CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9601260211 DOC.DATE: 96/01/22 NOTARIZED: NO DOCKET #
FACIL:50-387 Susquehanna Steam Electric Station, Unit 1, Pennsylva 05000387
50-388 Susquehanna Steam Electric Station, Unit 2, Pennsylva 05000388
AUTH.NAME AUTHOR AFFILIATION
BYRAM,R.G. Pennsylvania Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Responds to RAI re request request for approval of fuel TR PL-NF-90-001,Suppl 2, "Application of Reactor Analyses Methods for BWR Design & Analysis."

DISTRIBUTION CODE: A001D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 9
TITLE: OR Submittal: General Distribution

NOTES: 05000387

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD1-2 LA	1	1	PD1-2 PD	1	1
	POSLUSNY,C	1	1			
INTERNAL:	ACRS	6	6	FILE CENTER 01	1	1
	NRR/DE/EMCB	1	1	NRR/DRCH/HICB	1	1
	NRR/DSSA/SPLB	1	1	NRR/DSSA/SRXB	1	1
	NUDOCS-ABSTRACT	1	1	OGC/HDS2	1	0
EXTERNAL:	NOAC	1	1	NRC PDR	1	1
NOTES:		1	1			

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM OWFN 5D-5(EXT. 415-2083) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 19 ENCL 18

PP&L Pennsylvania Power & Light Company
Two North Ninth Street • Allentown, PA 18101-1179 • 610/774-5151

Robert G. Byram
Senior Vice President-Nuclear
610/774-7502
Fax: 610/774-5019

JAN 22 1996

U.S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Mail Stop P1 - 137
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
RELATED TO THE REQUEST FOR APPROVAL OF
FUEL TOPICAL REPORT TR PL-NF-90-001, SUPPLEMENT 2,
"APPLICATION OF REACTOR ANALYSES METHODS
FOR BWR DESIGN AND ANALYSIS" **Docket Nos. 50-387**
PLA-4404 FILES A17-2/R41-2 **and 50-388**

Reference: 1.) PLA-4348, R.G. Byram to U.S. NRC, "Request for Review of Topical Report PL-NF-90-001, Supplement 2" dated August 1, 1995

2.) U.S. NRC to R.G. Byram, "Request for Additional Information Related to the Request by Pennsylvania Power and Light Company for Approval of Fuel Topical Report TR PL-NF-90-001, Supplement 2, "Application of Reactor Analyses Methods for BWR Design & Analysis" (TAC Nos. M93267 and M93268), dated 12/5/95.

The purpose of this letter is to provide the Pennsylvania Power & Light (PP&L) Company response (as discussed via telecon) to the above referenced (Ref. 2) Request for Additional Information, necessary to complete the requested NRC review and approval (Ref. 1). The original report provided benchmarking results using the CASMO-3G lattice physics computer code along with describing modifications to PP&L's licensing analysis methods to use the Siemens Power Corporation (SPC) ANF-B critical power correlation. The attached response provides the clarifying information and specifically addresses each individual NRC question. Questions regarding this additional information should be directed to Mr. A. K. Maron at (610) 774-7727.

Very truly yours,

R. G. Byram

Attachment

A001 1/1

580045

copy: NRC Region I
Mr. C. Poslusny, Jr., NRC Sr. Project Manager - OWFN
Ms. M. Banerjee, NRC Sr. Resident Inspector - SSES
Mr. W. P. Dornsife, PA DEP

Question 1:

Provide more complete information on the range of fuels which have been validated by Pennsylvania Power & Light Company for use in CASMO-3G, and provide more information of the validation methodology to be used at Susquehanna for new fuel types.

Response:

The benchmark results provided in the PP&L supplement encompass GE 8x8, SPC 8x8, and SPC 9x9-2 assemblies. SPC's benchmarks in XN-NF-80-19 Supplement 3 (approved August 13, 1990) include 7x7, 8x8, and 9x9 assembly designs. Since that submittal, SPC has validated CASMO-3G for use in modeling the ATRIUM™-9B and ATRIUM™-10 assemblies.

When a new fuel design is being introduced in reload quantities at Susquehanna which has not been previously modeled, PP&L will validate the CASMO-3G model prior to use. Typically, this would involve comparison to vendor provided data or modeling techniques. For the instance of using ATRIUM™-10, SPC has already developed ATRIUM™-10 models using CASMO-3G and has benchmarked those models against Monte Carlo calculations. PP&L is using the same cross section library and similar modeling approaches to those used by SPC. The difference in modeling approaches between SPC and PP&L for lattice modeling have been investigated and result in a slight bias between the two methods. This modeling bias has been determined to be approximately 0.004 Δk at lattice exposures less than 40 GWD/MTU for current 9x9-2 assembly designs in use at Susquehanna. The causes of this bias are discussed in Section 2.3 of this supplement. Since PP&L will continue to use the same modeling approach as SPC for developing the inputs for the ATRIUM™-10, this bias will still be applicable for the newer assembly design. Therefore, the validation that SPC has performed for the ATRIUM™-10 is sufficient to allow modeling with CASMO-3G at PP&L. If a reload assembly design is being used at Susquehanna for which there is not a validated CASMO-3G model, PP&L would develop and validate a CASMO-3G model for that assembly design. This validation may include comparison to higher order calculations (such as Monte Carlo or other validated code results) or measured data from another reactor which has already loaded this fuel or data available from an LUA program. The validation might also be based on experience in modeling assemblies with similar design characteristics. For a fuel assembly used in an LUA program, reliance on fuel vendor calculations and data will be used extensively.

Question 2:

Describe the new fuel design validation process for the ANF-B correlation. Provide a complete list of fuels for which the correlation is currently valid.

Response:

Siemens Power Corporation performs validation of the ANF-B correlation for new fuel designs, as described in their NRC-approved "Generic Mechanical Design Criteria" document (Reference 1) and Reference 2. The basic form of the ANF-B correlation is not expected to change for future fuel designs. With the ANF-B correlation, new fuel design features are incorporated in the correlation by establishing "additive constants" determined in accordance with the NRC-approved procedure described in Reference 2. Boiling transition test programs form the basis for establishing these additive constants. PP&L will only use the ANF-B correlation to monitor fuel for which the correlation has been validated via this process.

Currently, sets of additive constants have been generated for the following fuel types:

SPC 8x8-2
SPC 9x9-2
SPC 9x9-5
SPC 9x9-9X
SPC 9x9-IX
SPC ATRIUM™-9B
SPC ATRIUM™-10A
SPC ATRIUM™-10

References:

1. "Generic Mechanical Deign Criteria for BWR Fuel Designs," ANF-89-98 (P) (A), Revision 1, May 1995.

2. "ANF-B Critical Power Correlation," ANF-1125 (P) (A), Supplements 1 and 2.

Question 3:

Have Supplements 3 and 4 of XN-NF-80-19 been previously reviewed and approved by the staff? If so, provide the date of approval.

Response:

XN-NF-80-19 Supplement 3 was reviewed and approved by the NRC staff via letter dated August 13, 1990 (Reference 1). Supplement 4 was provided to the NRC staff on March 16, 1990, and contains the responses to questions raised by the staff during the review of Supplement 3 (Reference 2).

References:

1. Letter, Ashok C. Thadani (NRC) to R. A. Copeland (SPC), "Acceptance for Referencing of Topical Report XN-NF-80-19(P), Volume 1, Supplement 3, 'Advanced Nuclear Fuels Methodology For Boiling Water Reactors; Benchmark Results for the CASMO-3G / MICROBURN-B Calculation Methodology," dated August 13, 1990.

2. Letter, R. A. Copeland (SPC) to R. C. Jones (NRC), "Responses to NRC questions on CASMO-3G / MICROBURN-B," dated March 16, 1990.

Question 4:

Provide more information on the extent to which CASMO-3G has been benchmarked against the Quad Cities gamma scan data.

Response:

PP&L did not reproduce the gamma scan benchmarks which were performed as part of the original PP&L CPM-2 methodology development. This was based on the similarity of results from the CASMO-3G / SIMULATE-E benchmarks and the original CPM-2 / SIMULATE-E benchmarks. These comparisons provide assurance that the CASMO-3G / SIMULATE-E model will accurately predict the Quad Cities bundle gamma scan data.

CASMO-3G pin powers have been benchmarked against the Quad Cities gamma scans by SPC in Section 4 of XN-NF-80-19 Supplement 3 (Reference 1). Additional benchmarks were performed by Studsvik (the code developer) and published in Nuclear Science and Engineering (Reference 2). Therefore, the gamma scan benchmarks were not repeated by PP&L.

References:

1. "Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G / MICROBURN-B Calculation Methodology," XN-NF-80-19(P)(A) Volume 1 Supplement 3, issue dated 11/27/90.

2. "Verification of the CASMO-3G / SIMULATE-3 Pin Power Accuracy by Comparison with Operating Boiling Water Reactor Measurements," Nuclear Science and Engineering: 114, 81-85 (1993).

Question 5:

Explain the reasons for extreme values of nodal RMS (larger than 10%).

Response:

Of the 226 TIP comparisons included in the submittal, only 10 sets (less than 5%) exhibit differences larger than 10%. Of the 10, six are in Unit 1 Cycle 3. The results from this cycle are somewhat atypical from the rest of the comparisons. For this cycle, the gadolinia depletion is slightly underpredicted causing an underprediction in reactivity in the lower portion of the core. This causes a corresponding underprediction of the power and exposure accumulation in the bottom portion of the core. As the reactor reaches the end of cycle, the underprediction in exposure accumulation in the bottom of the core through the middle of the cycle, causes an overestimate in reactivity in the bottom of the core. This leads to a corresponding overestimate in the power in the bottom of the core at the end of the cycle resulting in the larger than average TIP RMS error. Although the nodal TIP RMS error values are larger than average, the results are not indicative of a modeling deficiency.

Three of the remaining cases are at the end of Unit 1 Cycle 6. These differences are caused by a misprediction at the extreme bottom of the core (nodes 1 through 4) as the unit reaches end of cycle and enters coastdown operation. Larger errors under these conditions might be expected due to the fact that the SIMULATE-E model is developed and tuned to full power, full flow data. Although SIMULATE-E does well at off normal conditions, the calculation of power distribution and hence TIP RMS error will be slightly increased. For these three cases, the power in the bottom of the core was relatively low caused by end of cycle and coastdown conditions. Elimination of nodes 1 through 4 in the TIP comparisons for these exposure points reduces the TIP RMS by approximately 4% absolute. Since these are in lower powered nodes, the calculation of thermal limits are not adversely affected. (The TIP comparisons included in Supplement 2 include all 24 axial nodes.)

The last point is in Unit 2 Cycle 5. For this case, the measured TIP asymmetry is slightly greater than 6% indicating potential inaccuracies in the measurement.

These trends (as noted above) are also consistent with the CPM-2 calculated results. It should also be noted that for all of these cases the integral TIP response is very good (i.e., 3 % or less), providing an indication of the model's ability to calculate bundle power, which is the most important power distribution parameter for the CPR calculation.

Question 6:

Will other reload analyses be affected by these changes, particularly analysis of standby liquid control performance, shutdown margin and the ASME overpressurization event? How will reload-specific accident analyses including LOCA, rod-drop accident and fuel/equipment handling accidents be affected by the change to CASMO-3G?

Response:

The implementation of CASMO-3G as a replacement for CPM-2 will not affect the methods being used to analyze standby liquid control performance, shutdown margin or the ASME overpressurization event. CASMO-3G will be used as a replacement for CPM-2 to calculate cross section data for the SIMULATE-E and RETRAN calculations. The SIMULATE-E or RETRAN calculations will then be run to evaluate these events as defined Reference 1.

The LOCA analysis will be performed by the fuel vendor using their NRC-approved methods. The rod drop analysis is based on the SPC generic analysis which provides peak deposited enthalpy as a function of various neutronic inputs. The neutronic inputs (e.g., power distribution) would be calculated by PP&L using the CASMO-3G / SIMULATE-E model. The fuel and equipment handling accident will be performed by the fuel vendor.

References:

1. "Application of Reactor Analysis Methods for BWR Design and Analysis," PL-NF-90-001-A, July 1992, & Supplement 1, "Loss of Feedwater Heating Changes & Use of RETRAN Mod 5.1," September 1994.

Question 7:

Explain the relatively large difference between output of CASMO-CPM and actual data for the first Peach Bottom turbine trip test. This difference for the second and third turbine trip tests is considerably smaller.

Response:

The data available for the first Peach Bottom turbine trip test (TT1) is incomplete due to sensor problems experienced during the test. The turbine stop valve position was not available for TT1. The initiation of TT1 stop valve motion was estimated from the time difference between stop valve motion and turbine inlet pressure increase for TT2. Also, the stop valve closure time for TT1 was assumed to be the average of the TT2 and TT3 values. These assumptions have a significant effect on the calculated results.

The calculated dome and core pressures for TT1 are approximately 2 psi higher than the measured data. The dome and core pressures for TT2 and TT3 are slightly underpredicted. Small changes in pressure are significant for TT1 and TT2 since these events are near prompt critical due to the large increase in void reactivity that occurs before scram. In TT3, the scram occurs before void reactivity rises significantly. The calculation of a higher than measured initial pressure peak for TT1 may be due either to errors in the TSV timing data or some feature of the model which overpredicts the pressure transient for low power/high flow conditions. In any event, the overprediction of the pressure peak for TT1 results in a conservative overprediction of the core power.