BAW-10179-A, Rev. 1
February 1996

SAFETY CRITERIA AND METHODOLOGY FOR ACCEPTABLE CYCLE RELOAD ANALYSES

THE B&W OWNERS GROUP

Core Performance Committee

FRAMATOME COGEMA FUELS

FRAMATOME TECHNOLOGIES

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FRAMATOME COGEMA FUELS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 22, 1996

Mr. J.H. Willoughby, Chairman
B&W Owners Group
1700 Rockville Pike
Rockville, MD 20852

Subject: Acceptance of Revision 1 to Topical Report BAW-10179P, Safety Criteria and Methodology for Acceptable Cycle Reload Analyses, November 1995

Dear Mr. Willoughby:

By your letter of December 11, 1995, OG-95-1556, to the U.S. Nuclear Regulatory Commission (NRC), you submitted Revision 1 to Topical Report BAW-10179P, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses." This revision comprises the following references of five topical reports and one letter that have been approved by the NRC since the initial approval of BAW-10179P:

1) BAW-2149-A, "Evaluation of Replacement Rods in BWFC Fuel Assemblies," September 1993

2) BAW-10156-A, Rev. 1, "LYNXT Core Transient Thermal-Hydraulic Program," August 1993

3) BAW-10187P-A, "Statistical Core Design for B&W-Designed 177 FA Plants," March 1994

4) BAW-10184P-A, "GDTACO, Urania-Gadolinia Thermal Analysis Code," February 1995

5) BAW-10183P-A, "Fuel Rod Gas Pressure Criterion (FRGPC)," July 1995

6) Letter from Robert C. Jones to J.H. Taylor Concerning Fuel Rod Power History Uncertainty with TACO3, October 18, 1995

Since these references have already been approved and accepted versions for the topical reports have been published, the staff finds Revision 1 to BAW-10179P acceptable.

Sincerely yours,

Robert Jones, Chief
Reactor Systems Branch
Division of Systems Safety & Analysis

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 18, 1995

Mr. J. H. Taylor, Manager
Licensing Services
B&W Nuclear Technologies
3315 Old Forest Road
P. O. Box 10935
Lynchburg, VA 24506-0935

Dear Mr. Taylor:

In your letter dated July 19, 1995, you requested a revision of fuel rod power history uncertainty associated with the neutronics code used with the approved TACO3 fuel performance code. The request involves replacement of the old power history uncertainty calculated by the FLAME3 neutronics code with the new power history uncertainty calculated by the NEMO code for TACO3 reload applications. Our consultant PNL and the staff have reviewed your request. Based on previous approval of the NEMO neutronics code and its associated uncertainty for use in other reload applications, we conclude that your request is acceptable. Therefore, the power history uncertainty can be calculated by the NEMO rather than the FLAME3 neutronics code for future TACO3 reload applications. As agreed upon, this letter should be incorporated in the future revision of the approved overall reload methodology BAW-10179P-A ("Safety Criteria and Methodology for Acceptable Cycle Reload Analyses") that includes TACO3 and NEMO codes.

Sincerely yours,

Timothy E. Collins for

Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

SAFETY CRITERIA AND METHODOLOGY FOR ACCEPTABLE CYCLE RELOAD ANALYSES

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Appendix A

Additional NRC Approved Documents

The following NRC approved topical reports are being incorporated in BAW-10179P-A, Revision 1. These reports have received approval subsequent to the submittal of BAW-10179P-A and describe methodologies that replace or augment methodologies described in that report. Appendices B through F provide brief descriptions of each of the reports listed below.

1. BAW-2149-A, "Evaluation of Replacement Rods in BWFC Fuel Assemblies", B&W Fuel Company, Lynchburg, Virginia, September 1993.

2. BAW-10156-A, Rev. 1, "LYNXT Core Transient Thermal-Hydraulic Program", B&W Fuel Company, Lynchburg, Virginia, August 1993.

3. BAW-10187P-A, "Statistical Core Design for B&W-Designed 177 FA Plants", B&W Fuel Company, Lynchburg, Virginia, March 1994.

4. BAW-10184P-A, "GDTACO, Urania-Gadolinia Thermal Analysis Code", B&W Fuel Company, Lynchburg, Virginia, February 1995.

5. BAW-10183P-A, "Fuel Rod Gas Pressure Criterion (FRGPC)", B&W Fuel Company, Lynchburg, Virginia, July 1995.

6. Letter from Robert C. Jones to J. H. Taylor Concerning Fuel Rod Power History Uncertainty with TACO3, October 18, 1995.

Appendix B

Stainless Steel Replacement Rod Methodology - BAW-2149-A

The in-field repair of irradiated fuel assemblies with leaking rods involves the replacement of defective fuel rods with heat producing and/or non-heat producing rods. BAW-2149-A was prepared to justify the use of replacement rods without imposing unnecessary power peaking restrictions on the repaired fuel assemblies. This report addresses the nuclear, thermal-hydraulic, and mechanical aspects of the design that are affected by repair operations. The use of replacement rods for B&W Fuel Company (BWFC)-supplied fuel assemblies was determined to be acceptable by the NRC per the Safety Evaluation Report (SER) included in BAW-2149-A.

The conditions in the SER are as follows:

1. The approval is only applicable for reconstituted assemblies using up to ten solid Type 304 stainless rods.

2. Licensees referencing BAW-2149-A should observe the DNB and LOCA related limitations on radial flow and enthalpy changes resulting from power redistribution.

When using repaired fuel assemblies, DNB performance is evaluated on a cycle specific basis with the three-step procedure delineated below.

Step A. Determine the peaking increase, on a pin-by-pin basis, for the fuel assembly locations containing cold replacement rods and adjacent fuel assembly locations.

Step B. For limiting power distributions, verify that no peak exceeds the limiting hot fuel rod peak. In the event an affected fuel rod becomes the limiting hot fuel rod in the core, preservation of the maximum $F_{\Delta H}$ limit will be verified.

Step C. For situations where the fuel rod peaking increase is greater than 5 percent or the increased fuel rod peaks are within 2 percent of the limiting hot fuel rod peak, additional detailed thermal-hydraulic evaluations shall be performed to assure the minimum DNBR prediction for the affected fuel rods is bounded by the limiting hot fuel rod in the core and that any impact on the limiting hot fuel rod is determined.

These measures ensure that there are no secondary consequences that might affect the minimum DNBR prediction for fuel rods that are not immediately adjacent to the cold replacement rods.

The impact of the stainless steel replacement rods on the LOCA evaluation will be considered on a cycle-specific basis. Any impact on the maximum allowable linear heat rate limits generated in the LOCA analysis is determined by this evaluation.

The replacement of as many as ten fuel rods within a single fuel assembly is acceptable. When a stainless steel replacement rod is surrounded by heated rods or is on the periphery of an assembly, there is no penalty on the calculated CHF, since these configurations are explicitly included in the CHF data base. When one or more stainless steel replacement rods are placed next to guide tubes or instrument tubes, or are adjacent to each other, the new configuration is bounded, in terms of thermal-hydraulic performance (DNBR), by the original configuration without stainless steel replacement rods. Therefore, the BWFC CHF correlations and analysis methods described in Chapter 6 of BAW-10179P-A are applicable to the analysis of fuel assemblies in which up to 10 fuel rods per assembly have been replaced by cold replacement rods. From a corewide perspective, DNBR performance in a core with stainless steel replacement rods will be ensured by compliance to the $F_{\Delta H}$ design limit.

The introduction of replacement rods affects the power peaking in adjacent fuel assemblies as well as the assembly with the replaced rods. The changes are generally small in nature, but are considered in the power distribution analysis which is described in section 5.2 of BAW-10179P-A.

The stainless steel replacement rods weigh slightly less than Zircaloy-clad fuel rods, but the effect on fuel assembly weight of up to 10 replacement rods is negligible. Therefore, the use of stainless steel replacement rods has an insignificant effect on fuel assembly hydraulic lift.

Stainless steel replacement rods are designed and analyzed to ensure that there is no adverse impact on fuel assembly performance. The rods are designed to ensure that adequate performance with respect to differential thermal expansion, irradiation growth, seismic-LOCA response, grid relaxation, and fretting due to vibration will be maintained.

Appendix C

LYNXT Thermal-Hydraulics Code - BAW-10156-A, Revision 1

Revision 1 of BAW-10156-A incorporates the Pressure-Velocity Implicit Numerical Solution (PV) algorithm. The PV algorithm provides a supplemental solution technique that can be used as an alternative to the original (COBRAIV-1) implicit algorithm. This solution technique is very useful for the analysis of transients that are characterized by low coolant flow rates. Limitations associated with the use of the PV algorithm are as follows:

1. The application of the LYNXT, Rev. 1, PV algorithm is restricted to the following ranges:

   Mass flux (absolute value) - 0.0 to 3.0 x 10^6 lbm/(hr-ft^2),
   System pressure - 500 to 3000 psia,
   Local heat flux - 0.0 to 0.8 x 10^6 Btu/(hr-ft^2).

   It is the responsibility of the licensee to verify that the proper algorithm and algorithm input parameters have been selected for the analyses performed within these ranges.

2. When the LYNXT, Rev. 1, incorporating the PV algorithm, is used, the licensee is responsible for verifying the adequacy of the crossflow resistance whenever reverse and recirculating flows are observed in the analysis.

3. The B&W-2, BWC, BWCMV, and W3 CHF correlations may be used with the COBRA-IV-I implicit or PV solution algorithm. The application of each correlation is restricted to its range of applicability. Application of LYNXT, Rev. 1 to another CHF correlation (other than B&W-2, BWC, BWCMV, and W3) not developed with either of LYNXT, Rev. 1's flow solution algorithms will require a separate validation process.

Since the changes to LYNXT provide additional capabilities that can be used as an alternative to the original solution methods, all of the discussions regarding LYNXT in chapter 6 of BAW-10179P-A remain valid.

Appendix D

Statistical Core Design for B&W-Designed 177FA Plants - BAW-10187P-A

The design philosophy for core departure from nucleate boiling (DNB) protection described in section 6 of BAW-10179P-A follows a deterministic approach where uncertainties that affect the minimum DNB ratio (DNBR) are simultaneously assumed to be at their worst-case values. The minimum core DNBR is calculated using compounding of the uncertainties, and compared to the DNBR design limit associated with the applicable critical heat flux correlation.

A more realistic assessment of core DNB protection, called Statistical Core Design (SCD), has been developed by application of statistical techniques to treat the core state and bundle uncertainties. SCD is a widely accepted method that is utilized to reduce some of the undue conservatism of traditional methods, while still allowing for the traditional compounding of variables not amenable to statistical treatment.

BAW-10187P-A describes the application of SCD methodology to the analysis of B&W 177 fuel assembly cores operating with Mark-B fuel. A response surface model was used to obtain an overall uncertainty on the calculated DNBR. The response surface model was based on a full central composite design method in order to reduce uncertainty in the response surface model fit. The uncertainty distribution for each of the applicable variables was subjected to a Monte Carlo propagation analysis to determine an overall statistical DNBR uncertainty, which was used to establish a Statistical Design Limit (SDL). The SDL is higher than the CHF limit upon which it is based, because it contains allowances for all of the propagated uncertainties as well as the uncertainty on the original CHF correlation. When the minimum DNBR is calculated, the variables treated statistically are entered into the LYNXT thermal-hydraulic calculations at their nominal levels. Variables not treated in deriving the SDL continue to be entered at their most adverse allowable level.

Generic uncertainty allowances included in the SCD methodology for the 177-FA plants are described in BAW-10187P-A. Plant-specific verification of these allowances or determination of new allowances to be used with this method is performed for each application.

The SDL, defined in BAW-10187P-A, provides 95 percent protection at a 95 percent confidence level against hot pin DNB. The corresponding corewide protection on a pin-by-pin basis using real peaking distributions is greater than 99.9 percent. Thus, adequate core DNB protection is assured and quantified.

The SDL approved for application of BAW-10187P-A is 1.313. This value is based on Appendix F of the topical report. Appendix F of BAW-10187P-A describes determination of an SDL that is conservative for all axial power shapes, including axial power distributions that cause the minimum DNBR to be located at or near the core exit (core exit-limited cases). BWFC determined that a hot pin SDL of 1.313 bounds all cases, including core exit-limited cases, and provides a limiting hot pin 95 percent protection at a 95 percent confidence level against DNB.

The thermal design Limit (TDL) defined in Appendix F of BAW-10187P-A represents a retained DNB margin. The retained margin is available to offset penalties, such as transition core effects, or deviations in uncertainty values from those incorporated in the SDL, or to provide flexibility in the fuel cycle design.

Application of BAW-10187P-A in licensing evaluations with a statistical design limit of 1.313 is subject to the following limitations, which are stated in the NRC's safety evaluation report for the topical report:

1. The component uncertainties and their distributions must be reviewed on a plant-specific basis to determine their applicability.

2. The bounding assembly-wise power distribution assumed in the core-wide SDL calculation must be shown to bound the expected operating power distributions on a cycle-specific basis.

3. All core state variables that were not included in the statistical design must continue to be input to thermal-hydraulic computer codes at their most adverse allowable values rather than at their nominal values. This applies specifically to the axial peaking factor and the location of the axial peak, since these parameters were not included in the determination of the response surface model.

4. The response surface model will be validated and revised (as necessary) when applied to new fuel assembly designs and extended operating conditions, and with new computer codes and DNB correlations. The currently approved codes are LYNXT, LYNX1, and LYNX2, and the currently approved correlation is the BWC DNB correlation.

Section 6.6 of BAW-10179P-A describes the development and generation of Maximum Allowable Peaking (MAP) limits. MAP limits provide linkage between the DNBR analyses, which use design peaking distributions, and the core power distribution analysis described in section 5.2 of BAW-10179P-A. The MAP limits are used in DNB peaking margin calculations that determine the core protective and operating limits. SCD-based MAP limits are calculated with the methodology described, however, their generation is based on equivalence to the TDL (i.e., the SDL plus retained margin) instead of equivalence to the base CHF correlation limit. The calculation of and application of SCD-based MAP limits in licensing evaluations will remain as described in BAW-10179P-A. When DNB peaking margins are calculated, specific allowances will continue to be made for those factors not included in the SDL/TDL limit.

Appendix E

GDTACO: Gadolinia Fuel Rod Thermal Analysis Code - BAW-10184P-A

The approved GDTACO code is a modification of the approved TACO3 code that incorporates the physical material properties of gadolinium oxide (gadolinia) for calculating the thermal performance of urania-gadolinia fuel rods. The GDTACO code, with its gadolinia material property database, is used to calculate fuel melting, fuel rod internal gas pressure, cladding strain, cladding creep collapse code initialization, and loss-of-coolant accident (LOCA) analyses initialization parameters. The creep collapse code initialization calculations include predictions of rod internal gas pressure and cladding temperatures. The LOCA initialization calculations include predictions of local volumetric fuel temperature as a function of linear heat rate, fuel rod internal gas pressure, gas composition, and fuel rod dimensions and characteristics.

GDTACO includes models for pellet-to-cladding gap conductance, fuel densification and swelling, cladding creep and deformation, gap closure, and fission gas release calculations for individual fuel rods using urania-gadolinia fuel pellets as an integral absorber. The methodology employed with GDTACO is identical to that approved for TACO3 for the analyses discussed in chapter 4 of BAW-10179P-A.

The conditions of the SER are:

1. Approval applies to gadolinia concentrations up to 8 wt% nominal (8.3 wt% maximum pellet).

2. The approval applies to the GDTACO code and methodology application to BWFC urania-gadolinia fuel rod designs and is not limited to specific bundle types.

3. Both the urania only and the urania-gadolinia rods must be analyzed on a cycle-specific basis during each licensing application involving both fuel types.

Gadolinia LOCA issues require special attention. To avoid cycle-specific LOCA analyses of urania-gadolinia rods in support of reload applications, the LOCA initialization was performed for a limiting configuration. Features of this configuration are as follows:

1. Eight-weight percent gadolinia fuel was chosen since thermal conductivity decreases with increasing gadolinia concentration.

2. The analysis was performed at the 2-ft elevation since this elevation provides conservative results for linear heat rate, metal-water reaction, and peak cladding temperature when compared to the higher elevations.

3. A composite case was analyzed using EOL gadolinia fuel temperatures, MOL oxide thicknesses, and a rod internal pressure equal to system pressure.

4. The peak linear heat rate (LHR) for the gadolinia fuel was set to eighty-five percent of the 2-ft UO_2 LOCA LHR limit, although the 8 wt% gadolinia fuel is not expected to reach that fraction of UO_2 power.

For new fuel designs, the allowable gadolinia LOCA LHR limits will be evaluated using a similar approach. Limits on gadolinia rod power levels will be established such that these rods are bounded by UO_2 rod LOCA calculations.

Appendix F

Fuel Rod Gas Pressure Criterion (FRGPC) - BAW-10183P-A

The predicted fuel rod internal gas pressure can exceed the nominal reactor coolant system (RCS) pressure, according to the conditions established in BAW-10183P-A. The previous criterion, limiting internal pressures to less than the RCS pressure, had been chosen as a convenient and conservative basis for ensuring that the cladding mechanical integrity and subchannel flow characteristics were maintained within acceptable limits during operation. This limitation results in burnup restrictions for nuclear fuel.

The impetus for extending the burnup of nuclear fuel and, consequently, increasing the fuel rod internal gas pressure, led to the development and adoption of the following criterion:

"The internal pressure of the peak fuel rod in the reactor will be limited to a value below that which would cause (1) the fuel-clad gap to increase due to outward cladding creep during steady-state operation and (2) extensive DNB propagation to occur."

Item (1) above refers to the phenomenon of cladding liftoff. It is judged to occur when the cladding creep rate exceeds the fuel pellet diametral strain rate due to swelling. Cladding liftoff may occur at low linear heat generation rates but the consequences in terms of gap size increase and fuel temperature over time frames of interest are negligible and inconsequential. The linear heat generation rate below which cladding liftoff will be discounted is 3 kW/ft. For linear heat generation rates above 3 kW/ft, the fuel rod internal pressure will be limited to a proprietary value above the RCS pressure or the pressure required to achieve cladding liftoff, whichever is smaller.

Item (2), which relates to DNB propagation, is assured by BWFC's core protection evaluation which was performed by the response surface method. That evaluation determined that 99.99% of the fuel rods will not experience fuel failure when the failure probabilities due to DNB and pressure above system pressure are combined. If the critical input parameters, as defined in Tables 4.11 and 4.12 of BAW-10183P-A, or any of the other crucial assumptions or statistical distributions that form the input for the response surface application should change, then the core protection evaluation should be re-performed using the methods defined in BAW-10183P-A.

The above criterion supersedes the internal pressure criterion in section 4.2.8.1 of BAW-10179P. BAW-10183P is an alternative methodology for Analysis Method 4.2.8.2 when the fuel rod internal pressure exceeds reactor coolant system pressure.

Appendix G

Letter from Robert C. Jones Concerning Fuel Rod Power History Uncertainty with TACO3

Prior to 1993, neutronics calculations at BWFC were performed with the PDQ and FLAME codes. In 1993 the NRC approved the NEMO code for neutronics calculations and NEMO has replaced both codes listed previously. The nuclear uncertainty previously approved for PDQ and FLAME is 7.5%. The uncertainty approved for use with the NEMO code is 4.8%. On July 19, 1995 BWFC requested, via letter, permission to use the lower value for power history applications with TACO3. The larger value is given in the safety evaluation report (SER) for TACO3. The NRC approved the use of the NEMO code uncertainty in TACO3 fuel performance calculations in reference 6 of Appendix A.