ATTACHMENT 2

LASALLE COUNTY STATION
UNITS 1 AND 2

Docket Nos. 50-373 and 50-374

Facility Operating License Nos. NPF-11 and NPF-18

Proposed Technical Specifications Markups

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~either: (1)~~ Section 9.1.2 of the UFSAR~~, or (2) AREVA NP Inc. Report No. ANP-2843(P), "LaSalle Unit 2 Nuclear Power Station Spent Fuel Storage Pool Criticality Safety Analysis with Neutron Absorbing Inserts and Without Boraflex," Revision 1, dated August 2009, for the Unit 2 spent fuel storage racks with rack inserts~~.

b. A nominal 6.26 inch center to center distance between fuel assemblies placed in the storage racks.

c. For Unit 2 only, spent fuel shall only be stored in storage rack cells containing a neutron absorbing rack insert. The neutron absorbing rack inserts shall have a minimum certified ^{10}B areal density greater than or equal to 0.0086 grams $^{10}B/cm^2$. The approved inserts are those described in Attachment 4 to the letter from P. Simpson to the NRC, dated October 5, 2009.

d. Fuel assemblies having a maximum k_{inf} of 1.275 in the normal reactor core configuration at cold conditions.~~The combination of U-235 enrichment and gadolinia loading shall be limited to ensure fuel assemblies have a maximum k-infinity of 0.9185 for all lattices in the top of the assembly, a maximum k-infinity of 0.8869 for all lattices in the intermediate portion of the assembly, and a maximum k-infinity of 0.8843 for all lattices in the bottom of the assembly as determined at 4°C in the normal spent fuel pool in-rack configuration. The bottom, intermediate, and top zones are between 0"-96", 96"-126", and greater than 126" above the bottom of the active fuel.~~

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5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. The Rod Block Monitor Upscale Instrumentation Setpoint for the Rod Block Monitor—Upscale Function Allowable Value for Specification 3.3.2.1.

5. The OPRM setpoints for the trip function for SR 3.3.1.3.3.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

~~1. ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors."~~

~~2. ANF-913(P)(A), "COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis."~~

~~3. ANF-CC-33(P)(A), "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option."~~

~~4. XN-NF-80-19(P)(A), "Advanced Nuclear Fuel Methodology for Boiling Water Reactors."~~

~~5. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel."~~

~~6. EMF-CC-074(P)(A), Volume 4 – "BWR Stability Analysis: Assessment of STAIF with input from MICROBURN-B2."~~

~~7. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model."~~

~~8. XN-NF-84-105(P)(A), "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis."~~

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5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

~~9. EMF-2209(P)(A), "SPCB Critical Power Correlation."~~

~~10. ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs."~~

~~11~~1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."

~~12. NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods."~~

~~13. EMF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model."~~

~~14. EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2."~~

~~15. NEDC-33106P, "GEXL97 Correlation for Atrium-10 Fuel."~~

~~16. EMF-2245(P)(A), "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel."~~

~~17. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model."~~

~~18. NEDO-32465-A, "BWR Owners' Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," August 1996.~~

~~19. ANF-1358(P)(A), "The Loss of Feedwater Heating Transient in Boiling Water Reactors."~~

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