Babcock & Wilcox
a McDermott company

Nuclear Power Division

3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506-0935
(804) 385-2000

August 13, 1985

Mr. Cecil O. Thomas, Chief
Standardization and Special
Products Branch
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Request Number Two for Additional Information on BAW-10156

Dear Mr. Thomas:

The attachment provides B&W's response to your request for additional information to support your review of LYNXT. This response supports the use of LYNXT for transient analyses with DNBR limits derived from steady-state CHF test data. It is our belief that this application is reasonable and proper, therefore we are requesting approval of LYNXT for both steady-state and transient applications.

Should you have any questions concerning this letter, please call C. F. McPhatter at (804) 385-2401.

Sincerely,

J.H. Taylor
Manager, Licensing Services

GAM/lec

c: Y.H. Hsii - NRC
R.B. Borsum - B&W Bethesda
T.L. Baldwin
D.H. Moran - NRC

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Question

The staff evaluation of the LYNXT code has concluded that calculational models in LYNXT are acceptable for both steady-state and transient analyses of the core thermal hydraulics. However, since both BAW-2 (B&W-2) and BWC critical heat flux correlations and their respective DNBR limits were developed with steady-state CHF test data and steady-state thermal hydraulic codes, it is questionable as to whether these DNBR limits in connection with the LYNXT transient solution provide the required protection against departure from nucleate boiling at a 95/95 probability/confidence level. Therefore, until such proof can be made of the adequacy of the steady-state DNBR limits using the transient LYNXT solution, we find that only the steady-state solution of LYNXT is acceptable for DNBR analysis. For these transients which are DNBR limited, a quasi-steady state approach can be used in which the reactor coolant system analysis provides the time dependent boundary conditions to the core and the steady-state solution of LYNXT is used for the core thermal-hydraulic and DNBR calculation. Since you have expressed in a meeting with the staff that B&W intends to use the transient solution of LYNXT for the analyses of some transients which are not DNBR limited, please provide a list of these transients and your justifications as to why these analyses are acceptable.

Response

A) DNB - Limited Transients

The adequacy of steady-state CHF correlations for analyzing transients has previously been demonstrated by B&W during the development of the B&W-2 CHF correlation, Reference 1. As part of the B7 and B10 rod bundle tests, power, flow, and pressure transients were experimentally simulated. For the B7 bundle tests the following classes of transient tests were performed, at various initial conditions:

1) Power ramp of 5% (2% below steady-state CHF to 3% above) which simulates a single control rod ejection, and

2) Flow reductions which simulate one or all four reactor coolant pumps coasting down.

For the B10 bundle tests the following transient tests were run, at various inlet conditions:

1) Power ramps of a single 5% ramp (2% below steady-state CHF to 3% above) and a dual ramp totaling 30% (a ramp from 27% below steady-state CHF to 2% below, followed by another ramp from 2% below to 3% above),

2) Flow coastdowns simulating one or all four reactor coolant pumps coasting down, and

3) Pressure reduction transients where the pressure decreased approximately 20 psi/second.

The results of the B7 and B10 rod bundle transient tests (contained in References 2 and 3, respectively) indicated the following:

1) No premature CHF occurred as a result of the power, flow, and pressure transients.

2) The use of instantaneous system pressure, local flow, and quality in a steady-state CHF correlation provided acceptable, or conservative results.

LeTourneau and Green (Reference 4) used a rod bundle test section to experimentally simulate constant power flow coastdown transients. Their study showed that the use of steady-state CHF correlations and instantaneous local conditions provided conservative transient CHF predictions. The transient CHF rod bundle tests performed by B&W and LeTourneau and Green support the use of instantaneous local conditions and steady-state CHF correlations for predicting transient CHF (This same statement was used in a response to an NRC question on the use of B&W-2 in transients, Reference 5, submitted in support of the licensing of B&W-2).

The B&W conclusions presented in reference 5 are also supported by other investigators. Leung, in Reference 6 states that there are two primary problems in the analytical prediction of transient CHF:

1) Capability and accuracy of the computer codes used to predict the local conditions of flow, quality, and system pressure, and

2) Use of steady-state CHF correlations.

The LYNXT calculational models have been shown to be acceptable for both steady-state and transient analyses, which addresses Leung's first concern. In regard to the second concern, Leung investigated the domestic and foreign work on CHF for flow decays, power excursions, and depressurization (blowdown) transients. He concluded that steady-state CHF correlations and transient local conditions can be used to predict CHF for flow and power transients. Consistent with this conclusion, the NRC approved the use of B&W-2, in addition to other steady-state CHF correlations, for LOCA calculations, in Reference 7.

A number of other investigators have also concluded that steady-state CHF correlations are acceptable for transient CHF calculations. Cermak, et.al., Reference 8, used a rod bundle test section to justify the use of steady-state CHF correlations during pressure blowdown. A wide range of initial conditions were used in Cermak's transient tests. Snider, Reference 9, used COBRAIV-1 (the basis for LYNXT) in conjunction with several steady-state CHF correlations to examine transient CHF in the Semiscale facility during a LOCA. The comparisons of COBRAIV-1 and the experimental CHF indicated that B&W-2 was acceptable for blowdown calculations. Leung and Gallivan, Reference 10, examined the behavior of several steady-state CHF correlations during flow coastdown, rapid and exponential flow decay, combined flow/pressure transients, and a blowdown experiment with flow reversal. They concluded that instantaneous local conditions in conjunction with steady-state CHF correlations are able to predict transient CHF. Yuelys-Miksis and Shier, Reference 11, analyzed some Columbia University flow decay transients using a steady-state CHF correlation and CORBAIIIC, which is similar to LYNXT. The analyses indicated that the steady-state CHF correlation used in conjunction with the COBRAIIIC instantaneous local conditions accurately predicted transient CHF.

References 2 through 11 all show that steady-state CHF correlations are applicable to a wide range of flow, power, and pressure transients. In each of these references the conditions were based on experimental data. The accuracy, or conservatism, shown in these references indicates that the steady-state CHF correlation limits are also applicable to transients.

In conclusion, the use of steady-state CHF correlation and their associated DNBR limits for transient application is appropriate when instantaneous local conditions, as calculated by LYNXT, are used.

B) Non-DNB - Limited Transients

LYNXT's calculational models are appropriate for the analysis of core and fuel response to transients over a broad range of conditions, including those not limited by DNB criteria. Examples of such transients are the Locked Rotor Accident, for which the primary concern is peak clad temperature, and the Control Rod Assembly (CRA) Ejection accident, for which the maximum fuel enthalpy is evaluated. For the CRA Ejection, the use of LYNXT is documented in References 12 and 13. Other applications will be documented on a case-by-case basis.

References

1. Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May 1976, Babcock & Wilcox, Lynchburg, Virginia.

2. L.A. Zielke, R. H. Wilson, Transient Critical Heat Flux and Spacer Grid Studies, Nuclear Technology, 24, October 1974, pp. 13-19.

3. Critical Heat Flux Limits for CNSG Conditions-Final Report, BAW-1405, July 1973, Babcock & Wilcox, Lynchburg, Virginia.

4. B. W. LeTourneau, S. J. Green, Critical Heat Flux and Pressure Drop Tests with Parallel Upflow of High Pressure Water in Bundles of Twenty ½in. Rods, Nuclear Science and Engineering, 43, pp. 90-104.

5. Correlation of Critical Heat Flux In a Bundle Cooled by Pressurized Water, Supplement 1, BAW-10000, Supp. 1, March 1971, Babcock & Wilcox, Lynchburg, Virginia.

6. J. C. M. Leung, Critical Heat Flux Under Transient Conditions: A Literature Survey, NUREG/CR-0056, (ANL-78-39), June 1978, Argonne National Laboratory, Argonne, Illinois.

7. Part 50 Appendix K-ECCS Evaluation Models, Chapter 1-Nuclear Regulatory Commission, Code of Federal Regulations Title 10-Energy, January 1, 1983, Office of the Federal Register.

8. J. O. Cermak, et.al., The Departure From Nucleate Boiling In Rod Bundles During Pressure Blowdown, 70-HT-12, American Society of Mechanical Engineering, New York, New York.

9. D. M. Snider, Analysis of the Thermal-Hydraulic Behavior Resulting in Early Critical Heat Flux and Evaluation of CHF Correlations for the Semiscale Core, TREE-NUREG-1013, March 1977, EG&G Idaho, Idaho Falls, Idaho.

10. J. C. M. Leung, K. A. Gallivan, Prediction of Critical Heat Flux During Transients, Proceedings of the American Nuclear Society/European Nuclear Society Topical Meeting on Thermal Reactor Safety/Volume (CONF-800403/V-II), April 6-9, 1980, Knoxville, Tennesse, pp. 1229-1239.

11. C. Yuelys - Miksis, W. G. Shier, COBRA-IIIC Analysis of the Columbia University Critical Heat Flux Experiments During Flow Decay Transients, BNL-NUREG-30881, December 1981, Brookhaven National Laboratory, Upton, New York.

12. W. M. Herwig, Control Rod Assembly Ejection - Analysis of the CRA Ejection Accident in B&W Pressurized Water Reactors, BAW-10150, January 1982, Babcock & Wilcox, Lynchburg, Virginia.

13. Letter, C. O. Thomas (NRC) to J. H. Taylor (B&W), Acceptance for Referencing of Licensing Topical Report BAW-10150 (NP), "Control Rod Assembly Ejection - Analysis of the CRA Ejection Accident in B&W Pressurized Water Reactors, January, 1982," August 29, 1983.

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