December 5, 2007

Mr. Ronnie L. Gardner, Manager
AREVA NP
3315 Old Forrest Road
P.O. Box 10935
Lynchburg, VA 24506-0935

SUBJECT: FINAL SAFETY EVALUATION REPORT FOR ANP-10269P, "THE ACH-2 CHF CORRELLATION FOR U.S. EPR TOPICAL REPORT" (TAC NO. MD0026)

Dear Mr. Gardner:

By letter dated November 30, 2006 (NRC's ADAMS Accession Number ML063390039), as supplemented by letters dated May 29, 2007 (ML071500525), and August 15, 2007 (ML072620333), AREVA NP (AREVA) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review proprietary and non-proprietary versions of Topical Report (TR) ANP-10269, Revision 0, "The ACH-2 CHF Correlation for the U.S. EPR Topical Report." By letter dated September 27, 2007, a draft safety evaluation (SE) regarding our approval of ANP-10269(P) was provided for your review and comments (ML072630042). The staff has received verbal comments from AREVA regarding proprietary information inadvertently included in the draft SE and typographical errors. We have corrected the typographical errors and removed the proprietary information in the final SE enclosed with letter. AREVA also informed us of the existence of typographical errors in Section 7.1.1 of the topical report and proposed to correct them in the approved version of the SE. The staff agrees with AREVA's proposal to correct the errors in the approved version of the TR.

The staff has found that ANP-10269(P), Revision 0 is acceptable for referencing in licensing applications for U.S. EPR to the extent specified and under the limitations delineated in the TR and in the enclosed SE. The SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in regulatory applications, our review will ensure that the material presented applies to the specific application involved. Regulatory applications that deviate from this TR will be subject to further review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that AREVA publish the accepted version of this TR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE after the title page. Also, the accepted version must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, AREVA will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

If you have any questions, please contact me at gxt2@nrc.gov or (301) 415-3361.

Sincerely,

/RA/

Getachew Tesfaye, Sr. Project Manager
EPR Projects Branch
Division of New Reactor Licensing
Office of New Reactors

Project No. 733

Enclosure: Final Safety Evaluation

cc w/encl: U.S. EPR Service List

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DISTRIBUTION:
E-mail:
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NAME	GTesfaye	PCoates	JDonoghue*	JColaccino
DATE	11/27/07	11/28/07	12/4/07	12/05/07

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FINAL SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS

ANP-10269P, REVISION 0, "THE ACH-2 CHF CORRELATION

FOR THE U.S. EVOLUTIONARY PRESSURIZED WATER REACTOR (EPR)

TOPICAL REPORT" (TAC NO. MD0026)

PROJECT NO. 733

1.0 INTRODUCTION AND BACKGROUND

By letter dated November 30, 2006 (NRC's ADAMS Accession Number ML063390039), as supplemented by letters dated May 29, 2007 (ML071500525), and August 15, 2007 (ML072620333), AREVA NP (AREVA) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review proprietary and non-proprietary versions of Topical Report (TR) ANP-10269, Revision 0, "The ACH-2 CHF Correlation for the U.S. EPR Topical Report." AREVA requested that the NRC issue a safety evaluation report which approves the use of the correlation.

Topical Report ANP-10269P, Revision 0 (Reference 1) describes the development of a critical heat flux (CHF) correlation for the fuel geometry of the U.S. EPR fuel design. The fuel assembly for the AREVA's U.S. EPR will utilize high thermal performance (HTP) spacer grids. These grids provide lateral fuel rod support via a flow channel in the rod-to-rod gap. At the outlet (downstream edge) of the grid, the flow through these channels is diverted from the vertical to promote increased thermal mixing. AREVA has developed a CHF correlation specifically for the fuel geometry of the EPR fuel design. This correlation, known as ACH-2, is based solely on an EPR CHF database developed from CHF testing performed at AREVA's Karlstein Thermal Hydraulic Facility in Germany. The correlation was developed specifically for the EPR fuel design important variables of grid spacing, pin pitch, rod and guide tube diameter and heated length.

The functional form of the CHF correlation is empirical and is based solely on experimental observations of the relationship between the measured CHF and the correlation variables. The correlation includes the following variables: pressure, local mass velocity, local quality, axial elevation of CHF, standard BWU (Tong) F Factor, and guide tube outer diameter.

The NRC staff reviewed TR ANP-10269P, Revision 0 and its supplements (References 1, 2, and 3), and its evaluation and findings are documented below.

2.0 REGULATORY EVALUATION

Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR) requires that safety limits be included in the plant-specific technical specifications. Pursuant to 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 10, "Reactor Design," the reactor core and associated coolant, control, and protective systems, are required to be

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designed with an appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences. To ensure compliance with GDC 10, the NRC staff confirmed that the vendor performed the departure from nucleate boiling (DNB) analyses using NRC-approved methodologies as described in NUREG-0800, "Standard Review Plan," Section 4.4.

In addition, Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 34, "Contents of applications; technical information, " requires that safety analysis reports for plant specific designs be submitted that analyze the design and performance of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. As part of the core reload design process, licensees (or vendors) will perform plant specific reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle. To confirm that the analyses remain bounding, licensees will confirm those key inputs to the safety analyses (such as the CHF) are conservative with respect to the initial, as built, design cycle. If key safety analysis parameters are not bounded, a re-analysis or a re-evaluation of the affected transients or accidents will be performed to ensure that the applicable acceptance criteria are satisfied.

The NRC staff's review was based on the evaluation of the technical merit of the submittal and compliance with any applicable regulations associated with reviews of TR and its supplements (Reference 1, 2, and 3).

3.0 TECHNICAL EVALUATION

In nuclear reactor cores, energy is generated by uranium dioxide (or plutonium) fuel pellets enclosed in fuel rods. The energy leaves the fuel rod surface in the form of heat flux, which is removed by the coolant system flow. The normal mode of heat transfer to the coolant at high power densities is nucleate boiling. Nucleate boiling is an efficient mode of heat transfer, with heat transfer coefficients about 50,000 Btu/hr-ft^2-F^o. As the coolant temperature increases and its capacity to accept heat from the fuel rod surface and transfer it by bubble detachment to the coolant stream degrades, a continuous layer or film of steam starts to blanket the rod. The steam film insulates the rod and the heat transfer coefficient drops drastically to around 500 Btu/hr- ft^2-F^o. This is because the heat transfer mechanism is film boiling, primarily conduction through the steam layer. Reactor cores must be protected against possible damage that could result from the high clad temperatures that are experienced in the transition to, and during, film boiling. The heat flux at which the steam film starts to form is termed the CHF or the point of departure from DNB.

For design purposes, the departure from nucleate boiling ratio (DNBR) is used as an indicator of the margin to DNB. The DNBR is the ratio of the predicted CHF to the actual heat flux at the same condition. Therefore, the DNBR is a measure of the thermal margin to film boiling and its associated high temperatures. The greater the DNBR value, the greater the thermal margin. Because the phenomenon of CHF cannot be predicted from first principles, it is measured under laboratory conditions approximating those of a real reactor core. From these experimental measurements, a CHF correlation is developed. This correlation is essentially an empirical regression of the independent variables from the experiments. The method used for the correlation development by AREVA consists of several steps as described below.

The first step in the correlation process is to gather and to certify, as needed, the necessary CHF data. This involves determining the necessary geometric and thermal hydraulic ranges for testing.

The second step in developing a correlation is to establish the level or magnitude of each independent variable for each point of the database (each state point run in each of the tests). There are two classes of independent variables: local thermal hydraulic conditions (e.g., mass velocity and thermodynamic quality), and bundle global conditions (e.g., heated length and grid spacing). While the global values are known, the local conditions within the sub-channels must be calculated based on the measured bundle experimental values of flow, power and system pressure. This task is performed using a thermal-hydraulic computer code such as LYNXT (Reference 4).

The third step is to choose a correlation form that accurately describes the CHF phenomena for the available data (Reference 6). This includes choosing and justifying the independent variables.

The fourth step is to correlate the data, including sorting the data by flow regime and optimization of the correlation coefficients. A sequential optimization technique is normally used (Reference 5), with both a correlation and verification database.

The final step in developing a correlation is to calculate a 95/95 design limit (Reference 7) and to verify that the correlation describes the CHF phenomena accurately and without bias over its entire database. The verification process includes visual and numerical testing of all independent variables for bias.

The staff finds the method employed by AREVA for the correlation development acceptable.

3.1 DATABASE

Critical heat flux test data were taken at the AREVA's Karstein thermal hydraulic facility (KATHY). In the summer of 2006, the NRC staff observed the data collection at KATHY for the purpose of developing the ACH-2 CHF correlation for the U.S. EPR.

AREVA states that since the ACH-2 correlation is the first U.S. pressurized-water reactor application based on KATHY's data, it was necessary to demonstrate the acceptability, through qualification, of using the KATHY facility for PWR CHF data. This qualification was composed of a comparison of CHF performance measured in the KATHY facility and the Columbia University's heat transfer research facility (HTRF) for the same fuel design. AREVA's test geometry is provided in Reference 1. The test results indicate that there is no independent variable bias for either test over the entire test range, and confirms that KATHY CHF data is qualified for use in CHF correlation development. Reference 1 addresses the development and basis for the ACH-2 CHF correlation as a conservative predictor of the CHF performance of the EPR high thermal performance spacer grid. ACH-2 was developed from test data using distinct test geometries shown in Table 2.1 of Reference 1. This database includes data for heated lengths, unit cell and guide tube geometry and both cosine and uniform axial flux shapes.

AREVA indicated that the original EPR CHF test program was designed to produce a specific CHF correlation with applicability over and beyond the entire range of operating conditions of the U.S. EPR. Individual tests cover the local condition variables of pressure (P), mass velocity (G), and thermodynamic quality at CHF (X). Separate individual tests were also needed to establish geometry factors (unit cell and guide tube) and non-uniform AFS heat addition effects.

The non-uniform AFS data was limited to a heated specified length in Reference 1. Subsequent to the submittal of Reference 1, AREVA NP performed additional CHF testing for the length of the U.S. EPR design (14 ft) for the purpose of confirming the adequacy of the ACH-2 CHF correlation and its CHF design limit. The results and their applicability to the ACH-2 CHF correlation and departure from nucleate boiling ratio design limit (DNBRL) are evaluated by AREVA in Reference 2.

AREVA's original development of the ACH-2 CHF correlation (Reference 1) included several conservative provisions, one of which was a 0.05 increase in the application design limit. In order to remove the 0.05 application DL penalty, AREVA provided additional CHF test data (Reference 2) that confirmed the applicability of the ACH-2 CHF correlation for the U.S. EPR fuel.

Based on the NRC observation of the test data collection at KATHY, the staff finds that the CHF testing was conducted in an acceptable manner and in accordance with the test procedures. The staff agrees that the additional data provided by AREVA (Reference 2) justifies removal of the 0.05 increase in the application design limit.

3.2 SUITABILITY OF THE DATABASE

The staff reviewed all the test data points (Reference 2, Table S.5-2) that were used in the ACH-2 CHF correlation development and validation. The test data were grouped into several common groups for assessing the group statistics. A DNBRL value for each group is determined using both Owen (Reference 8) and Somerville (Reference 9) methods. Table S.5-2 (Reference 2) shows the statistics associated with various grouping of the CHF data, including the ACH-2 database from Reference 1.

Standard statistical tests were performed (References 1 and 2) to determine if all or selected data groups belong to the same population in order to be combined for the evaluation of the 95/95 DNBR tolerance limit. In addition, scatter plots were generated for each variable in the correlation to examine the correlation for trends or regions of non-conservatism. The Measured/Predicted CHF ratio was plotted as a function of the local mass velocity, the system pressure, the local quality, the axial location of CHF, and the non-uniform shape (Tong) F factor.

The NRC staff examined these plots and determined that no trends or regions of non-conservatism were evident, and determined that the results were typical.

3.3 CHF CORRELATION

AREVA NP has developed a CHF correlation specifically for the fuel geometry of the EPR fuel design. This correlation, known as ACH-2, is based solely on an EPR CHF database developed from CHF testing performed at AREVA's Karlstein Thermal Hydraulic Facility in Germany. The correlation was developed specifically for the EPR fuel design important variables. Reference 1 describes the CHF correlation in terms of: standard BWU (Tong) F Factor, natural log of pressure, mass velocity, thermodynamic quality fraction at CHF, axial elevation of CHF, and guide tube outer diameter. The correlation coefficients were based on a subset of the total test data, referred to as the correlation database.

The NRC staff has reviewed the methodology and its application, and found that the methodology is acceptable. In addition, the staff has traveled to Germany to observe data collection at KATHY which it found to be acceptable. Therefore, the NRC staff finds there is reasonable assurance that the database used to develop the ACH-2 CHF correlations for U.S. EPR was based on quality data representative of the design and that the statistical treatment of the database was based on widely recognized statistical methods (Reference 7).

4.0 CONCLUSION

The NRC staff has reviewed the subject reports (References 1 and 2) and the responses to the NRC staff RAIs (Reference 3), and found that the justification for the database and the methodology used for the proposed ACH-2 CHF correlation for the U.S. EPR are acceptable because: 1) an acceptable methodology and DNBR design limits (Reference 7) are used, and 2) at a minimum a 95 percent probability with 95 percent confidence (95/95) exists that a departure from nucleate boiling will not occur.

The NRC staff has found that the ACH-2 CHF correlation for the U.S. EPR TR (Reference 1) and its supplement (Reference 2) are acceptable for referencing in licensing applications for EPR pressurized-water reactors with HTP spacer grid designs, to the extent specified and under the limitations delineated in the TR with a range of applicability given below. The staff's acceptance applies only to material provided in the subject TR and its supplements.

<u>Fuel Design Limitations of the ACH-2 CHF Correlation:</u>
The local conditions ranges and Fuel design limitations of the ACH-2 CHF correlation are as follows:

<u>Local Conditions Ranges:</u>
Pressure: 284 to 2565 psia
Mass Velocity: 0.945 to 3.164 Mlb/hr-ft ^{2}
Thermodynamic Quality at CHF: less than 37%

<u>Fuel Design Limitations:</u>
Spacer Grid Type: AREVA NP EPR HTP
Grid Spacing: 18.5 to 20.0 inches
Fuel Rod OD: 0.374 inches
Guide Tube OD: 0.490 inches
Fuel Rod Pitch: 0.496 inches

<u>Application:</u>
Code: LYNXT (Reference 4)
DNBRL: 1.25

5.0 REFERENCES

1. The ACH-2 CHF Correlation for the U.S. EPR Topical Report (ANP-10269P, Revision 0), dated November 2006 (AREVA Proprietary).

2. The ACH-2 CHF Correlation for the U. S. EPR Topical Report Supplement (ANP-10269P, Revision 0, Supplement), dated August 2007(AREVA Proprietary).

3. Letter from R. L. Gardner (AREVA) to NRC, dated May 29, 2007, Response to a Request for Additional Information Regarding ANP-10269P, "The ACH-2 CHF Correlation for the U.S. EPR Topical Report" (TAC No. MD0026).

4. J. H. Jones, "LYNXT Core Transient Thermal Hydraulic Program," BAW-10156A, Rev. 01, August 1993.

5. D. A. Farnsworth, "Linearization and Sequential Optimization of Nonlinear Empirical Equations," Proceedings of the International Topical Meeting on Advances in Mathematics, Computations and Reactor Physics (Vol. 2), May 1991.

6. "USNRC Standard Review Plan," NUREG-0800.

7. "USNRC: Applying Statistics," Dan Lurie, February 1994.

8. D. B. Owen, "Factors for One-Sided Tolerance Limits and For Variables Sampling Plans," Sandia Corporation Monograph (SRC-607), March 1963.

9. Paul N. Somerville, "Tables for Obtaining Non-Parametric Tolerance Limits," Annals of Mathematical Statistics, Vol. 29, No. 2, pp. 599-601, June 1958.

Principal contributor: Fred M. Forsaty

Date: December 5, 2007

cc:

Mr. Glenn H. Archinoff
AECL Technologies
481 North Frederick Avenue
Suite 405
Gaithersburg, MD 20877

Marty Bowling
NUMARK Project Manager
86 WestBay Drive
Kilmarnock, VA 22482

Ms. Michele Boyd
Legislative Director
Energy Program
Public Citizens Critical Mass Energy
and Environmental Program
215 Pennsylvania Avenue, SE
Washington, DC 20003

W. Craig Conklin, Director
Chemical and Nuclear Preparedness &
Protection Division (CNPPD)
Office of Infrastructure Protection
Department of Homeland Security
Washington, DC 20528

Mr. Marvin Fertel
Senior Vice President
and Chief Nuclear Officer
Nuclear Energy Institute
1776 I Street, NW
Suite 400
Washington, DC 20006-3708

Mr. Ray Ganthner
AREVA, Framatome ANP, Inc.
3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506-0935

Mr. Paul Gaukler
Pillsbury, Winthrop, Shaw, Pittman
2300 N Street, NW
Washington, DC 20037

Dr. Charles L. King
Licensing Manager, IRIS Project
Westinghouse Electric Company
Science and Technology Department
20 International Drive
Windsor, CT 06095

Ms. Sherry McFaden
Framatome NP, Inc.
3315 Old Forest Road, OF-16
Lynchburg, VA 24501

Vanessa E. Quinn, Acting Director
Technological Hazards Division
National Preparedness Directorate
Federal Emergency Management Agency
500 C Street, NW
Washington, DC 20472

Mr. Steve Seitz
AREVA
100 Dean Road
East Lyme, CT 06333

Mr. Robert E. Sweeney
IBEX ESI
4641 Montgomery Avenue
Suite 350
Bethesda, MD 20814

Mr. Gary Wright, Director
Division of Nuclear Facility Safety
Illinois Emergency Management Agency
1035 Outer Park Drive
Springfield, IL 62704

<u>Email</u>

APH@NEI.org (Adrian Heymer)
awc@nei.org (Anne W. Cottingham)
bennettS2@bv.com (Steve A. Bennett)
bob.brown@ge.com (Robert E. Brown)
BrinkmCB@westinghouse.com (Charles Brinkman)
carey.fleming@constellation.com (Carey Fleming)
chris.maslak@ge.com (Chris Maslak)
cwaltman@roe.com (C. Waltman)
david.hinds@ge.com (David Hinds)
david.lewis@pillsburylaw.com (David Lewis)
dlochbaum@UCSUSA.org (David Lochbaum)
erg-xl@cox.net (Eddie R. Grant)
frankq@hursttech.com (Frank Quinn)
gcesare@enercon.com (Guy Cesare)
greshaja@westinghouse.com (James Gresham)
james.beard@gene.ge.com (James Beard)
jcurtiss@winston.com (Jim Curtiss)
jgutierrez@morganlewis.com (Jay M. Gutierrez)
jim.riccio@wdc.greenpeace.org (James Riccio)
JJD1@nrc.gov (John Donohue)
JJNesrsta@cpsenergy.com (James J. Nesrsta)
john.o'neil@pillsburylaw.com (John O'Neil)
Joseph.savage@ge.com (Joseph Savage)
Joseph_Hegner@dom.com (Joseph Hegner)
junichi_uchiyama@mnes-us.com (Junichi Uchiyama)
KSutton@morganlewis.com (Kathryn M. Sutton)
kwaugh@impact-net.org (Kenneth O. Waugh)
lynchs@gao.gov (Sarah Lynch - Meeting Notices Only)
Margaret.Bennet@dom.com (Margaret Bennet)
maria.webb@pillsburylaw.com (Maria Webb)
mark.beaumont@wsms.com (Mark Beaumont)
matias.travieso-diaz@pillsburylaw.com (Matias Travieso-Diaz)
media@nei.org (Scott Peterson)
mike_moran@fpl.com (Mike Moran)
mwetterhahn@winston.com (M. Wetterhahn)
nirsnet@nirs.org (Michael Mariotte)
patriciaL.campbell@ge.com (Patricia L. Campbell)
paul.gaukler@pillsburylaw.com (Paul Gaukler)
Paul@beyondnuclear.org (Paul Gunter)
pshastings@duke-energy.com (Peter Hastings)
RJB@NEI.org (Russell Bell)
RKTemple@cpsenergy.com (R.K. Temple)
roberta.swain@ge.com (Roberta Swain)
rod.krich@constellation.com (Mr. Rod Krich)
ronald.hagen@eia.doe.gov (Ronald Hagen)
Ronda.Daflucas@areva.com (Ronda Daflucas)
sandra.sloan@areva.com (Sandra Sloan)

sfrantz@morganlewis.com (Stephen P. Frantz)
steven.hucik@ge.com (Steven Hucik)
tkkibler@scana.com (Tria Kibler)
tom.miller@hq.doe.gov (Tom Miller)
trsmith@winston.com (Tyson Smith)
VictorB@bv.com (Bill Victor)
vijukrp@westinghouse.com (Ronald P. Vijuk)
waraksre@westinghouse.com (Rosemarie E. Waraks)
wayne.marquino@ge.com (Wayne Marquino)
whorin@winston.com (W. Horin)