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SUBJECT: Responds to NRC 910801 request for addl info on Topical Rept PP&L-NF-90-001 re licensing methods,plan for U1C7.

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AUG 29 1991

Director of Nuclear Reactor Regulation
Attention: Dr. W. R. Butler, Project Director
Project Directorate I-2
Division of Reactor Projects
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
LICENSING METHODS : PLAN FOR U1C7 — **Docket Nos. 50-387 and 50-388**
PLA-3641 FILES R41-2/A7-8C

References: 1. Letter, James J. Raleigh (NRR) to Harold W. Keiser (PP&L), "Request for Additional Information on PP&L Topical Report PL-NF-90-001, Susquehanna Steam Electric Station, Units 1 and 2, (TAC Nos. 75999 76000)," dated August 1, 1991.

2. "Application of Reactor Analysis Methods for BWR Design and Analysis", PL-NF-90-001, August, 1990.

3. "Qualification of Transient Analysis Methods for BWR Design and Analysis", PL-NF-89-005, December, 1989.

Dear Dr. Butler:

Attached are PP&L's responses to Questions 1 through 15 from your request for additional information (Reference 1), which PP&L received on August 5, 1991. PP&L currently plans to submit responses to the remaining questions, which relate to PP&L's proposed SCU methodology, at a later date. Although the staff posed these additional questions on the SCU method, the "general concerns" presented in Reference 1 clearly indicate the staff's reluctance to accept it. PP&L believes that its SCU methodology, although different from traditional methods, is technically valid, conforms to the applicable regulations, and has been shown to produce MCPR operating limits which are comparable to previously accepted methods. We do not concur with the staff's assertion that the SCU method "results in a substantial nonconservative reduction in CPR margin".

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Nevertheless, PP&L's most pressing need at this time is to determine an appropriate plan for licensing our next scheduled reload, which is Unit 1 Cycle 7 (U1C7). Our plan is to submit licensing analyses for U1C7 (scheduled for submittal in December 1991) based on PP&L methods as described in PL-NF-90-001 (Reference 2), with the exception that the SCU approach will not be used. The MCPR operating limits for those events specified in PL-NF-90-001 as using SCU will be generated by an alternate approach which is described in Attachment 2. The alternate approach utilizes separate calculations of the MCPR safety limit and the Δ CPR and is based on NRC approved methodologies. Although the alternate methodology produces slightly higher MCPR operating limits than the SCU methodology described in PL-NF-90-001, PP&L is willing to use the alternate approach for subsequent reload analyses, if the issue of the SCU methodology cannot be favorably resolved. It is our belief that this will allow our customers to receive a return on the substantial investment made in developing PP&L's licensing methodology at the earliest possible time, given the staff's current concerns.

The alternate approach for the pressurization transients (GLRWOB and FWCF) will be the same approach used in the NRC approved Siemens Nuclear Power Corporation (SNP - formerly Advanced Nuclear Fuels) methods. SNP is the current fuel supplier for PP&L. Instead of statistically combining the RETRAN code uncertainty with the uncertainties on other parameters (e.g., scram speed), a deterministic analysis will be performed to calculate the Δ CPR utilizing a 10% penalty on calculated integral power to account for RETRAN code uncertainty. The alternate approach for the rod withdrawal error will entail a deterministic calculation to calculate the Δ CPR.

Finally, PP&L requests that the staff review of the PP&L transient analysis methods topical report PL-NF-89-005 (Reference 2) be documented in an NRC safety evaluation as soon as possible; to the best of our knowledge, there are no outstanding NRC concerns on this document.

Should you wish to discuss this transmittal further, please contact Mr. R.R. Sgarro at (215) 774-7916.

Very truly yours,

H W Keiser

H. W. Keiser

Attachment

cc: NRC Document Control Desk (original)
NRC Region I
Mr. G. S. Barber, NRC Sr. Resident Inspector
Mr. J. J. Raleigh, NRC Project Manager
Mr. L. I. Kopp, NRR/SRXB - OWFN

ATTACHMENT 1

Responses to NRC Questions 1 to 15 on PL-NF-90-001

QUESTION 1

The analyses of Section 3.4 in which the limiting events are identified were performed for a specific core/plant configuration. How will it be insured that the bounding events remain bounding for future reload analyses (e.g., the turbine trip w/o bypass versus the generator load rejection w/o bypass)?

RESPONSE 1

The events presented in Section 3.4 of PL-NF-90-001 are:

1. Turbine Trip
2. MSIV Closure (Position Scram)
3. Loss of Condenser Vacuum
4. Recirculation Pump Trip
5. Inadvertent HPCI Startup

The Turbine Trip without Bypass (TTWOB) will be evaluated for each cycle to determine if the Generator Load Rejection without Bypass (GLRWOB) is the limiting event, and the limiting event will be used in the transient analyses.

The MSIV Closure with valve position scram is inherently less limiting than the GLRWOB, due to the much slower valve closure time (> 2 seconds as opposed to less than 0.150 seconds). The analysis presented in Section 3.4.2 produced a ΔCPR of 0.0 since power decreases rapidly due to the position scram.

The Loss of Condenser vacuum consists of a turbine trip with some bypass relieving capacity available. Thus, it is less limiting than the TTWOB which will be considered for each cycle.

The Recirculation Pump Trip (RPT) initiates a rapid flow decrease which, in turn, produces a rapid decrease in core power due to void reactivity feedback. The calculated ΔCPR for this event is 0.02. The small changes that occur in neutronic and flow related parameters from one cycle to the next (e.g., void and Doppler coefficients) will not cause this event to become limiting.

The analysis of the inadvertent HPCI startup presented in Section 3.4.5 contains significant conservatisms (lower than anticipated water temperature and no credit taken for the high flux and high thermal power trips). The value calculated for this event presented in Section 3.4.5 of PL-NF-90-001 was less than 0.10. The relatively small variations in Doppler and void coefficients from cycle to cycle are not expected to increase this value significantly (the ΔCPR would have to increase by more than a factor of 2 for this event to become the limiting event. Therefore, there are no anticipated cycle-to-cycle changes which would make the inadvertent HPCI startup the limiting event.

<u>QUESTION 2</u>

How is the most limiting core power for the generator load rejection w/o bypass (GLRWOB) determined (p. 108, Item 5)?

<u>RESPONSE 2</u>

The limiting power level for the GLRWOB is determined by a parametric study on core power for the specific reload core being analyzed.

<u>QUESTION 3</u>

Since the feedwater controller failure (FWCF) event is sensitive to the bypass valve capacity and a best estimate value is assumed, how will the uncertainty in this parameter be accommodated?

<u>RESPONSE 3</u>

The value used for bypass capacity is based on plant measurements performed on the as-built configuration. The bypass capacity based on plant measurements is 25.34% of rated steam flow, which is very close to the GE design value of 25%. The capacity of each bypass valve was determined by calculating the difference in main steam flow between the bypass valve open and closed conditions. The uncertainty on this difference in main steam flows is not expected to be significant.

The significant conservatisms inherent in the FWCF licensing method (described in Section 3.2.3 of PL-NF-90-001) are sufficient to produce a conservative ΔCPR for this event, and an additional penalty for bypass capacity uncertainty is not used.

QUESTION 4

How is the code and modeling uncertainty including kinetics parameters, RETRAN plant input parameters and core power distribution accounted for in the recirculation flow controller failure (RFCF) event?

RESPONSE 4

The limiting Recirculation Flow Controller (RFCF) event is a very slow two pump runup. The principal uncertainties which affect the calculated RCPR are the uncertainties on the Doppler and void coefficients. An allowance for these uncertainties is explicitly included in PP&L's licensing methodology for the RFCF as described in Section 3.3.2 of PL-NF-90-001.

The principal RETRAN input parameter which affects the calculated RCPR for the RFCF is the rate of flow increase. PP&L's methodology uses the most limiting rate of flow increase (i.e., the rate which produces the maximum RCPR) and, thus, the results bound all possible rates.

The RFCF event entails a moderate upward shift in the axial power distribution. Since the sensitivity of MCPR to axial power distribution is relatively small, an uncertainty in axial power distribution would not have a significant effect on calculated ΔCPR for the RFCF event and thus is not explicitly included in the RFCF analysis.

QUESTION 5

In the cycle-specific RFCF analysis of the effects of the maximum combined flow limit (MCFL), what changes will be made if the MCFL is reached? Will the moderator density corrected cross sections be required?

RESPONSE 5

During the RFCF event, at the time the Maximum Combined Flow Limit (MCFL) is reached, the ability of the pressure regulator to control further pressure increases is lost. As a result, the reactor pressure slowly increases until either a high reactor pressure, high neutron flux, or high simulated thermal power trip occurs. This phenomenon is modelled in PP&L's RETRAN analysis. The pressure rise obtained after reaching a steam flow greater than the MCFL is relatively slow compared to a GLRWOB and the moderator density correction is not needed, since the density changes are small. Analyses of the GLRWOB have shown that using the moderator density correction on the cross section parameters yields a slightly lower calculated ΔCPR than is obtained by not using the correction. Thus, the moderator density correction is not used for the RFCF event analysis, since this is conservative.

QUESTION 6

What is the effect of using the moderator density corrected cross sections on the MSIV overpressurization analysis? What change in core voids occurs up to the time of peak pressure? On what basis is the corrected cross section set applied to the GLRWOB and the uncorrected cross section set applied to the MSIV overpressurization analysis?

RESPONSE 6

Using the moderator density corrected cross sections for the MSIV closure overpressurization analysis would decrease the calculated peak pressure slightly (peak pressure was decreased by 1 psi as shown in Table 3.5-4 of PL-NF-90-001). For the sample MSIV closure analysis presented in Section 3.5.6 of PL-NF-90-001, the core exit void fraction changes from 71.2% to 63.6%

from time zero to the time of peak pressure. The unadjusted cross sections are used for the MSIV closure event, because it is conservative for peak pressure calculations.

QUESTION 7

Does the use of an axially averaged gap conductance give a conservative hot-bundle ΔCPR for both top and bottom peaked axial shapes? If not, how will this dependence be accounted for?

RESPONSE 7

A parametric study was performed for both the FWCF and GLRWOB events which included 8 different axial power distributions to determine the conservatism of using an axially averaged gap conductance. This study included both top-peaked and bottom-peaked axial power distributions, and demonstrated that an axially averaged gap conductance is conservative for use in hot bundle ΔCPR calculations.

QUESTION 8

Why is the core average gap conductance sensitive to the axial power shape, and the hot-bundle gap conductance insensitive to the axial power shape (p. 215, Note 4)?

RESPONSE 8

The core average gap conductance is somewhat sensitive to axial power distribution for the following reasons. Below approximately 6 kw/ft, the fuel pellet is modelled in ESCORE as elastic (i.e., no permanent deformation). Nodes having power densities above 6 kw/ft will experience permanent deformation which reduces the gap size and increases the gap conductance. Since a low value of gap conductance is conservative for the RETRAN system model, a "flat" axial power distribution minimizes this permanent deformation and, thus, the calculated gap conductance.

For the hot bundle, however, the axial average gap conductance is relatively insensitive to the axial power distribution. This conclusion was demonstrated by the results of a study involving 14 axial power distributions using the hot bundle gap conductance methods described in Appendix A of PL-NF-90-001. The hot bundle gap conductance methodology described in Appendix A of PL-NF-90-001 entails many axial nodes with power densities greater than 6 kw/ft. As a result, the higher power nodes experience more permanent deformation (and higher gap conductances) than the lower power nodes. An axial shape with some high power nodes (and, hence, high gap conductances) also implies some nodes with low power (and, hence low gap conductances). This fact produces a compensating effect on the axially averaged gap conductance used in the RETRAN hot bundle analysis which causes the relative insensitivity of hot bundle gap conductance to axial power distribution.

QUESTION 9

How are conservative core average axial power shapes and power histories determined in order to insure a conservative maximum gap conductance for the RFCF event?

RESPONSE 9

The limiting Recirculation Flow Controller Failure (RFCF) event is a slow (100 to 600 seconds) two pump flow runup as discussed in Section 3.3 of PL-NF-90-001. The slow nature of the event makes the calculated ΔCPRs relatively insensitive to gap conductance, since the power and heat flux are nearly in equilibrium. As stated in Appendix A of PL-NF-90-001, a conservatively high value of system model gap conductance is used for the RFCF analyses. Currently, a value of system model gap conductance is selected which is greater than or equal to the gap conductance corresponding to a core power of 125% of rated power.

QUESTION 10

In calculating the core average gap conductance, how many fuel-type/residence-time groups are used?

RESPONSE 10

The division of the core into fuel type/residence time groups for the purpose of core average gap conductance calculation is done as follows. A group is defined as a number of fuel bundles that either has a different fuel type (e.g., 9X9 vs. 8X8) or a significantly different power history from those of another group of bundles. Generally, all bundles of a certain type that are inserted during the same refueling outage are considered one group. For the sample calculations presented in PL-NF-90-001, the core was divided into 2 groups:

1. GE 8X8 twice burned fuel
2. ANF 9X9 once burned fuel

Current cycles would consist of at least 4 groups corresponding at EOC to once burned, twice burned, thrice burned, and fuel burned for four cycles.

QUESTION 11

The average power (AP) defined on the bottom of Page 209 is an energy rather than a power. How does this change in definition affect the analysis and the discussion of Sections A5 and A6?

RESPONSE 11

The parameter AP_i contained in the equation on page 209 of PL-NF-90-001 is, in fact, the average bundle energy (GWD/bundle) produced by that group of bundles during a given cycle. This parameter is used to determine the average power (MWt) of the bundles in that group to be used in the power history input to ESCORE. The average bundle power for a given batch is calculated by:

$$BP_i = \frac{AP_i}{CP} \times \frac{3293\ MWt}{764\ bundles}$$

where: CP = core average energy produced per bundle for the cycle of interest (GWD/bundle)

BP_i = average bundle power for batch i for the cycle of interest (MWt)

The analysis results and the discussions in Sections A.5 and A.6 are not affected.

QUESTION 12

Provide additional details describing how the three dimensional cycle-specific exposure and power distributions are used in determining the core average and hot bundle gap conductance using the fuel-type/residence-time groups.

RESPONSE 12

The Axial Power Distribution (APD) used for the system model ESCORE gap conductance calculations is selected to be conservative, and the APD used for the hot bundle ESCORE gap conductance calculations has little effect on the gap conductance, as discussed in the response to Question 8. Therefore, cycle specific SIMULATE-E calculated APDs are not used in the gap conductance calculations.

The power histories used for the hot bundle and system model gap conductance calculations are based on the SIMULATE-E calculated exposure distributions, as described in Section A.5 of PL-NF-90-001. The SIMULATE-E calculated cycle specific change in bundle average exposure for each fuel/residence time group is used to calculate the bundle average power for each fuel type/residence time group for the cycle being analyzed as discussed in the response to Question 11. These bundle average powers are used to produce the simplified fuel type/residence time group power histories used for the gap conductance calculations (see Appendix A of PL-NF-90-001).

QUESTION 13

How sensitive are the RETRAN GLRWOB and FWCF hot-bundle ΔCPR calculations to uncertainties in the assembly flow, and are the resulting variations in RCPR small compared to the observed ~ 0.05 RCPR variation relative to measurement?

RESPONSE 13

An analysis of the GLRWOB event for Unit 2 Cycle 5 (all 9X9 core) was performed to determine the effect of the bundle flow on calculated RCPR. A decrease in hot bundle flow rate of 2.6% resulted in a decrease in calculated RCPR of only .005. A similar analysis of a FWCF at 100% power/100% flow for Unit 2 Cycle 5 was performed to determine the effect of the bundle flow on calculated RCPR. A decrease in hot bundle flow rate of 2.4% resulted in an increase in calculated RCPR of only .001.

Thus, the effect of bundle flow on RCPR is an order of magnitude smaller than the RETRAN RCPR code uncertainty (based on comparison with measured data).

QUESTION 14

Does the FWCF ΔCPR calculation depend on the local power peaking factor and, if so, how is the correlation with the SL MCPR local power peaking uncertainty accounted for in the FWCF event?

RESPONSE 14

The FWCF ΔCPR is independent of the local peaking factor, because the internal form of the XN-3 correlation is limiting for this event and the ΔCPR for the internal form is independent of the local peaking factor. As a result, any correlation with the local peaking factor in the MCPR SL type calculation does not affect the calculated MCPR operating limit.

[illegible]

[illegible]

QUESTION 15

How is the bundle power in the GLRWOB and FWCF analyses selected so that it is bounding and independent of the bundle power used in the SL MCPR analysis?

RESPONSE 15

The hot bundle model used in the ΔCPR analyses does not represent an actual bundle in the core. Rather, it is a "pseudo-hot bundle" that will produce a ΔCPR that is expected to be greater than the actual hot bundles' ΔCPRs (e.g., a conservative gap conductance is used). An iterative procedure is used to select the initial hot bundle power that produces a minimum CHFR = 1.0 for the transient being analyzed. It should be noted that the hot bundle power iteration is performed assuming that the hot bundle flow is constant, which is consistent with the NRC approved methodology used by our fuel vendor. Thus, the RETRAN hot bundle power is unrelated to and independent of the power for the bundles analyzed in the MCPR safety limit type analyses.