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

February 24, 1997

Stephen M. Sohinki, Director
Office of Commercial Light
Water Reactor Production
Defense Programs
Department of Energy
Washington, DC 20585

SUBJECT: SUPPLEMENTAL GUIDANCE ON BENCHMARKING THE VIPRE CODE TO VALIDATE THE IMPLEMENTATION AND USER APPLICATION OF CHANGES TO ACCOMMODATE THE USE OF LITHIUM BURNABLE POISON RODS FOR THE PRODUCTION OF TRITIUM

Dear Mr. Sohinki:

By letter dated February 4, 1997, the staff transmitted the NRC safety evaluations which provide the conditions for acceptable use of the VIPRE-01 thermal hydraulic code to you for guidance. Upon review of your letter dated February 7, 1997, the staff concludes that the use of the VIPRE code needs to be further justified and benchmarked for the specific use described in the your "Report on the Evaluation of the Tritium Producing Burnable Absorber Rod Lead Test Assembly." In particular, the staff requests that detail on the modeling assumptions, choice of flow models and correlations, heat transfer correlations, CHF correlation and DNBR limit be presented and justified for the specified use. In addition, benchmarking of the VIPRE results to other NRC-approved codes should be included.

Enclosure 1 contains the NRC Safety Evaluation for use of the VIPRE-01 code at Seabrook, "Acceptance for Referencing of YAEC-1849P, 'Thermal-Hydraulic Analysis Methodology Using VIPRE-01 for PWR Applications,' for the Seabrook Station, Unit No. 1," dated August 15, 1994. This should provide more guidance regarding the information needed for review of the use of the VIPRE code for PWR licensing applications. In addition, Enclosure 2 contains the summary and abstract of NUREG/CR-4394, "Rod Bundle Film Boiling and Steam Cooling Data Base and Correlation Evaluation," which discusses the conservatisms of several commonly used film boiling and pure vapor heat transfer coefficients which may also be of assistance in the evaluation.

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PROJ

If you have any questions regarding this information, please contact the project manager, J. H. Wilson, at (301) 415-1108.

Sincerely,
David B. Matthews/for
Thomas T. Martin, Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Project No. 697

Enclosures: as stated

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for D B Matthews

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Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Project No. 697

Enclosures: as stated

cc: see next page

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

Docket File

August 15, 1994

Docket No. 50-443
Serial No. SEA-94-021

Mr. Ted C. Feigenbaum
Senior Vice President
and Chief Nuclear Officer
North Atlantic Energy Service Corporation
Post Office Box 300
Seabrook, New Hampshire 03874

Dear Mr. Feigenbaum:

SUBJECT: ACCEPTANCE FOR REFERENCING OF YAEC-1849P, "THERMAL-HYDRAULIC ANALYSIS METHODOLOGY USING VIPRE-01 FOR PWR APPLICATIONS", FOR THE SEABROOK STATION, UNIT NO. 1 (TAC M86958)

On February 2, 1993, North Atlantic Energy Service Corporation (North Atlantic) submitted for review three proprietary reports (YAEC-1849P, YAEC-1854P, and YAEC-1856P) relating to core reload analyses methodologies. North Atlantic has proposed to apply the methodologies described in these reports to support future operations at Seabrook Station, Unit No. 1 (Seabrook).

We have completed our review of YAEC-1849P which describes the methodology for core thermal-hydraulics analyses based on the VIPRE-01 code. The staff finds YAEC-1849P acceptable for referencing in licensing applications for Seabrook to the extent specified and under the limitations stated in YAEC-1849P and the enclosed Nuclear Regulatory Commission safety evaluation. The enclosed safety evaluation defines the basis for accepting YAEC-1849P for application to Seabrook.

If the NRC staff's criteria or regulations change such that the conclusions about the acceptability of the YAEC-1849P are invalidated, Yankee Atomic Electric Company (Yankee) should revise and resubmit the respective documentation or a justification should be submitted for the continued effective applicability of YAEC-1849P without revision.

The use of YAEC-1856P was approved previously (letter A. W. De Agazio to T. C. Feigenbaum, August 8, 1994). Our review of YAEC-1854P will be discussed in other correspondence.

Enclosure 1

The staff was assisted in this review by International Technical Services (ITS) Inc. under Contract No. NRC-03-90-027, JCN No. L1318, Task Order No. 021. Our safety evaluation (Enclosure 1) is based on the ITS Technical Evaluation Report (ITS/NRC/94-1) (Enclosure 2).

Sincerely,

Albert W. De Agazio

Albert W. De Agazio, Sr. Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Safety Evaluation
2. ITS Technical Evaluation Report

cc w/enclosures:
See next page

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE REVIEW AND APPROVAL OF YAEC-1849P - A THERMAL-HYDRAULIC

ANALYSIS METHODOLOGY USING VIPRE-01 FOR PWR APPLICATIONS

NORTH ATLANTIC ENERGY SERVICE CORPORATION, ET AL

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

On February 2, 1993, North Atlantic Energy Service Corporation (North Atlantic) submitted for review (Ref. 1) Yankee Atomic Electric Company (Yankee) report YAEC-1849P (Ref. 2). Additional supporting information was submitted on January 14, 1994 (Ref. 3). References 2 and 3 contain Yankee documentation related to the development of a core thermal-hydraulic methodology using the VIPRE-01 computer code (Ref. 4). The VIPRE-01 code would replace the COBRA-IIIC code in an NRC-approved methodology for the Yankee Atomic Power Station and the Maine Yankee Atomic Power Station (Ref. 5, 6, 7, 8).

North Atlantic intends to use VIPRE-01 for the Seabrook Station, Unit No. 1 (Seabrook) Departure from Nucleate Boiling (DNB) analysis. In the methodology submitted by North Atlantic, uncertainties are applied to the DNBR limit calculations for Seabrook using a statistical, rather than a deterministic, method by adapting the Westinghouse Revised Thermal Design Procedure (RTDP) (Ref. 9) as part of the DNB design basis approach. The RTDP method is a thermal-hydraulic analysis technique which computes DNB margin by statistically combining associated uncertainties.

The stated objectives of the YAEC-1849P are to:

- provide a description of the extension of NRC-approved Yankee subchannel analysis methodology to the VIPRE-01 code for application to Seabrook, and

- document the development of a fuel-design-specific WRB-1 correlation DNBR limit based on RTDP methodology to be used for Seabrook applications.

Towards these goals, the submitted documentation presents a plant specific geometric representation of the core, a selection of thermal-hydraulic models and correlations, and a description of the RTDP methodology using VIPRE-01 for Seabrook.

Since this is the first submittal based on use of the Yankee VIPRE-01 computer code, the review was also performed to assure fulfillment of VIPRE-01 safety evaluation report requirements (Ref. 4) and to assure conformity to the RTDP (Ref. 9) requirements.

2.0 SUMMARY OF YAEC-1489P

References 2 and 3 document descriptions of Yankee's VIPRE-01-based subchannel analysis methodology in support of core reloads for Seabrook and Maine Yankee. The subject methodology is an extension of the NRC approved methodology using COBRA-IIIC for use in Yankee and Maine Yankee applications. Descriptions are provided of plant specific core models, selections of thermal-hydraulic correlations, and Seabrook specific application of the Westinghouse RTDP methodology for determination of a safety limit on DNB. YAEC-1849P discusses applications of the methodology to Seabrook and other nuclear power plants. This SE is limited to the application of the methodology to Seabrook. Application to other facilities will be discussed elsewhere.

2.1 VIPRE-01 Computer Code

VIPRE-01 has been previously reviewed and approved for application to pressurized water reactors (PWR) in steady-state and transient analyses with heat transfer regimes up to the critical heat flux (CHF). The NRC safety evaluation (SE) on VIPRE-01 (Ref. 4) includes conditions requiring each user to document and submit to the NRC for approval its procedure for using VIPRE-01 and to provide justifications for the specific modeling assumptions made, the choice of particular two-phase flow models and correlations, heat transfer correlations, CHF correlation and DNBR limit, and input values of plant specific data such as turbulent mixing coefficient and grid loss coefficient including defaults. In References 2 and 3, the selection of the core model, use of certain thermal-hydraulic correlations, and other key input selections were described and justified. Details related to model selection are provided in Section 3.3 of this SE.

2.2 Revised Thermal Design Procedure Methodology

The RTDP was developed by Westinghouse to combine the system and correlation uncertainties associated with the statistical vice deterministic prediction of DNB. The RTDP methodology was approved by the NRC in 1989 with certain restrictions and is documented in WCAP-11397-P-A. These restrictions require the user to provide justification for any changes in DNB correlation, THINC-IV correlations, or parameter values listed in WCAP-11397-P-A that are outside of previously demonstrated acceptable ranges.

3.0 EVALUATION

3.1 VIPRE Model Description

VIPRE-01 correlations, including CHF correlations planned for use by Yankee for Seabrook is summarized in Table 3.3.1 of YAEC-1849P. The correlation selection is identical to that used with the COBRA-IIIC code which was reviewed and approved previously by the NRC, except for the turbulent mixing coefficients and CHF correlation.

Selection of each model or correlation is based upon one of the following considerations: (i) selection of some correlations/models are known to be insensitive to the Departure from Nucleate Boiling Ratio (DNBR) results; (ii) the selection is consistent with the current Final Safety Analysis Report (FSAR) methodology; (iii) the applicability range is appropriate; or (iv) sensitivity studies have demonstrated that some correlation selections were more conservative than those recommended in the VIPRE-01 manuals. Justification of the adequacy of the overall modeling approach was provided through benchmark calculations. This approach is acceptable to the staff.

3.1.1 Turbulent Mixing

The lateral momentum equation requires two parameters: a turbulent momentum factor (FTM); and a turbulent mixing coefficient. The FTM describes the efficiency of the momentum mixing. A conservative value was selected for FTM although the minimum DNBR is known to be insensitive to this parameter. The turbulent mixing coefficient is an important parameter since it determines the flow mixing rate. The value used for the mixing coefficient for Seabrook is documented in the current updated FSAR and was experimentally determined to be conservative based upon tests conducted using the 17x17 geometry and mixing vane grids on 26 inch spacing.

3.1.2 Critical Heat Flux Correlations

The VIPRE-01 SE requires that use of a new CHF correlation for VIPRE be qualified. The use of WRB-1 with VIPRE has been approved for other licensees. The WRB-1 correlation is used in the active fuel region above the first mixing vane grid. In an attempt to develop a Seabrook specific WRB-1 correlation limit, Yankee selected applicable (by fuel type) sets of test data from the original WRB data base to obtain a DNBR of 1.16 compared to the standard WRB-1 limit of 1.17. The staff, however did not accept this lower value because it is based on only a subset of the data. Yankee has agreed to use the currently approved DNBR limit for WRB-1 of 1.17, including the associated correlation statistics. This is acceptable to the staff.

The W-3 correlation will be used in situations where the W-3 (R-Grid) and WRB-1 correlations are not applicable. The use of the W-3 correlation has been approved previously for use with VIPRE-01. This is acceptable to the staff.

3.1.3 Other Model Selections

Yankee did not provide detailed justification for the selection of parameters or nodalizations discussed in this section. However, Yankee specifically addressed items in the VIPRE-01 SE to demonstrate that it had followed suggestions and recommendations cited in the SE in addition to the limitations described in the SE. This approach is acceptable to the staff. In addition, Yankee asserted that the ultimate justification is provided though the benchmark analyses discussed in Section 3.2 of this report. The staff agrees with this view.

3.1.3.1 VIPRE Nodalizations

The core nodalizations are presented in Reference 3. The radial and axial nodes used are presented in tables of VIPRE-01 Model Descriptions in Appendix B of YAEC-1849P. These nodalizations were reviewed and found acceptable.

3.1.3.2 Fuel Rod Modeling

The Conduction Rod model is used to compute the rod surface heat flux; when the heat flux is known, the Dummy Rod model is used. When the Conduction Rod model is used, the gap conductance is determined using the approved FROSSTEY code. The radial nodalization is similar to that used with the approved CHIC-KIN methodology currently used by Yankee for Maine Yankee, this is acceptable.

3.1.3.3 Inlet Flow Distribution

Yankee stated that the method used for determination of inlet flow maldistribution factor is consistent with the approved application of Yankee's subchannel methodology. The flow reduction factor is computed for each core to assess the effect of core-wide inlet flow maldistribution and of "mixed cores" on enthalpy rise in hot assembly locations for a range of power distributions. This flow reduction factor is applied to determine a conservative inlet flow for the subchannel model in the hot assembly, this is acceptable.

3.1.3.4 Power Distributions

The radial and axial power distributions are computed using an approved methodology using a combination of computer codes accounting for fuels, burnup, reactivity feedback, presence of burnable poisons and control rod position.

3.1.3.5 Numerical Solution Technique

The direct solution method was used for all of the Yankee plant models for most of the situations. Yankee will determine the use of the direct or RECIRC solution technique based upon whether the axial flow/crossflow ratio criteria is met, this is acceptable.

3.2 Seabrook Demonstration Analyses

Four cases were analyzed for the purpose of comparison with Westinghouse THINC-IV and THINC-III results. These cases were: (1) technical specification comparison, (2) Seabrook Core-1 steady state minimum departure from nucleate boiling ratio (MDNBR) vs power level, (3) Loss of Flow accident run as a transient case, and (4) Loss of Flow accident run as a series of static cases. In order for the comparison to be meaningful, VIPRE-01 was run using the W-3 correlation with a DNB limit of 1.3. Seabrook Station Technical Specification curves which define the reactor core safety limits in terms of RCS average temperature versus rated thermal power were generated using VIPRE-01. In Cases 1 and 2, Yankee's computed curves agree well between VIPRE and THINC-IV results. In Case 3, VIPRE transient results showed that VIPRE-01 was consistently more conservative than the THINC-III results. Yankee stated that against THINC-IV, the VIPRE results would be expected to be more comparable. When run as a series of steady state cases for the same loss of flow event, comparable results from those in Case 3 were obtained and comparison between the two code results were equally good.

3.3 Revised Thermal Design Procedure

The traditional method for accounting for the design and modeling uncertainties that enter into the determination of a DNBR assumes that key input parameters to the core thermal-hydraulic code are simultaneously at their worst level of uncertainty (a "deterministic method"). The RTDP was developed to remove excess conservatism of the deterministic method in that variations in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and DNB correlation predictions are considered statistically to obtain a DNB uncertainty factor. The methodology, therefore, assumes that the uncertainty associated with the DNB correlation can be statistically combined with the system uncertainties associated with seven independent parameters.

3.3.1 RTDP Parameter Statistical Distributions

The first step in development of the RTDP limits is to determine uncertainty distributions associated with seven independent parameters in RTDP. Uncertainties due to the computer codes used (VIPRE-01 and RETRAN) are also incorporated.

Yankee's use of a 4% uncertainty due to VIPRE-01 is an arbitrary selection. A 5% uncertainty for use in statistical DNB methodology has been approved previously by the NRC. There is no reason to believe that the code uncertainty has changed since such approval. Similarly, Yankee's use of a 1% uncertainty in transient analysis performed with RETRAN is equally arbitrary. Supporting data was not provided to justify the uncertainty values used. Based upon the only benchmark analysis provided comparing RETRAN calculations with plant data, at least 3% should be attributed for the RETRAN transient code/model uncertainty. Therefore the total value for code related uncertainty should be 8% instead of the 5% used by Yankee. Yankee stated (Ref. 10) that the use of about 8% in the total code uncertainties does not

affect the RTDP-based safety analysis limits. If the RTDP limits need to be re-evaluated in the future, the values stated above should be used for code uncertainties unless model improvements can be demonstrated to reduce uncertainties.

3.3.2 Determination of Sensitivity Factors

Sensitivity factors are determined over a wide range of statepoints covering operating conditions with MDNBR values near the expected design limit DNBR using RTDP. Sensitivity factors are defined as the percentage change in the VIPRE-01 calculated MDNBR resulting from a 1% change in a RTDP parameter. The most limiting statepoint was selected as the one for which the sensitivities resulted in the highest design-limit DNBR using RTDP.
This approach is acceptable.

3.3.3 RTDP DNBR Limits and Penalties

The RTDP DNBR limits are determined from standard deviation values, the independent parameters, the maximum sensitivity factors, and WRB-1 correlation statistics. The following DNBR penalties are applied to the computed RTDP DNBR limit to ensure a margin of safety:

- approximately 1% penalty for the fuel rod bowing effect,
- 3% DNBR penalty to account for lower plenum RCS flow anomaly, and
- 5% DNBR as margin available to offset potential future unidentified non-conservatisms, including cycle-to-cycle variations. This approach is acceptable to the staff.

4.0 CONCLUSION AND LIMITATIONS

YAEC-1849P, together with the information submitted with Reference 3 contain sufficient information to satisfy the SE requirements for VIPRE-01 and the RTDP. Furthermore the use of VIPRE-01 with appropriate CHF correlations for Seabrook is acceptable. The Yankee VIPRE-01 model for Main Steam Line Break and Rod Ejection analyses was reviewed previously (Ref. 10), and therefore, was not included in this review. The application of RTDP with VIPRE-01 for calculation of DNB limits for Seabrook is also acceptable subject to restrictions presented below:

1. Yankee continues to be subject to VIPRE-01 and RTDP SE requirements should any situation cited in the SE conditions arise to change the applicability of the current set of code/correlation/model combinations,

2. Use of the WRB-1 CHF correlation with VIPRE-01 is found acceptable with a DNBR limit of 1.17 instead of 1.16, and is limited to 26 inch spacing grids,

3. The uncertainties associated with VIPRE and RETRAN shall be 5% and 3%, respectively, and

4. Although the approach to the Yankee thermal-hydraulic subchannel methodology, as described in YAEC-1849P and Reference 3, is generally applicable to other PWR plants, this SE is applicable only to Seabrook due to the correlations and models selected as well as the use of specific uncertainties and distributions based upon plant specific data.

5.0 REFERENCES

1. Letter from T.C. Feigenbaum, North Atlantic Energy Systems, to USNRC, "Request for NRC Review and Approval of Analysis Methodologies to be Applied to Seabrook," February 2, 1993.

2. "Thermal-Hydraulic Analysis Methodology Using VIPRE-01 for PWR Applications," YAEC-1849P, October 1992.

3. Letter from T.C. Feigenbaum, North Atlantic Energy Systems, to NRC, "Response to Request for Additional Information (TAC M86957 and TAC M86958)," March 9, 1994.

4. Letter from C.E. Rossi NRC to J. A. Blaisdell (UGRA), "Acceptance for Referencing of Licensing Topical Report VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM, Vols. 1-4," May 1, 1986.

5. "Maine Yankee Core Thermal-Hydraulic Model Using COBRA-IIIC," YAEC-1102, June 1976.

6. "A Thermal-Hydraulic Analytical Model Using COBRA IIIC," YAEC-1058, May 1974.

7. "DNBR Limit Methodology and Application to the Maine Yankee Plant," YAEC-1296P, January 1982.

8. Letter from R. A. Clark NRC, to J. H. Garrity YAEC, Safety Evaluation by the Office of Nuclear Reactor Regulation on "Topical Report YAEC-1296P "DNBR Limit Methodology and Application to the Maine Yankee Plant," March 9, 1983.

9. Letter from A.C. Thadani NRC, to W. J. Johnson, Westinghouse, "Acceptance for Referencing of licensing Topical Report WCAP-11397, "Revised Thermal Design Procedure"," January 17, 1989.

10. "STAR Methodology Application for PWR's Control Rod Ejection, Main Steam Line Break," YAEC-1752-A, October 1990.

Principal Contributor: Lambros Lois
Date: August 15, 1994

ornl

NUREG/CR-4394
ORNL/TM-9628

OAK RIDGE
NATIONAL
LABORATORY

MARTIN MARIETTA

Rod Bundle Film Boiling and Steam Cooling Data Base and Correlation Evaluation

G. L. Yoder

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-551-75 and 40-552-75

OPERATED BY
MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

Enclosure 2

SUMMARY

In the process of boiling on vertical heated rods, vapor generated at the surface is swept away by the flowing liquid (forced convection) or buoyancy forces (pool boiling). As the rods are traversed axially from bottom to top, eventually one of two conditions must occur. Either the surface will become dry due to vapor void generation, or the surface will become dry because the critical heat flux (CHF) point has been reached. In either case, the transition between the wet and dry portions of a rod also marks a transition between heat transfer regimes. If the rods are wet, surface heat transfer is very good, with correspondingly low rod surface temperatures. If the surface of the rod is dry, heat transfer from the rod to the vapor is relatively poor, resulting in high rod surface temperatures.

The object of this report is to document a data base which has been assembled from experimental rod bundle data taken in the post-CHF regime. The data base includes data from 10 film boiling references and 5 steam cooling references in total, over 20,000 data points are included. These data have been used to evaluate several film boiling and steam cooling heat transfer correlations.

Film Boiling

- Dougall-Rohsenow
- Dougall-Rohsenow (wall Prandtl Number)
- Condie-Bengston IV
- Groeneveld 5.7
- Groeneveld 5.9
- Groeneveld-Delorme

Steam Cooling

- Dittus-Boelter
- Dittus-Boelter (film properties)
- Sozer

Comparisons indicated that the Condie-Bengston IV correlation predicted best in the film boiling regime. While all three of the steam cooling correlations examined performed similarly, the Dittus-Boelter (film) correlation and Sozer correlation predicted most conservatively.

ROD BUNDLE FILM BOILING AND STEAM COOLING DATA BASE AND CORRELATION EVALUATION

Graydon L. Yoder
Engineering Technology Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

ABSTRACT

Film boiling and steam cooling data from several rod bundle experiments have been compiled from fourteen references. Steam cooling data are presented for five references, while film boiling data are presented for ten references (two of these for tube geometry). The compilation includes experimental parameters necessary for characterization of the local rod heat transfer.

These data have been compared to several commonly used film boiling and pure vapor heat transfer coefficients. Results show that the Dougall-Rohsenow correlation tends to overpredict film boiling heat transfer, while Condie-Bengston IV and the Groeneveld 5. series correlations do a reasonable job of predicting film boiling heat fluxes. Of the three vapor correlations examined, the Dittus-Boelter correlation is least conservative, and the Sozer correlation most conservative.

1. INTRODUCTION

In the process of boiling on vertical heated rods, vapor generated at the surface is swept away by the flowing liquid (forced convection) or buoyancy forces (pool boiling). As the rods are traversed axially from bottom to top, eventually one of two conditions must occur. Either the surface will become dry due to vapor void generation, or the surface will become dry because the critical heat flux (CHF) point has been reached. In either case, the transition between the wet and dry portions of a rod also marks a transition between heat transfer regimes. If the rods are wet, surface heat transfer is very good, with correspondingly low rod surface temperatures. If the surface of the rod is dry, heat transfer from the rod to the vapor is relatively poor, resulting in high rod surface temperatures.

If dryout of the rod occurs due to vapor void generation (i.e., the void fraction and quality become one, steam cooling exists above the dryout point. If CHF occurs with quality less than one, film boiling exists above dryout, and although the surface of the rods are dry due to very high temperatures, liquid is still present in the flow.

High surface temperature under dry rod conditions has made film boiling a flow pattern which has been studied extensively over the years. Steam cooling has also been investigated, although to a lesser extent, since single phase flow behavior is much easier to understand and predict than two phase behavior. Until recently, the major portion of both two phase film boiling and single phase steam cooling heat transfer data was acquired using tubular test sections. Data of this type can be found in many references,[1-5] and evaluation of several heat transfer correlations has been performed using tube data.[6,7] Within the last ten years or so, larger facilities incorporating bundles of rods have been used to gather film boiling and steam cooling data. Data from several experiments have been gathered and evaluation of this rod bundle data is documented in this report.

A few general ground rules for data acceptance can be stated. Rod surface conditions (heat flux, temperature) and local flow conditions (quality, mass flow, temperature) must be provided by the experimenter or be calculable, in order to determine the local heat transfer. The limited scope of this investigation did not allow transient bundle calculations to be performed when only bundle boundary conditions or local instrument responses were available. The data base is assembled from bundle heat transfer data except where experimenters specifically asked that their tube data be included in the data base. Other criteria were also used and will be explained on a specific basis in Sect. 2. Data gathered during the course of this investigation will be placed on the INEL data bank.

Data from fourteen references are included in the data base and are compared to several commonly used correlations.

Film Boiling

- Dougall-Rohsenow[8]
- Dougall-Rohsenow[9] (wall Prandtl Number)
- Condie-Bengston IV[10]
- Groeneveld 5.7[11]
- Groeneveld 5.9[11]
- Groeneveld-Delorme[12]

Steam Cooling

- Dittus-Boelter[13]
- Dittus-Boelter (film properties)
- Sozer[14]

Five sets of data were acquired under steam cooling conditions while ten sets of data were taken under film boiling conditions (two of these in tube geometry). A summary of individual investigations and overall film boiling and steam cooling data ranges is presented in Table 1.1. This table also indicates the number of rods used in the bundle, the type of data gathered during testing (film boiling or steam cooling), an indication of whether heat transfer data is available near the bundle spacer grids, the type of test performed (steady state or transient), and the type of heating within the bundle (uniform or non-uniform), as well as the range of conditions covered during testing.

Table 1.1 Rod bundle film boiling/steam cooling data sources

Investigation	Ref no.	No. rods	Data type	Grid info?	Test Type	Bundle heating	Heat flux (kw/m^2)	Mass flux (kg/m^2s)	Pressure (MPa)	Quality (%)	Hydraulic diameter (cm)	Wetted to heated perimeter ratio
Adorni	15	7	Film	No	SS	UA-UR	210-1550	1100-3800	5.0	30-80	0.507, 0.709	1.5, 1.55
Groeneveld	16	3	Film	No	SS	Fuel elements	400-1100	1100-2000	6.5	40-60	1.23	1.94
Matzner	17	19	Film	No	SS	UA-NR	1000-1900	650-2650	7.0	20-50	0.755	1.41
Matzner	18	19	Film	No	SS	UA-NR	800-1200	700-2050	3.-8.	40-60	0.534	1.7
Yoder	19	64	Film	Yes	SS	UA-UR	320-940	250-800	4.5-13.	40-120	1.06	1.3
Yoder	20	64	F,S	Yes	SS	UA-UR	100-470	40-260	3.8-8.5	80-160	1.06	1.3
Anklam	21	64	Steam	Yes	SS	UA-UR	25-40	10-18	2.6-7.1	-	1.06	1.3
Anklam	22	64	Steam	Yes	SS	UA-UR	10-80	5-30	4.1-7.2	-	1.06	1.3
Wong	23	161	Steam	Yes	SS	NA-UR	0.5-5	3-23	0.3	-	1.18	~1.28
Morris	24	64	Film	Yes	T	UA-UR	160-1100	130-1090	5.2-12.4	20-150	1.06	1.3
Lee	25	161	Film	Yes	T	NA-UR	10-50	1-140	0.14-0.3	20-100	1.18	~1.28
Kawasaki	26	Tube	Film	No	SS	UA	40-300	250-550	4.8-14.7	50-120	1.08	1
Smith	27	Tube	Film	No	SS	CT	10-400	25-160	0.14-.3	10-80	1.25	1
Loftus	28	21	Steam	Yes	SS	NA-UR	0.4-4.5	3-17.5	0.14	-	0.817	1.55

Symbols

CT - controlled temperature
UA - uniform axially
UR - uniform radially
NA - nonuniform axially
NR - nonuniform radially
SS - steady state
T - transient

Data within the data base can be used to assess and develop correlations used to predict fuel rod behavior under film boiling or steam cooling conditions. Enough information was included for each data point that this should be a relatively painless exercise. The data base was designed to require only a water-steam thermodynamic and physical properties package to accomplish this task.

Section 2 describes each experiment individually, while section 3 shows details of the data parameter ranges. Section 4 describes the correlations evaluated and shows the results of the evaluations. Conclusions are presented in Section 5. The appendix describes the format used in tabulating the data for the INEL data bank.

Project No. 697
DOE Tritium Program

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