VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

May 9, 2003

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

Serial No. 03-314
NLOS/ETS
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-77

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
REQUEST FOR ADDITIONAL INFORMATION
PROPOSED TECHNICAL SPECIFICATIONS CHANGES AND EXEMPTION REQUEST
TO USE FRAMATOME ANP ADVANCED MARK-BW FUEL

In a March 28, 2002 letter (Serial No. 02-167), Virginia Electric and Power Company (Dominion) requested an amendment to Facility Operating License Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, and associated exemptions from 10 CFR 50.44 and 10 CFR 50.46. The amendments and associated exemptions will permit North Anna Units 1 and 2 to use Framatome ANP Advanced Mark-BW fuel. This fuel design has been evaluated by Framatome and Dominion for compatibility with the resident Westinghouse fuel and for compliance with fuel design limits. In several telephone calls in March and April 2003 the NRC staff requested additional information to complete the review of the proposed Technical Specification changes and fuel transition program. Attachment 1 to this letter provides the requested information as well as revised pages for the March 28, 2002 evaluation report requested by the NRC staff. Please substitute these pages into the March 28, 2002 evaluation report to complete your review.

Attachment 1 contains Framatome ANP proprietary information. Attachment 3 is a signed affidavit from Framatome ANP, the owner of the information, which provides the basis for classifying information in Attachment 1 as proprietary. In addition, the basis for classifying the additional revised information in Attachment 1 as proprietary was addressed (pursuant to 10 CFR 2.790(b)(1)) in the previous submittal and application for withholding provided in our March 28, 2002 letter and remains applicable to this submittal. To conform to the requirements of 10 CFR 2.790 concerning the protection of proprietary information, the proprietary information in Attachment 1 is contained within brackets. A non-proprietary redacted version of Attachment 1 is also provided in Attachment 2. Where the proprietary information has been deleted in the non-proprietary version, only the brackets remain. Accordingly, it is requested that the information that is proprietary to Framatome ANP in Attachment 1 be withheld from public disclosure in accordance with 10 CFR 2.790 of the Commission's regulations.

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In order to support use of Framatome Advanced Mark-BW fuel in North Anna Unit 2, Cycle 17, we request that NRC complete their review and approval of the amendment and exemptions by September 30, 2003. We appreciate your considerations of our technical and schedular requests. If you have any questions or require additional information, please contact us.

Very truly yours,

Leslie N. Hartz
Vice President - Nuclear Engineering

Commitments made in this letter: None

Attachments:

1. Request for Additional Information and Revised March 28, 2002 Evaluation Report Pages (Proprietary Version)
2. Request for Additional Information and Revised March 28, 2002 Evaluation Report Pages (Non-Proprietary Version)
3. Framatome ANP Affidavit

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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President - Nuclear Engineering, of Virginia Electric and Power Company. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 9th day of May, 2003.

My Commission Expires: March 31, 2004.

Maggie McClure
Notary Public

(SEAL)

Attachment 2

NON-PROPRIETARY VERSION

Response to Request for Additional Information

and

Revised Pages for March 2002 Evaluation Report

Framatome Fuel Transition Program
Technical Specification Change

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 and 2

Three separate requests for additional (RAI) information were made by the NRC to support staff review of the North Anna license amendment to use Framatome Advanced Mark-BW fuel. A response to each item is provided below, referenced to each of the three separate RAIs.

RAI Source

1. Facsimile from Stephen Monarque (USNRC), to Tom Shaub (Dominion) on March 31, 2003 entitled "North Anna Power Station, Units 1 and 2, Facsimile Transmission of Questions for Proposed Technical Specifications and Exemption Request, Use of Framatome ANP Advanced Mark-BW Fuel (TACS MB4700, MB4701, MB4714, AND MB4715."

2. Telephone discussion between Stephen Monarque (USNRC) and Tom Shaub (Dominion) on April 2, 2003, transmitting questions regarding the Spent Fuel Pool design basis and the proposed technical specifications and exemption request, for use of Framatome ANP Advanced Mark-BW Fuel.

3. Telephone discussion between Stephen Monarque (USNRC) and Tom Shaub (Dominion) on April 4, 2003, transmitting EMEB questions regarding fuel mechanical items for proposed technical specifications and exemption request, for use of Framatome ANP Advanced Mark-BW Fuel.

RAI SOURCE NUMBER 1 QUESTIONS

Q-1: (Page 2 of March 28, 2002 submittal) The reactor core SL has two different fuel centerline melt correlations based on the different vendor fuel types. Please provide the data used to develop and justify the Framatome fuel melt temperature line.

RESPONSE:

The fuel melt correlation for the Advanced Mark-BW fuel is shown in Appendix I of the NRC approved topical report BAW-10162P-A (Reference 1.1). The correlation reported is Equation I-3, converted to degrees Fahrenheit.

Reference 1.1: BAW-10162P-A, TACO3 – Fuel Pin Thermal Analysis Computer Code, October 1989.

Q-2: (Page 25) Under the control rod drop times section, the submittal makes a comparison between the Advanced Mark-BW and Westinghouse LOPAR. Have you used the LOPAR fuel design previously? And if so, what type of control rod drop times were experienced?

RESPONSE:

The Westinghouse LOPAR fuel design was the original licensing basis for the North Anna units. The fresh fuel loaded into North Anna Unit 1 for Cycles 1 through 8, and into North Anna Unit 2 for Cycles 1 through 7 was of this design. LOPAR fuel assemblies continued to be irradiated at North Anna though Cycle 14 in Unit 1 and Cycle 12 in Unit 2. LOPAR assemblies were last used in core locations containing control rods in North Anna 1 Cycle 11 and North Anna 2 Cycle 10.

The thimble tubes of the current North Anna Improved Fuel (NAIF) design have smaller diameter than the LOPAR design, which slightly increases the control rod insertion times. Accordingly, when the NAIF fuel design was introduced, the North Anna Units 1 and 2 Operating Licenses were amended (Amendments 139 and 122, respectively) to increase the allowable control rod drop time in the Technical Specifications from 2.2 seconds to 2.7 seconds to allow for the effects of the reduced thimble tube diameter. Control rod drop times measured during beginning of cycle hot rod drop testing for recent cycles are generally about 0.1 to 0.3 seconds longer than for the initial cycles at North Anna Units 1 and 2.

Q-3: (Page 26) The maximum grid impact forces for the SSE conditions are referred to. Please state what the maximum allowable grid impact force was and how it relates to the allowable elastic limit.

RESPONSE:

The [ ] spacer grid elastic limits were determined by impact tests using NRC approved methods described in Reference 3.1. The maximum grid impact forces for the SSE conditions were from peripheral fuel assemblies in the shortest rows, and were below the allowable elastic limits. [

]

The maximum grid impact forces for Advanced Mark-BW Grids under SSE conditions are:

Intermediate Spacer Grid: Maximum Impact Force = [ ] lb(≤ [ ] lb elastic limit)
Mid Span Mixing Grid: Maximum Impact Force = [ ] lb (≤ [ ] lb elastic limit)

Reference 3.1: Addendum 1 to BAW-10133P-A, Revision 1, Mark-C Fuel Assembly LOCA-Seismic Analyses, October 2000.

Q-4: (Page 26) When will the LOCA evaluation of Section 7.0 be submitted?

RESPONSE:

Dominion letter Serial No. 03-313, dated May 6, 2003 submitted results of the realistic large break LOCA analysis (RLBLOCA) for North Anna Unit 2. Two subsequent revisions of the LOCA documentation in Section 7.0 are scheduled: 1) small break LOCA (SBLOCA) analysis for Unit 2, to be submitted by May 30, 2003 and 2) RLBLOCA and SBLOCA analysis results for Unit 1, to be submitted by July 30, 2003.

Q-5: (Page 27) The submittal states that the TACO3 code is only licensed to 60,000 MWD/MTU and that North Anna has a peak pin burnup limit of 60,000 MWD/MTU. Please clarify these statements.

RESPONSE:

The statement on page 27 "the TACO3 is only licensed to 60,000 MWD/MTU" is incorrect. Although page xxxii of the SER for the TACO3 code (Reference 5.1) states the NRC approved use of the code to 60,000 MWD/MTU, the NRC has granted extended use of the TACO3 code to 62,000 MWD/MTU in Reference 5.2. The acceptable use of TACO3 to 62,000 MWD/MTU is reflected in the information documented in Reference 5.3. Although the TACO3 code is approved for use to 62,000 MWD/MTU, the Advanced Mark-BW fuel design is currently limited to a maximum approved rod average burnup of 60,000 MWD/MTU (Reference 5.3). Approval of Reference 5.4 will extend the limit to 62,000 MWD/MTU for the Advanced Mark-BW fuel design.

The 60,000 MWD/MTU North Anna limit refers to a limitation resulting from a NRC staff letter received in relation to a previous fuel enrichment licensing change. This burnup limit is discussed in the UFSAR (Section 4.3.1.1), but does not appear in Technical Specifications. The restriction appears in the following correspondence: Letters from B. C. Buckley and L. B. Engle (U.S. NRC) to W. L. Stewart (Virginia Electric and Power Company), "Surry, Units 1 and 2, and North Anna, Units 1 and 2 - Removal of 45,000 MWD/MTU Batch Average Burnup Restriction (TAC Nos. M87767, M87768, M87812, and M87813)," December 14, 1993 and April 20, 1994.

References:

5.1 BAW-10162P-A, TACO3 – Fuel Pin Thermal Analysis Computer Code, October 1989.
5.2 Letter, Robert C. Jones (NRC) to J. H. Taylor (B&W Nuclear Technologies), January 11, 1996.
5.3 BAW-10186P-A, Revision 1, Extended Burnup Evaluation, April 2000.
5.4 BAW-10186P, Revision 2, Supplement 1 to BAW-10186P-A, Rev. 1, November 2001

Q-6: (Page 27) The fuel rod cladding stress is stated as using conservative values. Please define what condition is meant by conservative values for all the cladding parameters listed.

RESPONSE:

The parameters which are inputs to the fuel rod cladding stress analysis are given below. The conservative nature of each parameter given is discussed. The approved methodology for the fuel rod stress analysis is contained in BAW-10227P-A.

Cladding wall thickness: For the calculation of the cladding stress state, the minimum cladding wall thickness allowed for the fuel rod design is used to calculate the stress state in the cladding. This wall thickness value is based on the design drawing dimensions and tolerance. For the Advanced Mark-BW design the minimum allowable wall thickness is [ ].

The use of the minimum cladding wall thickness in the calculation of the cladding stress state results in the largest stresses and is therefore conservative.

Cladding oxide: For the calculation of the cladding stress state, the cladding wall thickness is further reduced by using values of cladding oxide levels at end of life conditions which are greater than predicted for the rod design. For the Advanced Mark-BW stress calculations, a cladding oxide thickness of [ ] is used to bound the cladding corrosion levels of the M5™ cladding.

The use of a bounding value of cladding oxide to reduce the cladding wall thickness is conservative because the use of the thinner wall thickness in the stress calculation will produce higher levels of predicted cladding stress.

External pressure: For the cladding stress calculation, a value for the external system pressure corresponding to a value greater than the system design limit is used. For the application of the Advanced Mark-BW design at the North Anna Units, a value of external pressure of [ ] is used to bound the allowable system design maximum pressure of 2500 psia for the calculation of compressive stresses at BOL conditions. For the calculation of tensile cladding stresses at EOL conditions an external pressure equal to the system pressure of [ ] is used.

The use of a bounding value of external pressure for the fuel rod stress calculation results in higher calculated cladding stress values and is therefore conservative.

Internal rod pressure: A value of internal rod pressure that corresponds to a minimum value at BOL conditions and a maximum value at EOL conditions is used for the calculation of the fuel rod cladding stresses. An internal pressure value of [ ] at BOL conditions is less than the hot BOL fuel rod internal pressure. A value of [ ] at EOL conditions is [ ] greater than the system pressure.

These values bound the extremes of the fuel rod internal pressure during the lifetime of the rod and therefore result in a conservative calculation of the fuel rod cladding stress state.

Differential temperature: A bounding value of the differential cladding temperature is used in the stress analysis. For the Advanced Mark-BW fuel rod design, a temperature gradient across the cladding of [ ] is used.

The use of a bounding value of differential cladding temperature results in a conservative prediction of cladding thermal stresses.

Unirradiated cladding yield strength: A minimum value of the unirradiated yield strength is used to set the fuel rod cladding stress limitations. For the Advanced Mark-BW M5 fuel rod design, the value used is [ ].

Using the minimum value for unirradiated yield stress to set the cladding stress limit produces the minimum margin to cladding yield. With irradiation, the cladding yield strength increases to values greater than the unirradiated value.

Q-7: (Page 29) In the section on fuel rod cladding strain, it discusses the calculated allowable linear heat rates and mentions that they are typically not limiting. What is meant by typically not limiting? Are they limiting some times? Under what conditions?

RESPONSE:

The fuel rod cladding strain analysis calculates the linear heat rate at which the fuel rod cladding reaches a [ ] strain under Condition I & II transient conditions.

For each cycle specific reload analysis, the linear heat rate limits for the [ ] cladding strain limits are compared to the predicted maximum linear heat generation rate allowed during the cycle for Condition I or II overpower transients. This comparison verifies that the fuel rod [ ] transient strain linear heat rate limit will not be exceeded during the cycle. The calculation of the fuel rod transient strain limits uses a bounding power history and present plant operating conditions. Using a bounding power history, it is very unlikely that the design analysis which generates the strain limits will be violated by present North Anna cycle designs. Therefore, the transient strain limits are not limiting.

In the case of a significant change in the plant operating conditions, such as a power up-rate, the calculation of the fuel rod transient strain limits would be changed to reflect the new conditions. It is again unlikely that the transient strain limits would become limiting.

Q-8: (Page 31) Under the section for fuel rod cladding creep collapse, what is the fuel rod creep collapse lifetime? How close is the burnup limit to this calculated lifetime?

RESPONSE:

The fuel rod cladding creep collapse lifetime is calculated to be greater than [ ] burnup. The calculation is performed to a burnup of [ ], without creep collapse predicted for the Advanced Mark-BW fuel rod design. This value of burnup bounds the current NRC approved 60 GWD/MTU lead rod burnup limits for the North Anna units.

Q-9: (Page 32) Under the section for the fuel rod internal pressure, it states that pin power history and axial flux shapes were generated using Framatome approved methodologies with Dominion's NRC approved codes. Could you please provide references for these approved methodologies and approved codes? Also, please clarify if these approvals were for methodologies that are code independent and if the codes were approved independent of a methodology.

RESPONSE:

The internal pressure prediction is calculated with the NRC approved TACO3 methodology with its Fuel Rod Gas Pressure Criteria. The approved method for prediction of the fuel rod internal pressure for licensing application includes [ ] in addition to the steady state axial power shapes applied over the burnup history of the fuel rod. The basis for the [ ] is provided in Appendix I of BAW-10162P-A. The NRC approved methodology does not specify use of a particular neutronics code for these calculations, so that the approval of the methodology defined in References 9.1 and 9.2 is independent of the codes used.

The pin power history and axial flux shapes were generated by Dominion using the PDQ Two Zone model, which is described in Topical Report VEP-NAF-1, "The PDQ Two-Zone Model." The PDQ Two Zone model is used as part of the Dominion reload design process and is independent of any Framatome methodology.

VEP-NAF-1 was transmitted to the NRC for review and approval via Reference 9.3 and was implemented via 10 CFR 50.59 as described in Reference 9.4. Further information about the PDQ Two Zone model and the approval process used has been provided to the NRC in References 9.5 and 9.6 during NRC review of Dominion Topical Report VEP-FRD-42 Rev. 2, "Reload Nuclear Design Methodology."

References:

9.1 BAW-10162P-A, TACO3 – Fuel Pin Thermal Analysis Computer Code, October 1989.

9.2 BAW-10183P-A, Fuel Rod Gas Pressure Criterion (FRGPC), July 1995.

9.3 Letter from W. L. Stewart (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Virginia Electric And Power Company, Surry Power Station Units 1 & 2, North Anna Power Station Units 1 and 2 Topical Report-PDQ Two Zone Model", Serial No. 90-562, October 1, 1990.

9.4 Letter from W. L. Stewart (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Virginia Electric And Power Company, Surry Power Station

Units 1 & 2, North Anna Power Station Units 1 and 2 Topical Report Use Pursuant to 10 CFR 50.59", Serial No. 92-713, November 25, 1992.

9.5 Letter from L. N. Hartz (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Virginia Electric And Power Company (Dominion), North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2 Response To Request For Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report", Serial No. 02-280, May 13, 2002.

9.6 Letter from E. S. Grecheck (Virginia Electric and Power Company) to U.S. Nuclear Regulatory Commission, "Virginia Electric And Power Company (Dominion), North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2 Response To Request For Additional Information, Dominion's Reload Nuclear Design Methodology Topical Report", Serial No. 02-662, December 2, 2002.

Q-10: (Page 42) The last paragraph under DNB Correlations states that the BWU-Z Correlation is used above the mid-span mixing grids with an enhancement factor. Please describe the enhancement factor.

RESPONSE:

The terms "enhancement factor", "direct CHF multiplication factor", and "multiplicative enhancement factor" as used in the last paragraph on page 42 and in the first paragraph on page 43 refer to a single quantity. The enhancement factor, F_{MSM}, for application to the Advanced Mark BW fuel assembly in the mid-span region is [ ]. The development of the enhancement factor is documented in BAW-10199P, Addendum 2 which received NRC approval in a letter dated March 27, 2002.

Q-11: (Page 42) Please explain how the data base for the BWU-Z correlation extends its range of application?

RESPONSE:

The BWU-Z CHF correlation is being applied for North Anna (References 11.1, 11.2, and 11.3). The statement on page 42 was provided to emphasize the broader data base (e.g., wider range of pressure, mass flux and quality) and resulting application limits of the BWU-Z CHF correlation when compared to other Framatome correlations that are NRC approved. This discussion is reflected in revised pages 43 and 43a of the Dominion evaluation report, which are attached.

References:

11.1 BAW-10199P-A, "The BWU Critical Heat Flux Correlations," August 1996.
11.2 BAW-10199P-A, Addendum 1, "The BWU Critical Heat Flux Correlations," December 2000.

11.3 BAW-10199P-A, Addendum 2, "The BWU Critical Heat Flux Correlations," June 2002.

Q-12: (Page 43) Please describe how the grid form loss coefficients are analytically determined?

RESPONSE:

The loss coefficients were calculated from an axial pressure drop profile provided by Dominion for the resident NAIF. The pressure drop profile consisted of both the component and frictional losses and the hydraulic conditions on which they were based. The NRC approved thermal-hydraulics code LYNXT (Reference 12.1) was used to match both the component and frictional losses at the established hydraulic conditions by the selection of appropriate values for loss coefficient and surface roughness. The resulting loss coefficients, both component and friction, were used in the thermal-hydraulic analyses performed with the LYNXT code supporting implementation of the Advanced Mark-BW fuel in North Anna Cores 1 & 2.

Reference 12.1: BAW-10156-A Revision 1, LYNXT: Core Transient Thermal-Hydraulic Program, August 1993.

Q-13: (Page 50) Please provide details on the maximum span-averaged cross flow velocities, including the margin between the calculated and the limit.

RESPONSE:

Inter-assembly crossflow velocities are calculated with the NRC approved thermal-hydraulics code LYNXT (Reference 13.1). Conservative mixed core configurations of each fuel assembly type are modeled to capture the maximum crossflow velocity between dissimilar assemblies. The most conservative mixed core model of fuel assemblies that have a hydraulic mismatch is a single assembly occupying the limiting core location with the remaining core locations filled with the co-resident fuel design. The results of these analyses indicate the maximum crossflow velocity occurred for a single NAIF assembly in a core of Advanced Mark-BW fuel assemblies. The maximum calculated span averaged cross flow velocity was [ ] ft/sec and provides a margin of over [ ] to the maximum allowable criterion of [ ] ft/sec.

Reference 13.1: BAW-10156-A Revision 1, LYNXT: Core Transient Thermal-Hydraulic Program, August 1993.

Q-14: (Page 50) Please provide the reference for the Framatome Statistical Core Design methodology.

RESPONSE:

Revised Page 50, with reference citation is attached.

Q-15: (Page 51) Please provide additional information on the exceptions to using a full power radial power distribution factor limit of 1.587 and how these exceptions were determined.

RESPONSE:

This question appears directed at the approach for selecting specific statepoints employed in the Advanced Mark-BW thermal-hydraulic analysis. The ultimate goal of the thermal-hydraulic analysis was to support a full-power radial power distribution factor ($F_{\Delta H}^N$) limit of 1.587 for reload cores that include the Advanced Mark-BW fuel. To support this goal, the statepoint conditions for the Advanced Mark-BW included $F_{\Delta H}^N$ values for each condition that were scaled by the ratio of the ultimate full-power $F_{\Delta H}^N$ limit to the current full-power $F_{\Delta H}^N$ limit (i.e., 1.587 / 1.490). This scaling approach effectively increases the radial peak to a magnitude that would be expected for a core designed to a full-power $F_{\Delta H}^N$ limit of 1.587. The following items represent the exceptions to the general scaling approach.

- The nominal statepoint was developed at the current full-power $F_{\Delta H}^N$ limit of 1.490 to provide a DNBR value for an update to the North Anna UFSAR.

- The rod urgent failure is assumed as a pre-condition to Condition II events and must accommodate the desired full-power $F_{\Delta H}^N$ limit. Unbounded reload values are evaluated using the Reload Nuclear Design Methodology. The rod urgent failure statepoint was developed at a $F_{\Delta H}^N$ of 1.505, a 1% difference to the current full-power $F_{\Delta H}^N$ limit of 1.490, to obtain sensitivity results for use in evaluating future reload results.

- The main steamline break (MSLB) statepoint is significantly different in power peaking and flow such that the scaling approach is not suitable. Bounding values from previous core design calculations were used for MSLB.

- The loss of flow and locked rotor statepoints were evaluated with $F_{\Delta H}^N$ values of 1.538 and 1.587. The $F_{\Delta H}^N$ value of 1.538 assumed for loss of flow and locked rotor was necessary to accommodate the transition core penalty associated with the first batch application of the Advanced Mark-BW fuel. A $F_{\Delta H}^N$ value of 1.587 was assumed to support the ultimate goal of an increase in the full-power $F_{\Delta H}^N$ value of 1.587.

It is anticipated that the thermal-hydraulic analysis results will be used to redefine the Thermal Design Limit (TDL) and $F_{\Delta H}^N$ limit after the first transition cycle. These limits will be established in accordance with existing Dominion practices for managing retained margin.

Q-16: (Page 73) Please describe how the peak ejected F_Q and ejected rod reactivity parameters were modified for the EOC HZP case.

RESPONSE:

Initial analysis runs for the EOC HZP case assumed values of ejected F_Q and ejected rod worth equal to the values in the current reference analysis. These values are 19.2 and 990 pcm, respectively, and are reported in UFSAR Table 15.4-16. The assumed input values for ejected rod F_Q and ejected rod worth were reduced using an iterative process until the analysis results for the EOC HZP case were within the revised fuel enthalpy criterion proposed in Reference 37 of the Dominion evaluation report. The calculated results of FQ and rod worth in recent core designs have been confirmed to have margin with respect to these reduced parameter values.

Q-17: (Page 114) The statement is made that the axial flow difference in the IBDCF tests are much larger than expected in North Anna Units 1 and 2 between a NAIF and an Advanced Mark-BW. Please explain why?

RESPONSE:

The IBDCF experiments examined a range of inlet velocity upsets (V_R, ratio of the inlet velocity in assembly 1/inlet velocity in assembly 2), ranging from [ ]. The [ ] represents a larger inlet flow mismatch than encountered in any Condition I or II event in any mixed core. Figure A-6 of the Reference shows the LYNXT comparison against the experimental data. The LYNXT results are within [ ] of the test data. *The first node downstream of the inlet shows a greater* difference, but as noted in the reference, this is for very small relative velocity ratios near the inlet and not representative of those encountered a few hydraulic diameters downstream in the developing flow field.

Additional flow testing of the hydraulic mismatch between fuel assemblies, as in a mixed core configuration, was performed at the Commissariat a l'Energie Atomique's (CEA) Centre d'Etudes Nucleaires de Cadarache and are called the Marignan tests. The relative velocity ratios (V_Rs) between an NAIF and an Advanced Mark-BW (called an in-reactor condition) are expected to be of the same magnitude as for the Marignan test data. The minimum Marignan test V_R, based on Figure A-10 (Reference 17.1, page 127), is approximately [ ]. In Figure A-6 (page 127) the V_Rs are larger than the expected in-reactor V_Rs for half of the test section axial length. Additionally three of the inlet V_Rs used in the benchmark are as severe or more severe than the in-reactor V_Rs. Thus, the V_Rs from the IBDCF are more limiting than what is encountered between an NAIF and Advanced Mark-BW fuel assembly. The excellent agreement between LYNXT and the test data for both the IBDF and Marignan tests demonstrates LYNXT's ability to properly model the relative velocity mismatch encountered both in full and mixed cores.

Reference 17.1: Virginia Electric & Power Company North Anna Power Station Units 1 and 2 Proposed Technical Specifications Changes & Exemption Request Use of Framatome ANP Advanced Mark-BW Fuel, March 28, 2002.

SOURCE NUMBER 2 QUESTIONS – SPENT FUEL POOL DESIGN BASIS

Q-1: Explain how the fuel change to Framatome from Westinghouse fuel will affect Spent Fuel Pool structural support systems, including but not limited to decay heat and thermal capacities.

RESPONSE:

In an April 10, 2002 teleconference between Dominion and NRC staff, it was clarified that the intent of this question was to address any thermal and heat load considerations resulting from the proposed fuel change which could affect the Spent Fuel Pool structural design basis. The following response has been prepared with this understanding.

The potential for the Framatome Advanced Mark-BW fuel to result in direct structural effects upon the Spent Fuel Pool and associated equipment has been considered and it was concluded to be insignificant. The assemblies are indistinguishable from the existing NAIF fuel design in terms of physical appearance and interface with fuel handling equipment and Spent Fuel Pool features such as storage racks. Table 1.0-1 of the Dominion evaluation report (transmitted by Reference 1.1) provides a comparison of key fuel dimensions between Advanced Mark-BW and NAIF fuel designs. The Advanced Mark-BW assembly weight is slightly greater (approximately 1%) than the existing NAIF assembly design. The effect of this increased weight is insignificant, even assuming Spent Fuel Racks were completely loaded with Advanced Mark-BW fuel assemblies. The assessment of thermal and heat load considerations for the Advanced Mark-BW fuel is addressed in the response to Question 2 below.

References:

1.1 Letter 02-167 from L. N. Hartz "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2 – Proposed Technical Specifications Changes and Exemption Request Use of Framatome ANP Advanced Mark-BW Fuel," March 28, 2002.

Q-2: Explain how the fuel change will affect the design basis for the spent fuel and Spent Fuel Pool including but not limited to the fuel handling accident, maximum bulk temperature for planned and unplanned offloads.

RESPONSE:

In an April 10, 2002 teleconference between Dominion and NRC staff, it was clarified that only the radiological aspects of the fuel handling accident were of interest. Section 10.0 of the Dominion evaluation report (transmitted by Reference 1) documents the evaluation of radiological consequences, including specific consideration of the Fuel Handling Accident.

The Design Basis Analysis for the maximum spent fuel pool (SFP) decay heat load for North Anna contains several sensitivity studies that evaluate the effect of changes in various parameters on the SFP decay heat load. The results of this analysis were used to develop a table of bounding parameters/assumptions that are checked each reload to confirm that the cycle specific SFP heat load remains bounded. The parameter values chosen for the Design Basis Accident bound the nominal values anticipated for the Framatome ANP fuel. Therefore, there is no impact on the SFP structure, decay heat, thermal capacity, maximum bulk temperature, or time to boil from the use of Framatome ANP fuel.

As previously discussed in Reference 2.1, the SFP heat load is evaluated on a cycle by cycle basis. If the reload SFP decay heat load is not bounded by the Design Basis Analysis results, then an evaluation pursuant to 10CFR50.59 will be performed and Section 9.1 of the UFSAR will be updated as necessary.

References:

2.1 Letter from S. Monarque (USNRC) to D. A. Christian (Virginia Electric and Power Company), "North Anna Power Station, Units 1 and 2 - Issuance of Amendments RE: Technical Specifications Changes to Increase Fuel Enrichment and Spent Fuel Pool Soluble Boron and Fuel Burnup Credit," TAC NOS. MB0197 and MB0198), June 15, 2001.

SOURCE NUMBER 3 QUESTIONS – FUEL MECHANICAL DESIGN

Q-1. Table 1.0-1 provides a dimensional comparison for components between Advanced Mark-BW assemblies and the NAIF fuel design. Table 2.3-1 provides a summary of the Advanced Mark-BW tests. Please provide test results regarding lateral stiffness, natural frequencies, critical damping that are used for analysis of the Advanced Mark-BW fuel assemblies in comparison to those used for the NAIF design.

RESPONSE:

The following table shows the test, test benchmark, and analysis natural frequencies for the BW and W fuel assemblies. Consistent with the appropriate topical report (see response to question 3 below), Rayleigh damping equal to [ ] for the first mode and [ ] for the third was used for both fuel assembly analysis models.

Table 1.1: Test and Analysis Natural Frequencies

[ ]

No lateral stiffnesses were supplied for the Westinghouse fuel assemblies. Lateral stiffnesses were not used directly in the analysis in any case. The next table lists some of the lateral grid stiffnesses and other properties for the two fuel assemblies. Ranges of values were supplied for the Westinghouse assemblies. Thus, average values were used in this analysis. Gaps 1 and 2 refer to the gaps for spacer grids and mid-span mixing grids, respectively, between fuel assemblies (FA-FA) and between fuel assemblies and the baffle walls (FA-BAF).

Table 1.2: Summary of Spacer Grid Quantities

[ ]

Q-2. Provide a discussion of the effects of flow induced vibration on Advanced Mark-BW fuel assemblies and reactor internals based on the calculated maximum cross-flow velocity due to the fuel change at North Anna.

RESPONSE:

Maximum flow induced vibration (FIV) response values were calculated for axial and cross flow. (The calculated maximum cross-flow due to the fuel change is less than the design cross-flow.) These results for the design flows show that the FIV responses of the North Anna FA are small (RMS rod and assembly displacements [ ] mils respectively) and are therefore acceptable. The maximum assembly response occurs at [ ], the maximum rod response is [ ].

Q-3. Provide a summary describing methodology, dynamic models and inputs, including damping values to evaluate the structural response of the Advanced Mark-BW fuel assemblies due to seismic and LOCA loads for North Anna Power Station. Also provide calculated maximum stresses and cumulative fatigue usage factors in comparison to the allowable limits for the critical components of the Advanced Mark-BW fuel assembly.

RESPONSE:

Faulted analysis methodology is described in Topical Report BAW-10133P-A, Rev. 1 (Reference 3.1). A summary of the key elements of horizontal and faulted analysis is provided below:

<u>Horizontal Faulted Methods, Models, and Loads</u>

• FA Models
A single line of vertical beam elements is used to represent the horizontal properties of each fuel assembly (FA). The total mass of the fuel assembly is uniformly distributed along the length. Rotational springs at each grid elevation supply the stiffening effects of the grid restraints on the fuel rods. The models are benchmarked to match the first six natural frequencies from test results.

• Grid Models
Sets of spring and gap elements are used to represent the horizontal properties of each grid. The 'external' or 'through-grid' stiffness (K_E) is the stiffness between opposite sides established with tests. The 'internal' or 'in-grid' stiffness (K_I) is the stiffness between the FA beam model and the theoretical center of the grid. These values are also established by benchmarking the finite element model to test results. Half of the external spring ($2K_E$) is assigned to each side of the grid center node for possible interaction with adjacent fuel assemblies or baffle walls. Appropriate gaps between fuel assemblies and maximum grid strengths are also assigned to these elements.

• Core Row Models
Rows of FA models are used to represent possible configurations and mixtures of Advanced Mark-BW and Westinghouse NAIF fuel assemblies in the core. The shortest row is 3, and the longest is 15. Hydro-dynamic coupling to the baffle walls is modeled. Rayleigh damping due to confined axial flow is set to [ ] for the first mode and [ ] for the third.

• Loads
Seismic (SSE) and LOCA loading is supplied to the core row models by imposing reactor vessel baffle motions, and lower and upper core plate motions to the bottoms and tops of the fuel assemblies. These motions are obtained from Westinghouse system analyses. Since the Advanced Mark-BW and Westinghouse NAIF fuel lateral characteristics are comparable (similar weight and first and third frequencies within 15%), no corrections to the horizontal time histories were required.

<u>Vertical Faulted Methods, Models, and Loads</u>

• Models
Several lines of vertical beam elements are used to represent the vertical properties of a fuel assembly. These lines represent groups of restrained guide tubes, unrestrained guide tubes, and fuel rods. A variety of gaps and sliders represent the grid properties connecting the guide tubes and fuel rods. Scalar springs are used to model the lower end fitting, the upper end fitting, and the hold down springs.

• Loads
LOCA pressure loadings are applied to the vertical model at the end fittings and at each grid elevation. These pressures are obtained from reactor vessel and piping hydraulic analyses. These motions are obtained from Westinghouse system analyses and were corrected based on FA axial pressure drop characteristics.

The following table gives the limiting faulted stress results for fuel rods, guide thimbles and nozzles. No fatigue usage factors were calculated for faulted conditions.

Table 3.1: Faulted Stresses and Margins

Grid results are summarized below. The LOCA grid impact forces [

]

Advanced Mark-BW Grids, LOCA + SSE

Intermediate Spacer Grid [ ]

Mid Span Mixing Grid [ ]

Westinghouse Grid, LOCA + SSE

Intermediate Spacer Grid [ ]

[

] The core coolable geometry will be maintained for all the faulted loads.

Fatigue evaluations are summarized in Table 3.2:

Table 3.2: Fatigue Analysis Summary

Reference 3.1: BAW-10133P-A, Rev.1, Addenda 1 and 2, Mark-C Fuel Assembly LOCA-Seismic Analyses, October 2000.

Q-4. In reference to Section 3.0, you stated that the fuel assembly structural evaluation is based on the Standard Review Plan and ASME Boiler and Pressure Vessel Code. Provide a code of record, including the section of code and code edition used in the evaluation of the Advanced Mark-BW fuel assembly.

RESPONSE:

Section III, Subsection NG, 1989 Edition of the ASME Boiler and Pressure Vessel Code was used.

NON-PROPRIETARY VERSION

Revised Pages for March 2002 Evaluation Report

Framatome Fuel Transition Program
Technical Specification Change

Improved CHF performance, relative to that of the Mark-BW mixing vane grid, is obtained by the addition of three Mid-Span-Mixing-Grids (MSMGs). This additional performance is incorporated into the BWU-Z CHF correlation by means of a direct CHF multiplication factor. An addendum (References 13 and 42) to the BWU-Z CHF topical report has been approved by the NRC for application of the enhanced CHF performance of the MSMGs using the multiplicative enhancement factor applied to the BWU-Z CHF correlation. When using the BWU-Z correlation in this manner, referenced specifically in the Addendum 2 to BAW-10199P, it is referred to as BWU-ZM. The approved range of application in the Reference 42 SER has been reviewed and confirmed to create no impact on the limiting NAPS analyses.

For the evaluation of DNB effects for NAIF in the mixed-core (Section 4.4.4), the BWU-N and BWU-I (N – non-mixing vane grid design, I – mixing vane grid design) CHF correlations (BAW-10199P-A, References 13 and 42) are used. Framatome justifies the application of these correlations to the NAIF on the fact that their databases include CHF data representative of the configuration for the Vantage 5H grids used on the Westinghouse NAIF fuel design (References 13 and 42). Therefore, these correlations apply without modification.

4.2.4 Form Loss Coefficients

In addition to modeling the assembly and core geometry, it is necessary to model the hydraulic characteristics of the assemblies and subchannels using form loss coefficients. The Advanced Mark-BW grid form loss coefficients were developed from a series of flow tests performed in the HERMES P loop in Cadarache, France. The HERMES P loop operates at PWR primary coolant conditions (i.e., 600°F, 2250 psia). [

] The combined results from these tests and analytical information form the basis for the current component form loss coefficient set. Subchannel form loss coefficients were determined analytically from the total spacer grid form loss coefficients. These grid and subchannel form loss coefficients are used in LYNXT to model the fuel assembly flow characteristics for both DNBR and pressure drop/hydraulic loads/crossflow velocity calculations for the resident fuel product.

4.2.5 Mixing Coefficients

Based on analysis of Laser Doppler Velocimeter testing, a turbulent mixing coefficient has been determined for the Mark-BW fuel design. The test, performed by Nuclear Fuel Industries (NFI) of Japan, provided an indication of the turbulent intensity at various distances downstream of the spacer grids. Research has shown that the turbulent mixing coefficient is proportional to the turbulent intensity. A value of 0.038 is conservatively applied in thermal-hydraulic analyses.

The thermal diffusion coefficient determined for the Mark-BW assembly is conservative for use with the Advanced Mark-BW. The coefficient is statistically derived from Laser Doppler Velocimeter (LDV) measurements of the three cross flow velocities are generated for a mixed-core configuration consisting of a single NAIF in the center location with the remainder of the core being Advanced Mark-BW. The maximum span-average cross flow velocities are below the [ ] and the cross flow velocity at every individual node is less than [ ].

4.4 DNB Performance Evaluation

To demonstrate that the DNB performance of the Advanced Mark-BW is acceptable, Framatome performed calculations for full-core and mixed-core configurations. The full-core DNB analyses demonstrated that the Advanced Mark-BW has margin to the applicable DNB limits (as described in section 4.4.2). The process of evaluating DNB in mixed-core configurations uses the Framatome mixed-core methodology outlined in Appendix A. The calculations are performed using the LYNXT computer code [ ].

The Framatome Statistical Core Design (SCD) methodology is used for applicable DNB analyses (Reference 19). The evaluation criterion for these analyses is that the minimum DNBR must be equal to or greater than the thermal design limits (Section 4.4.2). The evaluation criteria for non-SCD analysis is that the minimum DNBR must be equal to or greater than the CHF correlation design limits.

4.4.1 Statepoints for DNB Calculations

A set of more than 150 statepoint conditions was defined by Dominion for use in the full-core and mixed-core analyses to demonstrate acceptable DNB performance. The statepoints represent points on the safety limit lines, limiting axial flux shapes at several axial offsets and statepoints for several transient events including misaligned rod, loss of flow, rod withdrawal at power, locked rotor, rod urgent failure, rod withdrawal from subcritical and steam line break. The statepoints for the rod withdrawal from subcritical and steam line break are evaluated with deterministic (i.e., non-statistical) DNB methods. The remaining statepoints are evaluated using statistical DNB methods.

A smaller set of 35 statepoint conditions was developed for use in the statistical core design process (Section 4.4.2). Most of the statepoint conditions are identical to those defined for the DNB analysis. This set of statepoints covers the expected range of each of the boundary conditions on power, flow, inlet temperature, system pressure, and radial and axial peaking. Therefore, this set of statepoints was determined to be adequate for the development of the statistical design limits in the statistical core design process.

As noted in Section 4.2.11, the ultimate goal of the thermal-hydraulic analysis was to support a full-power radial power distribution factor ($F_{\Delta H}^{N}$) limit of 1.587 for reload cores that include the Advanced Mark-BW fuel. Thus, the statepoint conditions for the Advanced Mark-BW included $F_{\Delta H}^{N}$ values for each condition that were scaled by

Attachment 3

Framatome ANP Affidavit
Response to Request for Additional Information

Framatome Fuel Transition Program
Technical Specification Change

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Units 1 and 2

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is James F. Mallay. I am Director, Regulatory Affairs, for Framatome ANP ("FANP"), and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the information contained in Attachment 1 to a letter from Dominion Generation to the NRC (serial number 03-314). This attachment contains responses to draft RAIs and is referred to herein as "Document." Information contained in this Document has been classified by FANP as proprietary in accordance with the policies established by FANP 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 FANP 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 the Document be withheld from public disclosure.

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

(a) The information reveals details of FANP'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 FANP.

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

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

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

8. FANP 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 6th day of May, 2003.

Ella F. Carr-Payne

Ella F. Carr-Payne
NOTARY PUBLIC, STATE OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/05

ELLA F. CARR-PAYNE
Notary Public
Commonwealth of Virginia
My Commission Exps. Aug. 31, 2005