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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station OP1-17
Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION
REQUEST FOR ADDITIONAL INFORMATION
REGARDING PROPOSED AMENDMENT NO. 238
TO LICENSE NPF-14: MCPR SAFETY LIMITS
PLA- 5406 **Docket No. 50-387**

Reference: 1) PLA-5320, R. G. Byram (PPL) to USNRC, "Proposed Amendment No. 238 to License NPF-14: MCPR Safety Limits", dated May 31, 2001.

On May 31, 2001, PPL Susquehanna, LLC (PPL) proposed a revision to the Susquehanna Steam Electric Station Unit 1 Technical Specifications (Reference 1). The revisions, if approved, would update the MCPR safety limit in Technical Specification Section 2.1.1.2.

The need for additional information was identified during a teleconference held November 14, 2001. The purpose of this letter is to provide the additional information, which is contained in Attachment 1.

The questions and our responses are contained in Attachment 1.

If you have any questions, please contact Mr. M. H. Crowthers at (610) 774-7766.

Sincerely,

R. G. Byram

Attachment

copy: NRC Region I
Mr. S. Hansell, NRC Sr. Resident Inspector
Mr. D. S. Collins, NRC Project Manager
Mr. D. J. Allard, PA DEP

A001

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC : Docket No. 50-387

REQUEST FOR ADDITIONAL INFORMATION REGARDING PROPOSED AMENDMENT NO. 238 TO LICENSE NPF-14: MCPR SAFETY LIMITS

Licensee, PPL Susquehanna, LLC, hereby files a revision to its Facility Operating License No. NPF-14 dated July 17, 1982.

This amendment contains a revision to the Susquehanna SES Unit 1 Technical Specifications.

PPL Susquehanna, LLC
By:

R. G. Byram
Sr. Vice-President and Chief Nuclear Officer

Sworn to and subscribed before me this 5th day of December, 2001.

Nancy J. Lannen

Notary Public

Notarial Seal
Nancy J. Lannen, Notary Public
Allentown, Lehigh County
My Commission Expires June 14, 2004

Attachment 1 to PLA-5406

Response to NRC Request for Additional Information

Response to NRC Request for Additional Information

Question #1

NRC requested a flow chart be provided to: (1) show which TS 5.6.5.b analysis methods are used to calculate the MCPR Safety Limit, (2) identify the calculation flow and (3) indicate which analyses are performed by PPL and which are performed by Framatome - ANP.

Response

Susquehanna Unit 1 Technical Specification Section 5.6.5.b lists a number of approved methodologies that are applied to analyze the behavior of SSES during Cycle 13. Several of these methodologies are applicable to the calculation of the MCPR Safety Limit. Figure 1 provides a flowchart which illustrates the NRC approved methods listed in Technical Specification Section 5.6.5.b, which apply to the separate portions of the Safety Limit analysis and the organization that performs the analysis (PPL Susquehanna, LLC or Framatome - ANP).

Question #2

The NRC staff noted that the MCPR Safety Limit for Cycle 13 has increased by 0.01, and requested an explanation of the phenomena causing the U1C13 MCPR Safety Limit to increase. Specifically, NRC questioned the impact of the uniform core of ATRIUM™-10 fuel and the small power uprate. In addition, NRC requested an explanation of which aspects of the cycle core design caused the change.

Response

NRC has requested further clarification on the increase in the MCPR Safety Limit values between U1C12 and U1C13. Specifically, three factors are addressed: the transition from a mixed core (containing both Framatome 9x9-2 and ATRIUM™-10 fuel) to an all ATRIUM™-10 core, power uprate, and core design. This discussion is provided below.

The presence of 9x9-2 fuel in the prior cycle is not a contributor to the difference in calculated MCPR Safety Limit values. This is due to the fact that the 9x9-2 assemblies were high exposure / low power assemblies that do not contribute any calculated pins in boiling transition. Thus, the transition from a mixed core (containing 9x9-2 and ATRIUM™-10 fuel) to the U1C13 (all ATRIUM™-10) core did not affect the calculated MCPR Safety Limit.

For a given core configuration, an increase in core power flattens the core radial power distribution due to void feedback, and a flatter distribution (i.e, more fuel assemblies at or near the peak assembly power) will increase the number of pins calculated to be in boiling transition. However, due to the fact that the power increase from U1C12 to U1C13 is so small (i.e., 1.5%), the increased power level is only a small contributor to the increase in the calculated MCPR Safety Limit between U1C12 and U1C13.

Past reload analyses in which no change in rated core power occurred have shown that increases in calculated MCPR Safety Limit (i.e., 0.01) can occur solely as a result of the core design. Core designs have changed over time due to both the move to 24-month cycles and the emphasis on improving fuel utilization. While maintaining the required safety margins, designers balance the economic and operating margin considerations while developing the core design. The core design affects the radial power distributions used in the MCPR Safety Limit calculations due to loading pattern and control rod sequencing. The radial power distribution is calculated by the references listed below as identified in Figure 1. The radial power distribution, particularly for the highest power assemblies, is the dominant contributor to the increase in the safety limit. Figure 2 provides a graphical comparison of the U1C12 and U1C13 radial power histograms. Since more assemblies are at or near the peak assembly power in U1C13 than in the previous cycle, this would tend to put more pins closer to boiling transition. As a result, it is expected that U1C13 would require a higher Safety Limit than U1C12 to assure that 99.9% of the pins are expected to avoid boiling transition. Thus, the small increase in MCPR Safety Limit from U1C12 to U1C13 is mainly due to the specific U1C13 core design.

To summarize, the increase in the MCPR Safety Limit is principally attributable to the cycle specific core design (i.e., radial power distribution of fresh, high power assemblies), and is not principally attributable to either the small power uprate or to the transition to a full core of ATRIUM™-10 fuel.

References:

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July 1992.

2. PL-NF-90-001-A, Supplement 2-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: CASMO-3G Code and ANFB Critical Power Correlation," July 1996.

Figure 1
MCPR Safety Limit Methodology

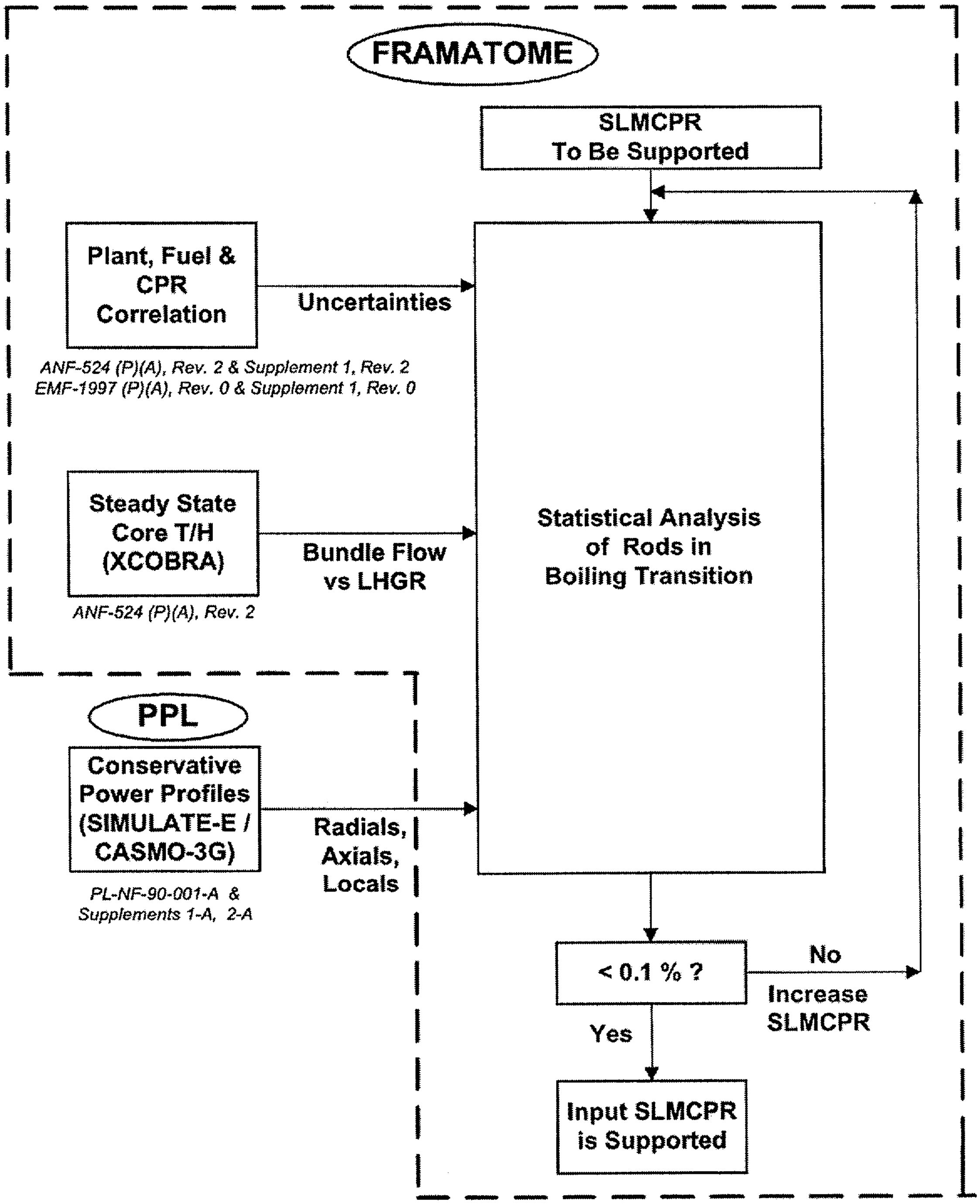

FIGURE 2: Relative Radial Power Distribution
(Top 50 Assemblies in Quarter Core)