Submitted
Proposed Amendment of Facility Operating
License No. NPF-62

Dear Sir:

By letter dated September 6, 1988 (reference U-601239), Illinois Power (IP) requested amendment of Facility Operating License No. NPF-62 for the Clinton Power Station (CPS) to support the first refueling and subsequent reactor operation in the Maximum Extended Operating Domain (MEOD) and/or with reduced feedwater temperatures. IP's request was subsequently approved on January 31, 1989 as Amendment Number 18 to the CPS Operating License. IP recently identified an error in Attachments 6 and 7 to the September 6, 1988 letter (i.e., "SUPPLEMENTAL RELOAD LICENSING SUBMITTAL FOR CLINTON POWER STATION UNIT 1, RELOAD 1, CYCLE 2, 23A5921 Rev. 0" and "MAXIMUM EXTENDED OPERATING DOMAIN AND FEEDWATER HEATER OUT-OF-SERVICE ANALYSIS FOR CLINTON POWER STATION, NEDC-31546P, August 1988", respectively). This letter is being provided to identify this error and provide an evaluation of its significance and its impact on the results of the noted analyses.

As required by Section III of the ASME Boiler and Pressure Vessel Code, the analyses presented in Attachments 6 and 7 of IP's September 6, 1988 letter contained the results of overpressurization analyses. These overpressurization analyses were based upon a main steam line isolation valve (MSIV) closure transient which is terminated by a reactor scram as a result of high neutron flux (reference Section 13 of Attachment 6 to U-601239 and Section 2.5 of Attachment 7 to U-601239). The Attachment 6 results for the reload analysis were performed utilizing GEMINI methods and were based upon reactor operation at 102% power at the time of the transient. The results of this analysis, which yielded a peak vessel pressure of 1247 psig, are graphically depicted on Figure 5 of Attachment 6 to U-601239. The Attachment 7 results for operation in the MEOD were based upon reactor operation at 102% power and 107% core flow at the time of the transient. The results of this analysis, which yielded a peak vessel pressure of 1245 psig, are summarily shown on Table 2-5 and are graphically depicted on Figure 2-9 of Attachment 7 to U-601239.

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These overpressurization analyses were performed to demonstrate that the installed Safety/Relief Valve (SRV) capacity at CPS is adequate to prevent the reactor vessel pressure from exceeding the Reactor Coolant System Pressure Safety Limit which is equivalent to the ASME Code limit of 1375 psig. As described in NEDE-24011-P-A-8, "General Electric Standard Application for Reactor Fuel," May 1986, demonstration of compliance with Section III of the ASME Boiler and Pressure Vessel Code should have been based upon actuation of the SRVs in the spring "safety" mode of operation. However, as shown on the noted figures of U-601239, reactor vessel pressure relief was assumed to be based upon SRV actuation in the power-actuated "relief" mode of operation.

Following discovery that the noted analyses for CPS operating cycle 2 were performed assuming SRV operation in the "relief" mode, IP requested General Electric (GE) to determine the impact of this nonconservative assumption on the results of the above analyses. Because GE was performing the reload analyses for CPS operating cycle 3 at the time, an evaluation of overpressurization protection during CPS operating cycle 3 was performed for both modes of SRV operation. This evaluation concluded that the peak reactor vessel pressure would be approximately 0.5 psig higher when all SRVs are assumed to actuate in the "safety" mode than actuate in the "relief" mode.

Because the highest calculated peak vessel pressure during CPS operating cycle 2 was 1247 psig (based upon the reload analyses presented in Attachment 6 to U-601239) and the ASME Code limit is 1375 psig, IP has concluded that the error (less than 0.5 psig) contained in U-601239 is not safety significant. Further, adequate margin existed during CPS operating cycle 2 to prevent the reactor vessel pressure from exceeding the Reactor Coolant System Safety Limit and the ASME Code limit.

Sincerely yours,

J. S. Perry
Vice President

DAS/rgw

cc: Regional Administrator, Region III, USNRC
NRC Clinton Licensing Project Manager
NRC Resident Office
Illinois Department of Nuclear Safety