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Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Richard W. Borchardt, Director
Standardization Project Directorate

**Subject: NRC Requests for Additional Information (RAIs) on the Simplified Boiling Water Reactor (SBWR) Design**

Reference: Transmittal of Requests for Additional Information (RAIs) Regarding the SBWR Design, Letter from M. Malloy to P. W. Marriott dated April 8, 1994.

The Reference letter requested additional information regarding SBWR fuel bundle testing in the ATLAS facility. In fulfillment of this request, GE is submitting Attachment 1 to this letter which transmits the responses to RAIs 900.63 and 900.64.

Sincerely,

P. W. Marriott, Manager
Advanced Plant Technologies
M/C 781, (408) 925-6948

Attachment 1, "Responses to NRC RAIs"

cc: P. A. Boehnert (w/1 copy of Attachment)
M. Malloy (w/2 copies of Attachment)
F. W. Hasselberg (w/1 copy of Attachment)

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**Fuel Performance Testing**

RAI Number: 900.63

Question:

In a letter dated October 8, 1993 (MFN No. 164-93), GE Nuclear Energy (GE) responded to the staff's request for additional information (RAI) Q900.1 regarding fuel performance testing. Contrary to the position stated in Q900.1 that the staff considers the ATLAS tests to be part of the testing program for SBWR design certification, GE's response asserts that this testing is not part of the certification testing program.

The staff reaffirms that it considers the ATLAS tests to be part of the testing program required by 10 CFR 50.47(b)(2) for SBWR design certification and requires GE to revise its documentation on the SBWR testing program to include the ATLAS tests.

GE Response:

10CFR50.47(b) requires that :

The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof; and that sufficient data exist on the safety features to assess analytical tools used for safety analysis. GE believes that there is a sufficient data and analytical basis to predict critical power for the SBWR fuel bundle within ±5%. This assertion is substantiated in response to the next RAI (900.64). This value of the uncertainty will be used in the MCPR analysis for SBWR transients and accidents.

SBWR-unique fuel performance testing will be performed as part of the detailed fuel design process, and will be performed after SBWR fuel is ordered. This should allow the uncertainty in MCPR prediction to be reduced to approximately 3%, consistent with GEXL predictions of other fuel designs.

RAI Number: 900.64

Question:

GE's response to RAI Q900.1 (MFN No. 164-93, October 8, 1993) does not include the details necessary for the staff to perform an evaluation. Please provide, in much greater detail, the following information:

a. Detailed information to support GE's conclusion that the code and the model are applicable over the range of geometric (e.g., fuel length, pitch, diameter, spacer configuration) and the thermal-hydraulic (e.g., mass flow rate, power profile) parameters representative of the SBWR fuel design.

b. Provide details of the planned testing, including the test specification, test matrix, and test schedule.

c. Provide details of the analyses planned in conjunction with the fuel performance test program, including documentation for the thermal-hydraulic correlations to be validated for use in SBWR safety analyses.

d. Provide verification that the ATLAS facility is able to match approximately the appropriate thermal-hydraulic conditions for SBWR fuel. In particular, how will ATLAS simulate the natural circulation flow behavior that will exist in the SBWR?

GE Response: (a, b, c)

The SBWR fuel bundle is of the GE 8X8E and/or GE 8X8EB design. The GE8 design is an 8x8 lattice with two water rods. This design has been licensed by the NRC (Ref. 900.64-1) and has been operated extensively in operating reactors (>1 million rods). The SBWR fuel will have identical fuel rods, but of a shorter length (9 ft vs 12 ft fuel column). The fuel bundles will have identical lower tieplate, spacers and upper tieplate . Thus, the lattice configuration and local pressure drop components are identical. The only factors that need to be considered are the difference in length and the range of flow parameters during normal operation.

The bundle critical power is predicted using the GEXL02 correlation. The GEXL02 correlation has a data base of 1516 points. The critical power is predicted with a mean value of 0.9998 and a standard deviation of 2.68%. The range of the data base is :

Mass Flux : 0.1 to $1.50 \times 10^6$ lb/ft$^2$-hr (130-1950 Kg/m$^2$-s),
Pressure: 800-1400 psi (56- 84 bar),
Inlet Subcooling: 0 to 100°F (0-55 K),
Axial power profiles: 888 points cosine and 628 inlet-peaked,
Local (rod-to-rod) peaking : various; uniform to 1.6 corner to 1.4 interior.

These data are for a 12 ft heated length.

The SBWR operating conditions of

> mass flux: 0.48 -0.63 x$10^6$ lb/ft$^2$-hr (650 -850 Kg/m$^2$-s),
> pressure: 1040 psi ( 71.7 bar),
> subcooling: 30°-47°F (17-26 K)

are well within this data base. The only remaining issue is the effect of length. This is addressed next.

First, it should be recognized that the boiling length type of correlation is well suited to predict the effects of length and axial power shape. This was one of the main reasons it was chosen to replace the local conditions based CHF correlation. Figure 900.64-1 shows data from various 6 ft and 12 ft lengths correlated by the boiling length correlation. The GEXL01 correlation (which preceded GEXL02 and is of similar form) included in its data base 84 data points with a 16 rod bundle with a 6 foot heated length.

Second, it is well known that the plot of critical power vs. length flattens out with increasing length. Figure 900.64-2 shows data taken at Columbia University for various lengths. At SBWR mass fluxes, the difference in critical power between a 9 ft long bundle and a 12 ft long bundle is less than 10%. The error in predicting this effect is clearly a small fraction of this absolute difference.

Third, GEXL02 has been compared against predictions from a mechanistically based, accurate subchannel analysis code (COBRAG) for the SBWR bundle. COBRAG is a multi-field two-phase flow model. The vapor, liquid film(s) and droplets in a subchannel are treated as separate fields. Boiling transition is calculated mechanistically based on liquid film dryout. Excellent comparisons have been obtained with a wide range of critical power data in full scale rod bundles. Figure 900.64-3 shows predictions of data from 8x8, 9x9, and 10x 10 bundles. Figure 900.64-4 shows predictions for different axial power shapes. Figure 900.64-5 shows a range of spacer types. Figure 900.64-6 shows a range of mass fluxes. Almost all data are predicted within 5%, with a standard deviation of less than 3%. COBRA predictions were also made of the Columbia data with a 6 ft heated length, with excellent results (see Figure 900.64- 7). Thus, COBRA provides a basic and completely independent means of predicting BWR bundle critical power. In Figure 900.64-8, critical power calculated by GEXL02 for a range of SBWR conditions for the SBWR bundle is compared with critical power calculated by COBRAG. The standard deviation of the difference between the two independent methods is less than 3%. When this is combined with the standard deviation of the error in COBRAG predictions (< 3%), an RMS value of 4.2% is obtained. GE proposes to use an uncertainty value (1 sigma) of 5% for the prediction of critical power in the SBWR bundle.

GE Response: (d)

The ATLAS test facility can be configured to test a shorter length bundle. This would be done using the same setup which exists at present.

The difference between forced and natural circulation conditions, or 'hard' vs. 'soft' inlet conditions, is important when considering thermal-hydraulic stability. If the stability threshold for the bundle were to be reached before critical heat flux in the SBWR, the ATLAS results would be nonconservative, as the ATLAS tests are performed with 'hard' or forced flow inlet conditions. In actuality, the critical power threshold is reached well before the stability threshold and the ATLAS type of measurements are valid. This is demonstrated below.

The critical power for the SBWR bundle at rated conditions is of the order of 5.7 MW. At this power level, calculations were performed with the stability code (FABLE). The decay ratio is in the range of 0.16 to 0.36 based on axial peaking (nominal vs. design basis shape). The core decay ratio at the scram power level is less than 0.5. Thus, there is a large margin to the onset of either channel or corewide (and therefore, regional) oscillation. These large margins to instability have also been independently confirmed by calculations made by J. March-Leuba at ORNL.

Thus, the use of 'hard' inlet conditions for determination of critical power is justified.

*References:*

900.64-1 Letter, MFN-082-85, C. O. Thomas (NRC) to J. S. Charnley, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-PA-6, Amendment 10, 'General Electric Standard Application for Reactor Fuel'," May 28, 1985.

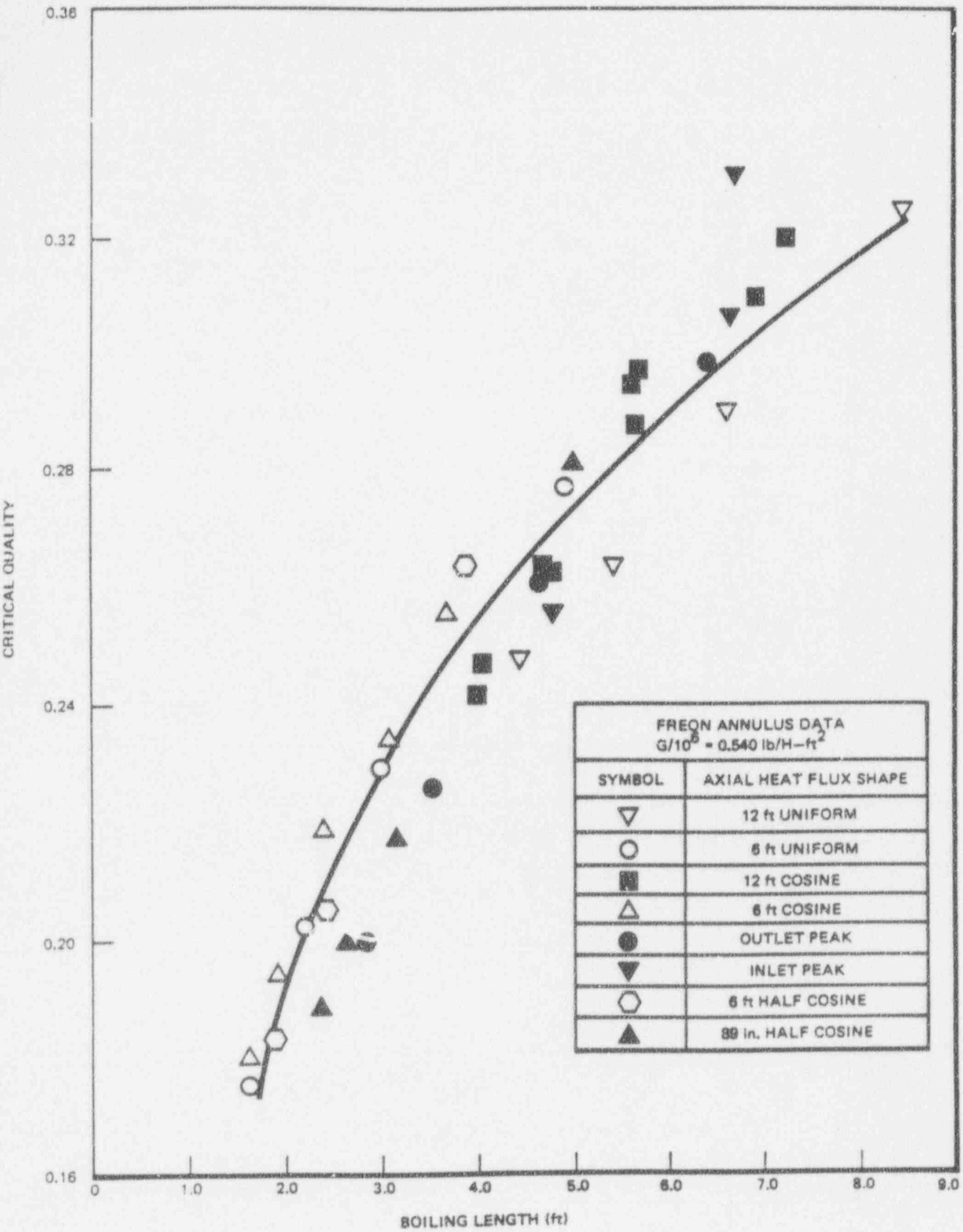



Figure 900.64-1 *Critical Quality vs. Boiling Length, Freon-114 Annulus Data*

# Critical Power for 9 ft. channel

*Critical Power vs. Length*

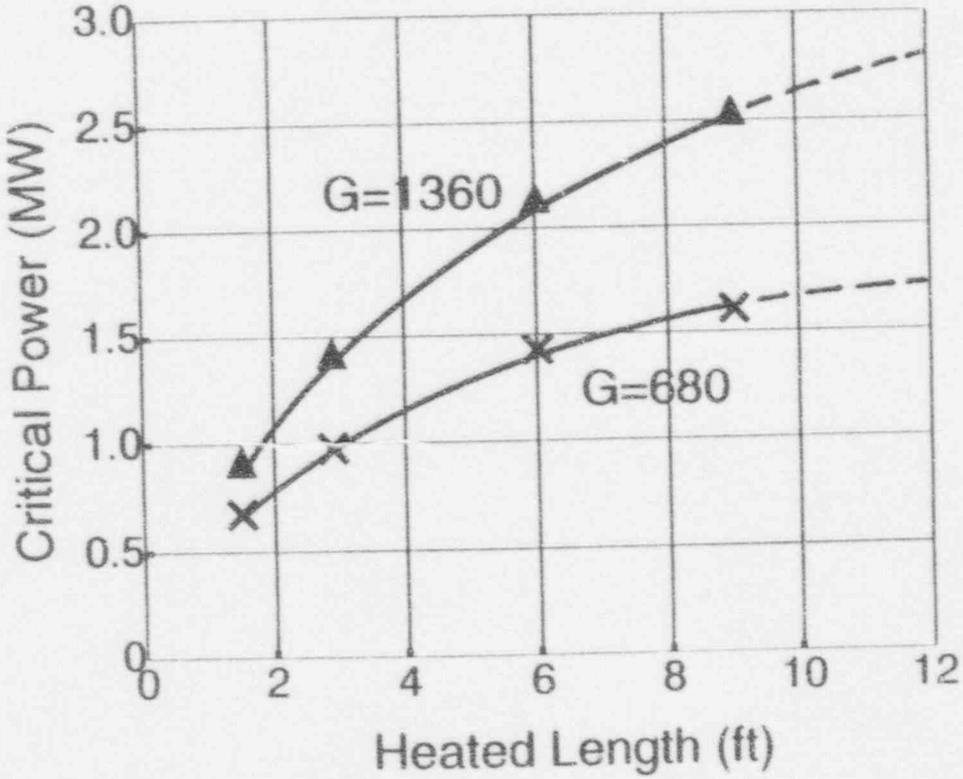



Figure 900.64-2

# COBRAG

## Critical Power Comparisons– Various Lattices

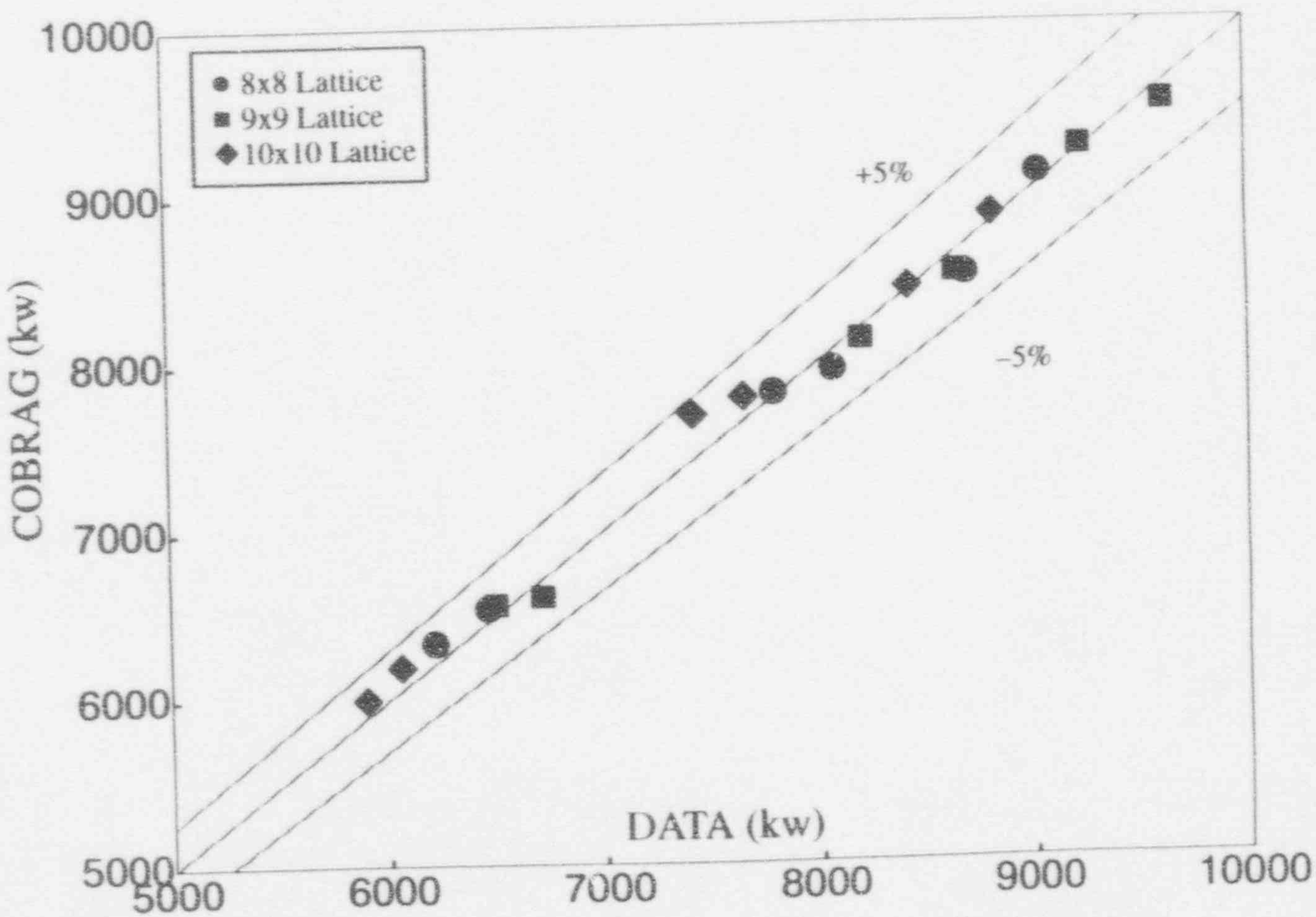



Figure 900.64-3

# COBRAG

## Critical Power Comparisons– Axial Power Shapes

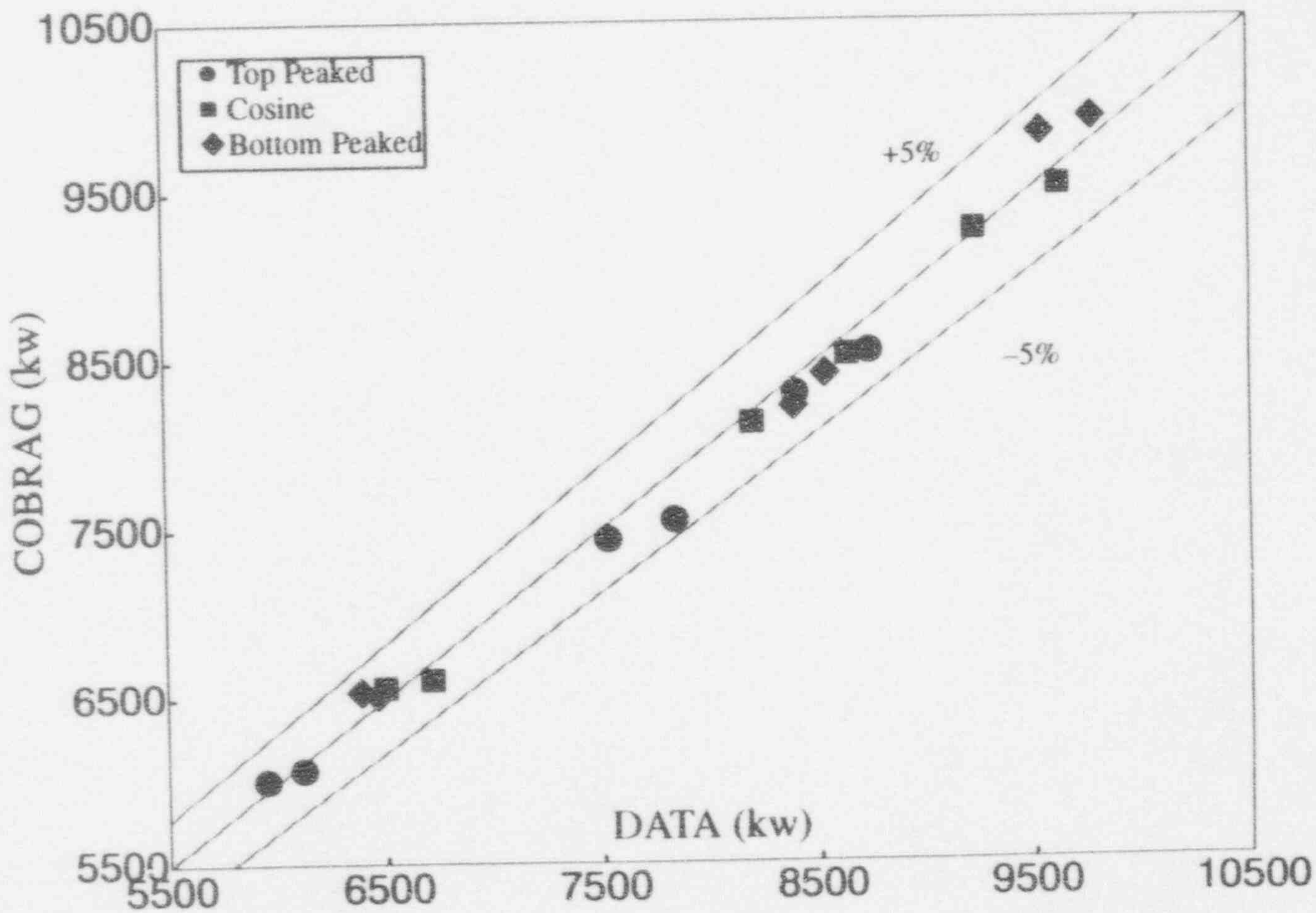



Figure 900.64-4

# COBRAG

## *Critical Power Comparisons– Different Spacer Types*

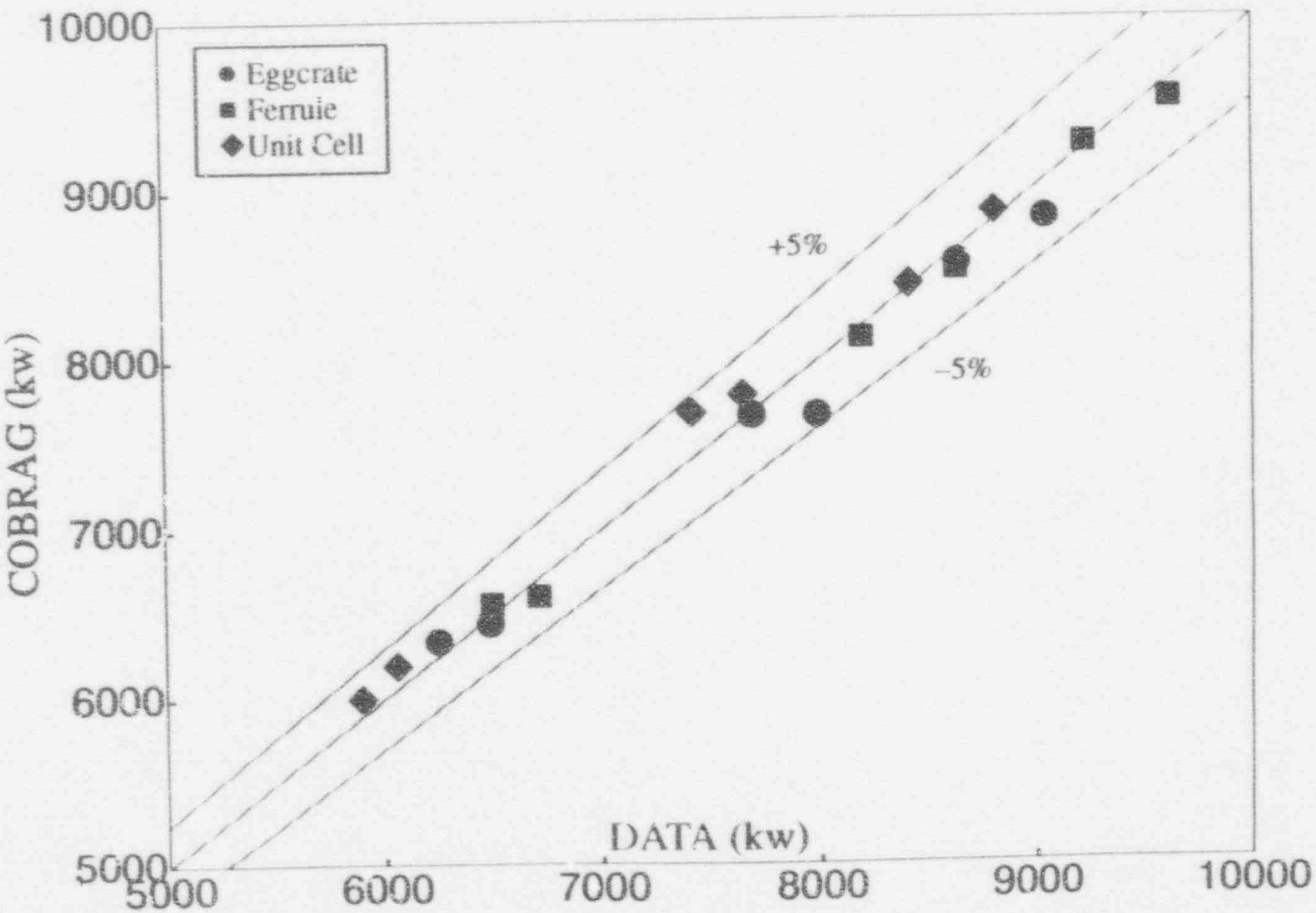



Figure 900.64-5

# COBRAG

## Critical Power Comparisons– Various Mass Fluxes

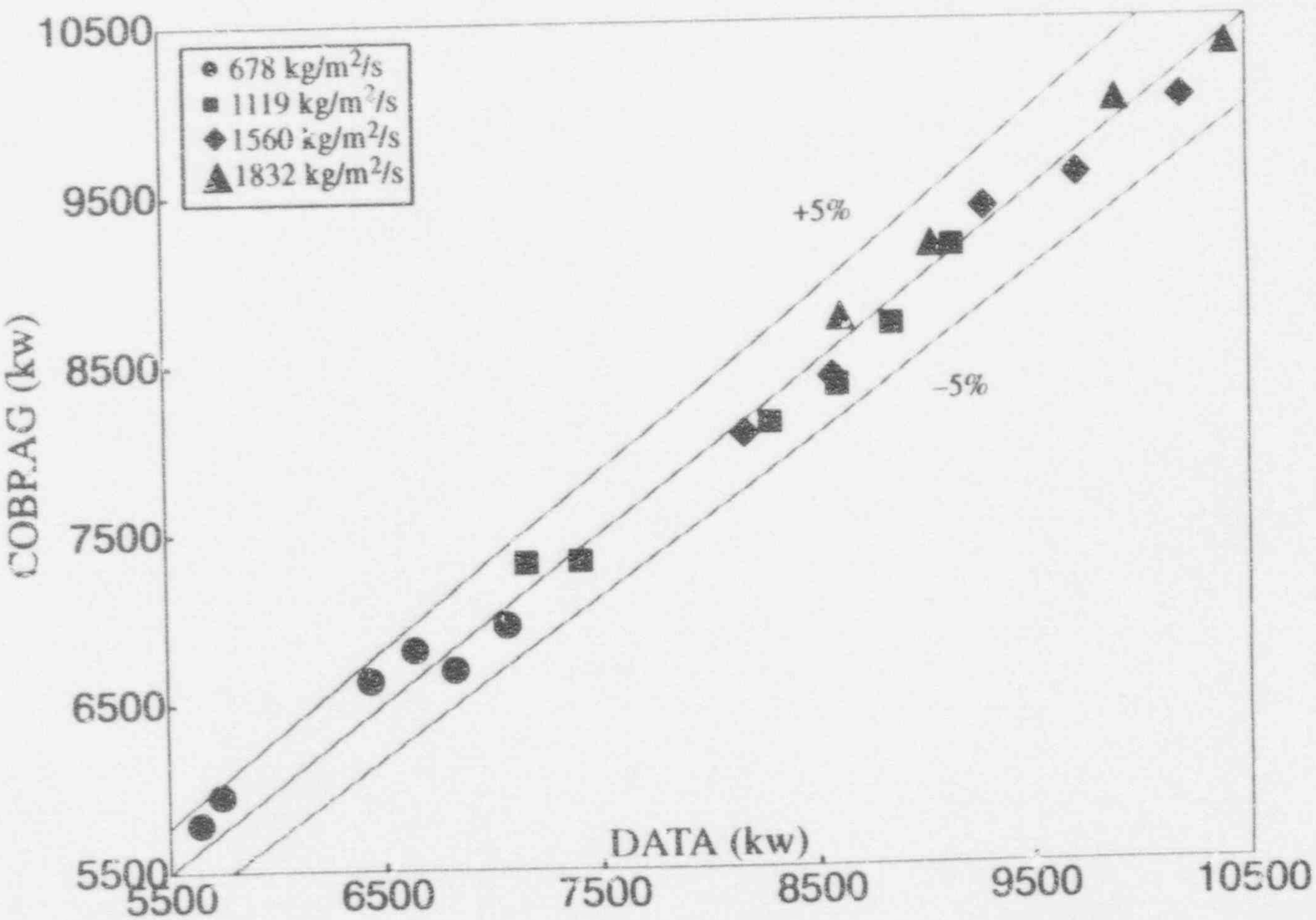
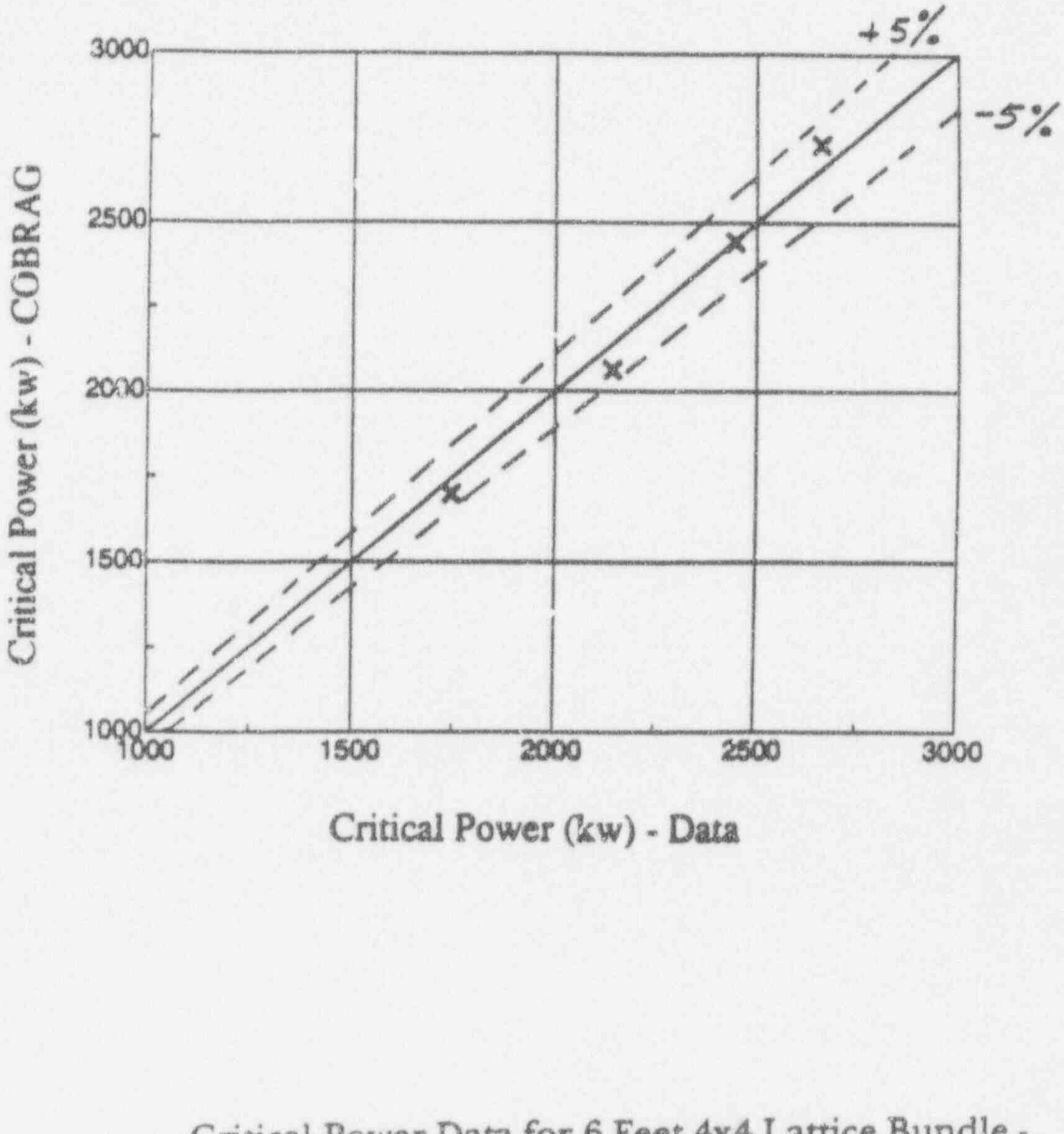
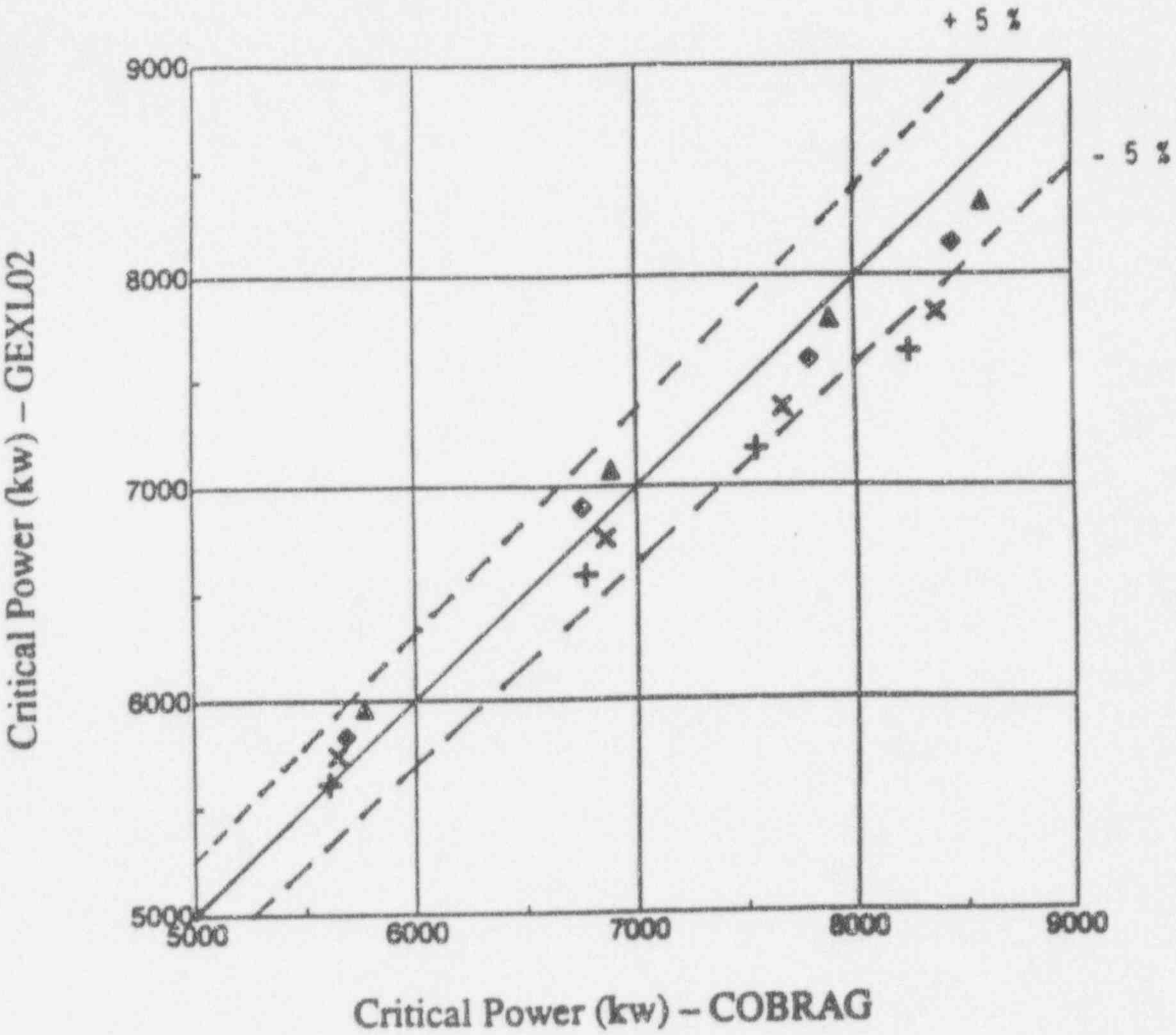



Figure 900.64-6




Critical Power Data for 6 Feet 4x4 Lattice Bundle - Columbia University Test Data Versus COBRA-G

**Figure 900.64-7**




Critical Power Data for 9 Feet GE8 Fuel Bundle - GEXL02 Correlation Versus. COBRA-G

Figure 900.64-8