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CENG
a joint venture of
Constellation Energy eDF

CALVERT CLIFFS
NUCLEAR POWER PLANT

August 9, 2010

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Response to Request for Additional Information – Proposed Transition from Westinghouse to AREVA Nuclear Fuel

REFERENCES:
(a) Letter from Mr. D. V. Pickett (NRC) to Mr. G. H. Gellrich (CCNPP), dated July 12, 2010, Request for Additional Information Re: Proposed Transition from Westinghouse to Areva Nuclear Fuel - Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 – (TAC Nos. ME2831 and ME2832)
(b) Letter from Mr. T. E. Trepanier (CCNPP), to Document Control Desk (NRC), dated November 23, 2009, License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel

Reference (a) requested additional information related to the proposed license amendment to support the transition from Westinghouse to AREVA Advanced CE-14 High Thermal Performance fuel. Attachment (1) contains the response to that request. This attachment contains information that is proprietary to AREVA, therefore it is accompanied by an affidavit signed by AREVA, owner of the information (Attachment (2). The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission, and addresses, with specificity, the considerations listed in 10 CFR 2.390(b)(4). Accordingly, it is requested that the information that is proprietary to AREVA be withheld from public disclosure. The non-proprietary version of the responses is provided in Attachment (3).

This response does not change the No Significant Hazards determination previously provided in Reference (b).

A001
NRR

Should you have questions regarding this matter, please contact Mr. Douglas E. Lauver at (410) 495-5219.

Very truly yours,

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, George H. Gellrich, being duly sworn, state that I am Vice President - Calvert Cliffs Nuclear Power Plant, LLC (CCNPP), and that I am duly authorized to execute and file this License Amendment Request on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of St. Mary's, this 9th day of August, 2010.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires:

March 1, 2011
Date

GHG/PSF/bjd

Attachments: (1) Request for Additional Information Re: Realistic Large Break Loss-of-Coolant Accident (Proprietary Version)
Enclosure: (1) Appendix B, Radial Temperature Distribution ANF-90-145(P)(A), Volume 1, Revision 0
(2) AREVA Proprietary Affidavit
(3) Request for Additional Information Re: Realistic Large Break Loss-of-Coolant Accident (Non-Proprietary Version)
Enclosure: (1) Appendix B, Radial Temperature Distribution ANF-90-145(P)(A), Volume 1, Revision 0

cc: **(Without Attachment 1)**

D. V. Pickett, NRC
M. L. Dapas, NRC
Resident Inspector, NRC
S. Gray, DNR

ATTACHMENT (2)

AREVA PROPRIETARY AFFIDAVIT

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the report ANP-2834Q(P), Revision 000, entitled "Calvert Cliffs Nuclear Plant Unit 1 Cycle 21 & Unit 2 Cycle 19 Realistic Large Break LOCA Summary Report," dated July 2010 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(c) and 6(e) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 19th

day of July 2010.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/10
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2010

ATTACHMENT (3)

REQUEST FOR ADDITIONAL INFORMATION RE: REALISTIC LARGE BREAK LOSS-OF-COOLANT ACCIDENT (Non-Proprietary Version)

Calvert Cliffs Nuclear Power Plant, LLC
August 9, 2010

NRC RAI 1.a.i.

9. *Please provide more information about the management of the fuel thermal conductivity degradation issue identified in NRC Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation." Specifically:*

   a. *ANP-2834(P), Page 1-3, states, "For each specific time in cycle, the fuel conditions are computed using RODEX3A prior to starting the S-RELAP5 portion of the analysis. A steady state condition for the given time in cycle using S-RELAP5 is established. A base fuel centerline temperature is established in this process. Then two-transformation adjustment to the base fuel centerline temperature is computed. The first transformation is a linear adjustment for an exposure of 10 MWd/MTU or higher. In the new process, a polynomial transformation is used in the first transformation instead of a linear transformation." Please clarify the following:*

      i. *Explain how the fuel pellet radial temperature profile is computed.*

CCNPP Response to RAI 1.a.i.

The RODEX3 topical report, Reference 4, Appendix B details the calculation of the radial temperature distribution (See Enclosure 1).

NRC RAI 1.a.ii.

ii. *Explain which code is used to calculate this profile, both for initial conditions and through the postulated accident.*

CCNPP Response to RAI 1.a.ii.

The RODEX3A fuel model was incorporated into the S-RELAP5 code to calculate fuel response for transient analyses. The S-RELAP5/RODEX3A model does not calculate the burnup response of the fuel. Instead, fuel conditions at the burnup of interest are transferred via a binary data file from RODEX3A to S-RELAP5, establishing the initial state of the fuel prior to the transient. The data transferred from RODEX3A describes the fuel at zero power. A steady-state S-RELAP5 calculation is required to establish the fuel state at power. The transient fuel pellet radial temperature profile is computed by solving the conduction equation of S-RELAP5. Material properties are taken from RODEX3A and incorporated into S-RELAP5.

NRC RAI 1.a.iii.

iii. *Explain whether the polynomial transformation is applied merely to the centerline temperature, or to the entire pellet temperature.*

CCNPP Response to RAI 1.a.iii.

The adjustment is applied to the entire fuel pellet. The polynomial transformation provides a bias adjustment to the fuel centerline temperature. A sampled parameter provides a random assessment and adjustment of the centerline temperature uncertainty. These are combined and the total adjustment is achieved by iterating a multiplicative adjustment to the fuel thermal conductivity until the desired fuel centerline temperature is reached. Thus, the adjustment is applied to the entire pellet but with variance according to the nodal pellet temperature and the distance from the node to the pellet surface.

NRC RAI 1.b.

b. *Provide additional information to describe the polynomial transformation. Summarize data used to develop the polynomial transformation and discuss consideration of applicable uncertainties.*

CCNPP Response to RAI 1.b.

Original:

The first transformation applies a linear adjustment if the analysis is being performed for fuel which has an exposure of 10 MWD/MtU or higher.

[ ]

Where:

T_{new} = New fuel centerline temperature (°F)

B = Burnup (GWD/MtU or MWD/KgU)

$T_{original}$ = Base fuel centerline temperature (°F)

The second transformation adds a value which is determined from a random sampling range of [ ] from a Gaussian distribution.

Revised:

For the 1st transformation, instead of adding the linear transformation after 10 MWD/MtU, a different form of correction factor should be applied.

[ ]

where:

T_{new} = New fuel centerline temperature (K)

B = Burnup (GWD/MtU or MWD/KgU)

$T_{original}$ = Base fuel centerline temperature (K)

The second transformation remains the same as the original method.

The justification for this process comes from analyzing the fuel rod database used for the development of RODEX4. A calculation was created that used RODEX3A to compute fuel centerline temperatures using all the points in the RODEX4 database (Reference 3). Three cases (cases 432R2, 432R6, and 597R8) were not used from the RODEX4 database. Case 597R8 was not needed for the present application. Cases 432R2 and 432R6 were rod studies that were not configured in a manner which are to be used in these types of comparisons. These fuel rods were not representative of commercial PWR fuel.

The fractional difference between the RODEX3A calculated results and the data in the RODEX4 database was calculated. The temperature fraction for each point in the database was computed as follows.

$$T_{fraction} = \frac{T_{rodex3A} - T_{data}}{T_{rodex3A}}$$

where:

$T_{fraction}$ = Delta fractional temperature of computed to data (K)

$T_{rodex3A}$ = Temperature computed by RODEX3A (K)

T_{data} = Temperature from the RODEX4 database (K)

A polynomial curve fit was generated from this data set. Figure 2-1 is the plot of this data and the curve fit.

[ ]

Figure 2-1 Fractional Fuel Centerline Temperature Delta Between RODEX3A and Data

The curve fit was then inverted about the zero axis. This new polynomial correction is applied regardless of fuel exposure. Figure 2-2 shows how the new correction factor changes the results. The data for this plot were created by subtracting $T_{rodex3A}$ from T_{data} as a function of burnup.

[ ]

Figure 2-2 Fuel Centerline Temperature Delta of RODEX3A Calculations to Data (original and new correlation)

The new fuel centerline temperatures no longer have a bias off of the zero error line. The approach to use a fractional based correction algorithm was requested by the NRC. Based on the plot of T_{rodex} - T_{data}, the uncertainty used in the original basis does not need to be altered. No specific temperature bias is identified in the uncertainty of the data. Therefore retaining the current Gaussian distribution sampled from [ ] is acceptable.

NRC RAI 2

10. *The current licensing basis, deterministic loss of coolant accident (LOCA) analysis concluded that the limiting condition did not involve a worst-case single failure, but rather that it depended on injected coolant delivered in such a condition that the resultant containment environment, specifically the lower containment pressure, contributed to the limiting peak cladding temperature (PCT). Please provide information describing how this potentially limiting scenario was evaluated using the proposed best-estimate methodology.*

CCNPP Response to RAI 2

AREVA's NRC-approved RLBLOCA evaluation model prescribes the worst-case single failure as the loss of one complete train of ECCS pumped injection. The evaluation model also conservatively prescribes:

(4) The use of full containment sprays without a time delay at the minimum technical specification temperature;

(5) Pumped ECCS injection at the maximum technical specification temperature; and

(6) Sampling of the containment volume (indirectly sampling containment pressure) from its nominal volume to its empty volume.

Deviations would require deviating from the approved evaluation model. Studies comparing several failure assumptions, including a no-failure assumption (Reference 1, RAI responses #26 and #111), validate that the ECCS and containment modeling of the AREVA methodology trends to the conservative. The containment pressure response is indirectly ranged by sampling the containment volume. The possible range to be sampled from was 1.989E+6 to 2.148E+6 ft^3 for Calvert Cliffs. Figure 4-21 in Enclosure 1 of Reference 2 shows that there is little sensitivity between containment volume (indirectly pressure) and PCT for a statistical application. Thus, the methodology is responsive to the goal of a realistic, yet slightly conservative, evaluation.

NRC RAI 3

11. *Please provide additional information summarizing the single-failure evaluation performed to establish compliance with General Design Criterion (GDC) 35 requirements. Identify which single failures were considered, discuss whether each failure was evaluated or explicitly analyzed, and for those failures which were explicitly analyzed, explain whether they were analyzed in a reference case or explicitly as a part of the statistical methodology. Also discuss the basis for the single failure evaluation. For example, were single failures considered as a matter of experience with CCNPP specifically, or with a generic Combustion Engineering nuclear steam supply system design?*

CCNPP Response to RAI 3

Section 4.9 in Enclosure 1 of Reference 2 discusses GDC 35. The single failure prescribed by Reference 1 is a loss of one train of ECCS (See response to RAI #2 above).

AREVA satisfies the GDC-35 criteria by running one set of 59 cases with offsite power available and one set of 59 cases with no offsite power available. The sampling seeds are held constant between these two case sets, with the only difference being the offsite power assumption. The case set that produces the most limiting PCT is reported, for Calvert Cliffs, this was no offsite power available. Figure 3-22 in Enclosure 1 of Reference 2 displays the results from the two case sets.

NRC RAI 4

12. *Page 3-6 states, "the RLBLOCA transients are of sufficiently short duration that the switchover to sump cooling water (i.e., RAS) for ECCS pumped injection need not be considered." For the limiting transient, the collapsed core liquid level from 200-350 seconds appears to trend downward (Figure 3-20). An indication of stable and increasing collapsed liquid level would substantiate the statement quoted above, but this is not the case for Figure 3-20. Is the SRELAP-5 model of the limiting case capable of generating credible results after 350s? If so, please provide results for a period of the transient sufficient to demonstrate that the core collapsed liquid levels are stable or increasing.*

CCNPP Response to RAI 4

S-RELAP5 is capable of generating credible RLBLOCA results beyond 350 seconds. Figure 3-11 in Enclosure 1 of Reference 2 shows the PCT independent of elevation quenched at approximately 245 seconds. In essence, the PCT independent of elevation means the entire core is quenched, as this plot represents the highest PCT for all hot rods in the core, which would bound the hot assembly, surrounding assemblies, average core and peripheral core. The case terminated at 346.9 seconds, which is 100 seconds after the core quenched. The end of the transient in Figure 3-20 (Enclosure 1 of Reference 2) shows an increase in liquid level and from 200 seconds onward it remains relatively constant.

Figure 3-16 in Enclosure 1 of Reference 2 shows the ECCS flow is at a total constant of about 270 lbm/sec at transient termination. Comparing this to the break flow in Figure 3-12 (Reference 2, Enclosure 1), shows the ECCS flow is greater than the break flow. Figure 3-18 (Reference 2, Enclosure 1) displays the downcomer liquid level remaining relatively constant at a liquid level of approximately 17 ft for the last 100 seconds of the transient. Figure 2-3, below, plots the reactor vessel liquid mass showing that the vessel inventory is increasing.

[ ]

Figure 2-3 Reactor Vessel Liquid Mass (lbm) vs Time (sec)

All evidence indicates that the core will remain quenched and that the reactor vessel inventory will continue to gradually increase. Therefore, it is unnecessary to extend the run time for the limiting case to resolve the collapsed liquid level any more based on the supporting evidence in the figures discussed above.

NRC RAI 5

13. *Please provide information to enable comparison between Technical Specifications (TS) requirements and analytic input parameters for Pressurizer Level. The TS requirement is given in inches and the input parameters are specified in percent span.*

CCNPP Response to RAI 5

Technical Specification 3.4.9, Pressurizer, states:

"The pressurizer shall be OPERABLE with:

c. Pressurizer water level ≥ 133 inches and ≤ 225 inches; and
d. Two banks of pressurizer heaters OPERABLE with the capacity of each bank ≥ 150kW and capable of being powered from an emergency power supply."

Pressurizer level indication is provided by level instruments (1(2)-LI-103). The calibrated range, or span, of these instruments is 0 to 360 inches of water.

The sampled range for the liquid level uncertainty in the pressurizer was 32.2% to 67.2% of the span, which corresponds to a water level range of 115.9 inches to 241.9 inches. Therefore, the sampled range encompasses the Technical Specification limits.

NRC RAI 6

14. *Please provide discussion to confirm that the assumed 60°F containment temperature is an acceptable minimum without a TS requirement.*

CCNPP Response to RAI 6

Air temperatures in containment are logged every 12 hours and if they exceed the predetermined limits, corrective actions are taken. In the case of a minimum containment temperature, 70°F is the lower limit. If the temperature drops below that limit, actions are taken to reduce the cooling in the containment by either reducing cooling water flow to the containment air coolers or by shutting off the containment air coolers.

Plant data from January, 2006 to the end of June, 2010 was reviewed to ensure that the minimum and maximum temperatures are routinely achievable. The minimum measured cavity temperatures of 68°F (Unit 1) and 65.7°F (Unit 2) include margin to accommodate uncertainty and support the assumed minimum value of 60°F.

NRC RAI 7

15. *The TS minimum for the refueling water storage tank (RWST) temperature is 45°F. Previous, deterministic analyses demonstrated that minimum safety injection temperatures resulted in a limiting PCT. In light of this information, please explain why a minimum RWST temperature case was not evaluated, or if a minimum RWST temperature case was evaluated, please summarize the evaluation and discuss its conclusions.*

CCNPP Response to RAI 7

As stated in the response to RAI #2, the NRC-approved RLBLOCA evaluation model, Reference 1, prescribes use of the maximum temperature for the ECCS pumped injection (100°F) and use of the minimum temperature (40°F not 45°F was used) for the containment sprays. While inconsistent, the choice of the two temperatures is conservative. AREVA's RLBLOCA analysis complies with, and does not deviate from, the approved evaluation model requirements

NRC RAI 8

16. *As noted in Section 1 of ANP-2834(P), deviations from the approved RLBLOCA evaluation model (EMF-2103(P)(A), Revision 0) are necessary to demonstrate compliance with 10 CFR 50.46 requirements. Please provide a commitment to adhere to the deviations noted in Section 1 of ANP-2834(P)(A) until such time as:*

    a. *AREVA develops a new revision of EMF-2103,*
    b. *The NRC approves the new revision of EMF-2103, and*
    c. *CCNPP implements the new, NRC-approved revision of EMF-2103.*

    The commitment should include language to indicate that meeting Conditions a, b, and c, above, or submitting a license action request to implement a different evaluation method, will obviate the need for this commitment.

CCNPP Response to RAI 8

CENG commits to the following:

Calvert Cliffs will adhere to the deviations noted in Section 1 of Enclosure 1 of Reference 2 until such time as:

- AREVA develops a new revision of EMF-2103,
- The NRC approves the new revision of EMF-2103, and
- CCNPP implements the new, NRC-approved revision of EMF-2103.

This commitment will terminate when the above items are met or a license amendment is approved to permit the use of a different evaluation method to replace Enclosure 1 of Reference 2.

REFERENCES:

1. EMF-2103(P)(A), Revision 0, Realistic Large Break LOCA Methodology, Framatome ANP, Inc., April 2003
2. Letter from Mr. T. E. Trepanier (CCNPP), to Document Control Desk (NRC), dated November 23, 2009, License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel
3. EMF-2994(P), Revision 4, RODEX4: Thermal-Mechancial Fuel Rod Performance Code Theory Manual, December 2009
4. ANF-90-145(P)(A), RODEX3 Fuel Thermal-Mechanical Response Evaluation Model, April 1996

ENCLOSURE (1)

Appendix B, Radial Temperature Distribution

ANF-90-145(P)(A), Volume 1, Revision 0

(Non-Proprietary)

Note: Although the document that this Enclosures comes from is proprietary, the contents of this Enclosure contains no proprietary information.

APPENDIX B
RADIAL TEMPERATURE DISTRIBUTION
-(CYLTEM SUBROUTINE)-

1.0 EVALUATION OF THE HEAT CONDUCTION EQUATION

Assuming no axial or azimuthal heat conduction, the heat conduction equation is:

$$\nabla \cdot (K \cdot \nabla T) = \frac{1}{r} \cdot \frac{d}{dr}\left(r \cdot K \cdot \frac{dT}{dr}\right) = -Q \tag{B-1}$$

where:

- r = radial dimension
- K = thermal conductivity
- T = temperature
- Q = volumetric heat generation per unit length

Knowing the temperature, T_J, and the heat flux, ϕ_J, at the outer regional boundary $r = R_J$, integration of the conduction equation yields:

$$R_J \cdot \phi_J - r \cdot \left(-K \cdot \frac{dT}{dr}\right) = \int_r^{R_J} Q \cdot x dx \tag{B-2}$$

where:

$$\phi_J = -K \cdot \frac{dT}{dr} \bigg|_{R_J}$$

Integration of Equation B-2 for a temperature dependent thermal conductivity yields:

$$\int_{T_J}^{T(r)} K \cdot dt = R_J \cdot \phi_J \cdot \log (R_J / r) - \int_r^{R_J} \frac{dy}{y} \int_y^{R_J} Q \cdot x dx \tag{B-3}$$

This relation is used to calculate the temperature T_{J-1} at the inner boundary of the annulus (R_{J-1}) assuming that the power distribution (Q_{J-1}) in the annulus is constant and that the thermal conductivity varies linearly with temperature. The non-linear temperature functions for the thermal conductivity are used to determine the appropriate linear variation over the annulus, which is approximated by:

$$K(T) = K_J + \Delta K \cdot (T - T_J) / \Delta TE \tag{B-4}$$

where:

$K_J = \kappa(T_J)$
$\Delta K = \kappa(T_J + \Delta TE) - K_J$
$\kappa(T)$ = non-linear function representing thermal conductivity
ΔTE = estimated temperature rise over ring assuming conductivity K_J

The temperature rise, $\Delta T = T_{J-1} - T_J$ is found by substituting in a constant heat generation rate Q_{J-1}, $r = R_{J-1}$ and Equation B-4 into Equation B-3. The resultant relation obtained by performing these operations is:

$$K_J \cdot \Delta T + \frac{\Delta K}{2 \cdot \Delta TE} \cdot \Delta T^2 = \left(R_J \cdot \phi_J - Q_{J-1} \cdot R_J^2 / 2 \right) \cdot \log \left(R_J / R_{J-1} \right) + \left(Q_{J-1} / 4 \right) \cdot \left(R_J^2 - R_{J-1}^2 \right) = PF_{J-1} \tag{B-5}$$

The right side of Equation B-5 can be determined for known power distributions.

The estimated temperature rise used for calculating the linear variation in conductivity is:

$$\Delta TE = PF_{J-1} / K_J \tag{B-6}$$

and the solution of Equation B-5 for the temperature rise is:

$$\Delta T = \Delta TE \cdot \left(-1 + \sqrt{1 + 2 \cdot \Delta K / K_J} \right) / \left(\Delta K / K_J \right) \tag{B-7}$$

Equation B-7 is indeterminant for constant conductivity, thus when $|\Delta K/K| < 0.1$, the second order expansion of Equation B-7 is used:

$$\Delta T = \Delta TE \cdot [\ 1 - 0.5 \cdot (\ \Delta K / K_J\ ) + 0.5 \cdot (\Delta K / K_J\ )\ ]^2 \tag{B-8}$$

These numerical evaluations are performed in the CYLTEM subroutine. The calculations are performed for one annular ring and the calling program for CYLTEM must supply PF_{J-1}, the material designation, outer surface temperature, and material parameters for the thermal conductivity subroutine. .

2.0 EVALUATION OF THE THERMAL CONDUCTIVITY

The thermal conductivity (Appendix M) of the fuel pellet depends upon temperature, material type, void fraction and composition (weight fractions of urania, gadolinia, plutonium, and rare earths). These properties, other than temperature, are assumed constant within a radial region, but vary with each radial region. They also vary with exposure. For example, rare earths are created by the fission process. The rare earths are treated as if they are gadolinia.

The void fraction used to compute the thermal conductivity of the fuel pellet is composed of four components:

VOIFD This void fraction accounts for voids created by incomplete densification of the fuel pellet.

FOISG This void fraction accounts for voids created by solid and gaseous bubbles which are not accommodated by existing voids in the fuel matrix.

VOICR This void fraction accounts for voids created by the displacement of circumferential cracks (see Appendix J, Term v_c). The void fraction is computed by dividing v_c by the mean radius of the radial region to convert the displacement into a specific volume.

VOIGP This void fraction accounts for the separation of pellet fragments which displace the pellet surface into the gap region. This displacement increases the thermal resistance of the pellet. Fuel temperatures[B1], measured with thermocouples placed in various radial locations, showed large temperature gradients that could be explained only by reduced pellet thermal conductivities. Grain growth measurements for irradiated fuel[B2], and derived temperature distributions from those measurements, also supported lower pellet thermal conductivities in the lower temperature regions of the fuel. The offset thermocouple tests[B1] further showed that the displacements of pellet fragments into the gap had both hot and cold gap contributions. The following relationship used to model this contribution in RODEX3 is:

$$VOIGP = 0.04 + (5.0 \cdot TDGP + 1.4 \cdot COGAP) \cdot [2 \cdot r_{op}/(r_{op}^2 - r_{ip}^2)]$$

where:

VOIGP = equivalent void contribution due to gap

TDGP = hot gap separation minus the effective separation used in the conductance model, (in)

COGAP = cold gap evaluated for the current irradiated state, (in)

r_{ip} = radius of pellet annulus, (in)

r_{op} = outer radius of pellet, (in)

The effective void fraction, VOIDV, is the sum of the four void contributions, adjusted to reduce their effect on the photon contribution to the thermal conductivity at high temperatures[B3]:

$$VOIDV = (VOIFD+VOISG+VOICR+VOIGP) / \{0.75+EXP\ [(T-1859)/360]\}$$

where:

T = mean radial region fuel temperature, (F)

The void fraction used to compute the thermal conductivity (VOIDX), based on a fuel volume that includes the voids, is:

$$VOIDX = \text{minimum of } [0.5 \text{ and } VOIDV/(1.0 + VOIDV)].$$

3.0 REFERENCES

B1 R.W. Garner, D.T. Sparks, R.H. Smith, P.H. Klink, D.H. Schwieder, and P.E. MacDonald, "Gap Conductance Test Series-2, Test Results Report for Tests GC 2-1, GC 2-2, and GC 2-3", NUREG/CR-0300, TREE-1268, November 1978.

B2 C. Bagger, "Radial Temperature Profiles in ANF Fuel", The Third RISØ Fission Gas Project, RISØ-FGP-M38, April 1989.

B3 D.L. Hagrman, G.A. Reymann, R.E. Mason, MATPRO-Version 11 (Rev.2), NUREG/CR-0479, TREE-1280, Rev. 2, Aug. 1981.