VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

May 6, 2003

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

Serial No. 03-313
NLOS/ETS
Docket Nos. 50-338/339
License Nos. NPF-4/7

Gentlemen:

**VIRGINIA ELECTRIC AND POWER COMPANY**
**NORTH ANNA POWER STATION UNITS 1 AND 2**
**REALISTIC LARGE BREAK LOSS OF COOLANT ACCIDENT (RLBLOCA) ANALYSIS RESULTS FOR THE PROPOSED TECHNICAL SPECIFICATIONS CHANGES AND EXEMPTION REQUEST FOR USE OF 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 and associated exemptions from 10 CFR 50.44 and 10 CFR 50.46 for North Anna Power Station Units 1 and 2. The amendments and 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. Subsequent to the March 28, 2002 letter, Dominion submitted supplements on REFLOD3B (July 25, 2002, Serial No. 02-167B), small break LOCA (August 2, 2002, Serial No. 02-167C), and large break LOCA (August 16, 2002, Serial No. 02-167D). Based on further discussions with the NRC, Dominion withdrew the LBLOCA and REFLOD3B submittals (November 15, 2002, Serial No. 02-167E) and agreed to submit a RLBLOCA analysis, and a revised small break LOCA (SBLOCA) analysis.

Attachment 1 to this letter provides the RLBLOCA results for Advanced Mark-BW fuel in North Anna Unit 2. The RLBLOCA information is presented in the form of a supplement to the evaluation report provided in our March 28, 2002 letter (specifically, report Section 7.0). Attachment 2 provides revised pages containing clarifications requested by the NRC staff for the remainder of the March 28, 2002 evaluation report. Please substitute these pages into the March 28, 2002 evaluation report to complete your review.

Attachments 1 and 2 contain Framatome ANP proprietary information. Attachment 4 is a signed affidavit from Framatome ANP, the owner of RLBLOCA information, which provides the basis for classifying the information in Attachment 1 as proprietary. The basis for classifying the information in Attachment 2 as proprietary was addressed

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(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 Attachments 1 and 2 is contained within brackets. A non-proprietary redacted version of Attachments 1 and 2 is also provided in Attachment 3. 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 Attachments 1 and 2 be withheld from public disclosure in accordance with 10 CFR 2.790 of the Commission's regulations.

Three additional submittals are planned that will provide the remaining technical evaluations necessary to support the amendment request for both North Anna Units 1 and 2. The first evaluation will provide the SBLOCA analysis results for North Anna Unit 2, and is planned for submittal by May 30, 2003. The second evaluation will revise the Core Operating Limits Report (Technical Specification 5.6.5) to include the RLBLOCA and SBLOCA topical reports, and is planned for submittal by June 15, 2003. The last evaluation will provide the RLBLOCA and SBLOCA analysis results for North Anna Unit 1, and is planned for submittal by July 30, 2003. These dates were discussed with the NRC in a telephone conversation on March 31, 2003.

To support the use of Framatome Advanced Mark-BW fuel in North Anna Unit 2, Cycle 17, we respectfully request the NRC to complete their review and approval of the license amendment and associated exemptions by September 30, 2003. We appreciate your consideration 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. Realistic Large Break LOCA Analysis Results – Unit 2 (Proprietary Version)
2. Revisions to March 2002 Edition of Licensing Analysis Report (Proprietary Version)
3. Revisions to March 2002 Edition of Licensing Analysis Report (Non-Proprietary Version)
4. Framatome ANP Affidavit, Realistic Large Break LOCA Analysis Results – Unit 2

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North Anna Power Station

SN: 03-313
Docket Nos.: 50-338/339
Subject: Proposed TS Change – Framatome Fuel Transition
RLBLOCA

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 6th day of May, 2003.

My Commission Expires: March 31, 2004.

Maggie McClure
Notary Public

(SEAL)

Attachment 3

NON-PROPRIETARY VERSION

# Realistic Large Break LOCA Analysis Results – Unit 2

and

# Revisions to March 2002 Edition of Licensing Analysis Report

Framatome Fuel Transition Program
Technical Specification Change

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

**NON-PROPRIETARY VERSION**

# Realistic Large Break LOCA Analysis Results – Unit 2

## Framatome Fuel Transition Program
## Technical Specification Change

# 7.0 LOCA Analysis

Framatome ANP (FANP) will deliver Advanced Mark-BW reload fuel to the North Anna Power Station (NAPS) Units 1 and 2 starting in the first quarter of 2004. These units are Westinghouse-designed, three-loop plants operating at a rated thermal power of 2,893 MWt. The plants have conventional emergency core cooling systems (ECCS) and dry, sub-atmospheric containment buildings. In accordance with 10CFR50.46, an evaluation of ECCS performance was performed for the FANP reload fuel. The evaluation and results are discussed and summarized in the text that follows. All analysis results and evaluations confirm that the acceptance criteria of 10CFR50.46 are met. The assessment serves as a basis for operation with FANP Advanced Mark-BW fuel at NAPS.

Section 7.1 discusses the computer codes and calculative methods used in the NAPS analyses. The realistic large break loss-of-coolant-accident (RLBLOCA) analysis is discussed in Section 7.2. Compliance with the coolable geometry and long-term cooling criteria is presented in Sections 7.2.5 and 7.2.6, respectively. Section 7.2.7 contains concluding large break LOCA remarks. The small break LOCA (SBLOCA) analysis is presented in Section 7.3. Section 7.4 summarizes the LOCA results, and references are provided in Section 7.5.

## 7.1 LOCA Computer Codes and Methods

The re-circulating steam generator (RSG) LOCA evaluation models (References 7-1 through 7-3) consist of computer codes and input models for performing large and small break LOCA calculations. Both evaluation models are NRC-approved for application to Westinghouse 3-loop plants, such as the NAPS units. The EMs are briefly described below.

For large break LOCA predictions, the FANP realistic LOCA evaluation model (EM) is used. The realistic large break LOCA EM is described in Reference 7-1 and approved in Reference 7-2. The EM consists of two computer codes, S-RELAP5 and RODEX3A. The S-RELAP5 code calculates blowdown and refill/reflood system thermal-hydraulics, core power generation, and core thermal responses. The minimum containment backpressure is calculated by S-RELAP5. RODEX3A provides steady-state fuel inputs to S-RELAP5. The realistic approach requires that a set of "sampled" cases be run. For each case, "key LOCA parameters" are randomly sampled over their uncertainty or operating limit ranges, and the cases are then run to steady state before transient initiation. Limiting PCT case set results are qualified at a 95/95 level.

Small break LOCA calculations are performed using the FANP deterministic EM (Reference 7-3, Volume II). The entire transient is predicted by RELAP5 (Reference 7-4). TACO3 (Reference 7-5) provides steady-state fuel inputs to the RELAP5 analysis.

## 7.2 Realistic Large Break LOCA Analysis

This section describes the RLBLOCA analysis supporting operation of the NAPS units with FANP Advanced Mark-BW fuel. The modeling applied represents the initial core composition accounting for thermal and hydraulic influences of both NAIF and Advanced Mark-BW fuel assemblies in proportion to their count and distribution. Although the reactor core as constituted for this analysis is specific to the initial fuel reload cycle, the results bound the differing follow-on cycles as the transition to all FANP fuel is accomplished.

The non-parametric statistical methods inherent to the FANP RLBLOCA methodology provide for consideration of a full spectrum of break sizes, break configurations (guillotine or split breaks), axial shapes, and plant operational parameters. The analysis assumes a conservative single-failure.

The analysis verifies Technical Specification peaking factor limits and the adequacy of the ECCS by demonstrating, with a high level of probability, that the following 10CFR 50.46(b) criteria are met:

- The calculated maximum fuel element cladding temperature shall not exceed 2,200 °F.
- The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel excluding the cladding surrounding the plenum volume, were to react.

Compliance with the remaining 10CFR50.46 criteria is discussed in Sections 7.2.5 and 7.2.6. Section 7.2.1 of this report describes the postulated LBLOCA event. Section 7.2.2 describes the models used in the analyses. Section 7.2.3 describes the 3-loop PWR NAPS plants and summarizes the system parameters used in the analyses. Section 7.2.4 summarizes the results of the RLBLOCA analyses for NAPS Unit 2.

### 7.2.1 LBLOCA Event Description

A LBLOCA is initiated by a postulated large break in the reactor coolant system (RCS) primary piping. The RLBLOCA EM identifies that the worst break location is in the cold leg piping between the reactor coolant pump and the reactor vessel for the RCS loop containing the pressurizer. The break initiates a rapid depressurization of the RCS. A reactor trip signal is initiated when the low pressurizer pressure trip setpoint is reached; however, reactor trip is conservatively neglected in the analysis. The reactor is shut down by coolant voiding in the core.

The plant is assumed to be operating normally at full power prior to the accident. The large cold leg break is assumed to open instantaneously. For this break, a rapid depressurization occurs, along with a core flow stagnation and reversal. This causes the fuel rods to experience departure from nucleate boiling (DNB). Subsequently, the limiting fuel rods are cooled by film and transition boiling heat transfer. The coolant voiding creates a strong negative reactivity effect, and core fission ends. As heat transfer from the rods is reduced, the cladding temperature rises.

Coolant in all regions of the RCS begins to flash. At the break plane, the loss of subcooling in the coolant results in substantially reduced break flow. This reduces the depressurization rate and may lead to a period of positive core flow or reduced downflow as the reactor coolant pumps in the intact loops continue to provide flow to the vessel. Cladding temperatures may be reduced, and some portions of the core may rewet during this period.

This positive core flow or reduced downflow period ends as two-phase conditions occur in the reactor coolant pumps, reducing their effectiveness. Again, the core flow reverses as most of the vessel mass flows out through the broken cold leg.

Mitigation of the LBLOCA begins when the safety injection (SI) actuation signal occurs. This signal is initiated by either high containment pressure or low-low pressurizer pressure. Regulations require that a worst case single failure be considered for ECCS safety analysis. This single failure has been determined to be the loss of one ECCS train, including one high-head safety injection (HHSI) pump and one low-head safety injection pump (LHSI) pump. The FANP RLBLOCA methodology assumes a conservatively early start and normal lineups of the containment spray to reduce containment pressure and increase break flow. Hence, the analysis assumes that one HHSI pump, one LHSI pump, and all containment spray pumps are operating (Reference 7-1).

When the RCS pressure falls below the accumulator pressure, fluid from the accumulators is injected into the cold legs. In the early delivery of accumulator water, high pressure and high break flow will cause some of this fluid to bypass the core. During this bypass period, core heat transfer is degraded, and fuel rod cladding temperatures increase. As RCS and containment pressures equilibrate, ECCS water begins to fill the lower plenum and eventually the lower portions of the core. As this occurs, core heat transfer improves, and cladding temperatures decrease.

Eventually, the relatively large volume of accumulator water is exhausted, and core recovery must rely on SI coolant delivery alone. As the accumulators empty, the nitrogen gas used to pressurize the accumulators exits through the break. This gas release may result in a short period of improved core heat transfer as the nitrogen gas displaces water in the downcomer. After the nitrogen gas has been expelled, the ECCS temporarily may not be able to sustain full core cooling because of the core decay heat and the higher steam temperatures created by quenching in the lower portions of the core. Peak fuel rod cladding temperatures may increase for a short period until more energy is removed from the core by the safety injection than the decay heat produces. Steam generated from fuel rod rewet will entrain liquid as it passes through the core, vessel upper plenum, the hot legs, the steam generator, and the reactor coolant pump before it is vented out the break. The hydraulic resistance of this flow path to the steam flow is balanced by the driving force of water filling the downcomer. This resistance may act to retard the progression of the core reflood and postpone core-wide cooling. Eventually (within a few minutes of the accident), the core reflood will progress sufficiently to ensure core-wide cooling.

Full core quench occurs within a few minutes after core-wide cooling. Long-term cooling is then sustained with the low-head safety injection.

### 7.2.2 Description of Analytical Models

The RLBLOCA methodology is documented in EMF-2103, "Realistic Large Break LOCA Methodology," (Reference 7-1) and approved in Reference 7-2. The methodology follows the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (Reference 7-6). This method outlines an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifies the uncertainties in a LOCA analysis.

The RLBLOCA methodology consists of the following computer codes:

- RODEX3A for computation of the initial fuel stored energy, fission gas release, and fuel-cladding gap conductance.
- S-RELAP5 for the system calculation.

The governing two-fluid (plus non-condensibles) model with conservation equations for mass, energy, and momentum transfer is used. The reactor core is modeled in S-RELAP5 with heat generation rates determined from reactor kinetics equations (point kinetics) with reactivity feedback, and with actinide and decay heating.

The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on another are accounted for by interfacial friction, and heat and mass transfer interaction terms in the conservation equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ.

The modeling of plant components is performed by following guidelines developed to ensure accurate accounting for physical dimensions and the dominant phenomena expected during LBLOCA. The basic building blocks for modeling are the hydraulic volumes for fluid paths and the heat structures for heat transfer surfaces. In addition, special purpose components exist to represent specific components such as the pumps or the steam generator separators. Plant geometry is modeled at the resolution necessary to best resolve the flow field and the phenomena being modeled within practical computational limitations.

A typical calculation for each of the "sampled" cases using S-RELAP5 begins with the establishment of a steady-state initial condition with all loops intact. Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops. The evolution of the transient through blowdown, refill, and reflood is computed continuously using S-RELAP5.

The methods used in the application of S-RELAP5 to the large break LOCA are described in Reference 7-1. A detailed assessment of this computer code was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the ability of the code to predict important physical phenomena in a PWR large break LOCA. The final step of the realistic LOCA methodology is to combine all the uncertainties related to the code and plant parameters and estimate the peak clad temperature (PCT) at 95 percent probability. The steps taken to derive the PCT uncertainty estimate are summarized below:

*1. Base Plant Input File Development*

First, base RODEX3A and S-RELAP5 input files for the plant (including the containment input file) are developed based on plant-specific, customer-supplied information. Code input development guidelines are applied to ensure that the model nodalization is consistent with the model nodalization used in the code validation.

*2. Sampled Case Development*

The non-parametric statistical approach requires that many "sampled" cases be created and processed. For every set of input created, each "key LOCA parameter" is randomly sampled over a range established through code uncertainty assessment or expected operating limits (provided by the customer through plant Technical Specifications, data, etc.). Those parameters considered "key LOCA parameters" are listed in Table 7.2-1. This list includes both parameters related to LOCA phenomena (based on the PIRT provided in Reference 7-1) and to plant operating parameters.

*3. Determination of Adequacy of ECCS*

The RLBLOCA methodology uses a non-parametric statistical approach to determine values of PCT, peak local oxidation, and total oxidation. The PCT is determined at a 95 percent probability level with 95 percent confidence. The peak local oxidation and total oxidation are reported for the limiting PCT case. The adequacy of the ECCS is demonstrated when these results satisfy the criteria set forth in Section 7.2.

### 7.2.3 Plant Description and Summary of Analysis Parameters

The RLBLOCA plant analyses presented in this report are for NAPS Units 1 and 2. The units are similar except that Unit 1 has an upflow barrel-baffle configuration while Unit 2 has a downflow configuration. The plants are Westinghouse-designed pressurized water reactors (PWRs), that have three loops, each with a hot leg, a U-tube steam generator, and a cold leg with a reactor coolant pump (RCP). The plants are bottom reflooded. The RCS includes a pressurizer and pressurizer surgeline. The ECCS includes an accumulator path and a LHSI/HHSI path per RCS loop. The HHSI and LHSI feed into a common header that connects to each cold leg pipe downstream of the RCP discharge.

The S-RELAP5 model explicitly describes the RCS, reactor vessel, pressurizer, and passive ECCS back to the accumulators. The ECCS pumped injection is modeled as a table of flow versus backpressure. This model also describes the secondary-side steam generator that is instantaneously isolated (closed main steamline isolation valve (MSIV) and feedwater trip) at the time of the break. A symmetric steam generator tube plugging level of 12 percent (customer-provided) per steam generator was modeled.

As described in the FANP RLBLOCA methodology, many parameters associated with LBLOCA phenomenological uncertainties and plant operation ranges (including appropriate Technical Specification allowed ranges) are sampled. A summary of those parameters sampled is given in Table 7.2-1. The LBLOCA phenomenological uncertainties are provided in Reference 7-1. Values for plant parameters are given in Table 7.2-2. The table provides values describing the physical plant, plant initial operating conditions, including operating ranges, and accident boundary conditions. Diesel start time is set consistent with the loss-of-offsite-power assumption for ECC pumped injection. Table 7.2-3 presents process parameters and statistical distributions used in the analyses. The distributions in the table provide additional margin to accommodate parameter uncertainties.

Summarizing, the NAPS plant models were constructed, using customer-supplied, plant-specific inputs, in accordance with RLBLOCA EM requirements. The customer-supplied inputs describe the plants to the extent required by the FANP RLBLOCA methodology, covering all necessary modeling and analysis aspects, such as: geometry, setpoints, thermal-hydraulic conditions, operating ranges, etc. The degree of detail in the NAPS models corresponds to that in the Westinghouse 3-loop sample problem supplied to the NRC for review along with the RLBLOCA EM (Reference 7-1).

7.2.3.1 SER Compliance

A number of requirements on the methodology are stipulated in the conclusions section of the SER for the RLBLOCA methodology (Reference 7-2). These requirements have all been complied with during the application of the methodology. Three SER-requested discussion points are (1) nodalization, (2) range of applicability, and (3) analysis results. These three points are addressed below.

- The nodalization of the NAPS units is essentially identical to the Westinghouse 3-loop sample calculation that was submitted to the NRC for review along with the RLBLOCA EM (Reference 7-1). Figure 7.2-1 shows the loop noding for both NAPS units. (Note only loop 1 is shown in the figure; loops 2 and 3 are identical to loop 1, except that only loop 1 contains the pressurizer and the break.) NAPS loop noding is identical to that of the sample problem excepting ECC pumped injection modeling. The sample problem models the ECC pumped injection lines to the point of a common header. For the NAPS application, ECC pumped injection is based on a line network analyses, separate and distinct from the RLBLOCA analyses. The model applies the result of the network analyses, in the form of flow versus system backpressure, as a problem boundary condition. The injection location is unchanged from the sample problem. Both modeling methods consider the ECC pumped injection line resistance network and are equally valid for use within the RLBLOCA EM.

  Figures 7.2-2 and 7.2-3 show the RV noding diagram for NAPS Units 1 and 2, respectively. The RV noding for Unit 2 (Figure 7.2-3) is identical to the sample problem. The RV noding for Unit 1 is identical to the sample problem, excepting the barrel-baffle configuration. Unit

1 is an upflow plant; the sample problem is a downflow plant (as is NAPS Unit 2). Within its application envelope, the RLBLOCA EM accommodates modeling of specific plant-to-plant differences, for example, the NAPS barrel-baffle differences between units. Hence, Unit 1 barrel-baffle modeling, as shown in Figure 7.2-2, is valid for use within the RLBLOCA EM.

Axially, the core for both NAPS units contains two more nodes than does the sample calculation. Core noding derives from a formulation based on the number of grids per fuel assembly. FANP Advanced Mark-BW fuel for both NAPS units contains three mid-span mixing vane grids (MSMGs), located above the core mid-plane, in addition to structural grids. The fuel modeled in the sample calculation contains only structural grids. Hence, to comply with EM modeling requirements, two axial nodes were added to the core, thereby properly modeling the Advanced Mark-BW fuel with MSMGs.

- Simulation of clad temperature response is a function of phenomenological correlations that have been derived either analytically or experimentally. The important correlations have been validated for the RLBLOCA methodology and a statement of the range of applicability has been documented. The correlations of interest are the set of heat transfer correlations as described in Reference 7-7. Table 7.2-4 presents the summary of the full range of applicability for the important heat transfer correlations as presented in Reference 7-7. Table 7.2-5 provides a cross-reference to Table 7.2-4 that identifies the parameters present in the limiting calculation from the North Anna plant-specific RLBLOCA analysis. As is evident, the plant-specific parameters fall within the methodology's range of applicability.

- Analysis results are discussed in Section 7.2.4.

7.2.3.2 Mixed-Core Description

The North Anna core model is representative of the number and placement of fuel assemblies that are anticipated for the first cycle of operation with Advanced Mark-BW fuel. The mixed-core configuration affects the hydraulic features of the core model in both the radial and axial directions. The RLBLOCA methodology simulates the core in four radial regions: (1) the hot assembly, (2) the assemblies surrounding the hot assembly, (3) the average core, and (4) the core periphery. The hot assembly (and hot pin) is simulated by an Advanced Mark-BW fuel assembly. The core periphery is simulated by NAIF assemblies. The ring of fuel assemblies surrounding the hot fuel assembly and the average core are constructed to hydraulically simulate the axial and crossflow resistance of an approximate 50:50 mixture of NAIF and Advanced Mark-BW fuel assemblies. The axial modeling of the core accounts for the increased hydraulic resistance, due to the Advanced Mark-BW MSMGs, in radial regions where FANP fuel assemblies are located.

The core model bounds the results for subsequent cycles of operation, comprising progressively larger percentages of FANP fuel. Appendix B of Reference 7-1 established that the analysis of fresh fuel bounds the analysis results for fuel in its second or third cycle of irradiation. The hydraulic characteristics of the NAIF and the Advanced Mark-BW assemblies create a preference for flow diversion toward the NAIF assemblies and away from the Advanced Mark-BW assemblies in the upper half of the core. The potential for this diversion is maximized during the first cycle of Advanced Mark-BW operation because of the small percentage of FANP fuel that is present in the core. As the percentage of FANP fuel increases in subsequent reload cycles, the potential for flow diversion is lowered. Because provision for this flow diversion is explicitly modeled in the North Anna mixed-core RLBLOCA calculations, the expected results for subsequent reload cycles would demonstrate lower PCTs and oxidation results. Together, the results of the Reference 7-1, Appendix B study and the increase in the number of Advanced Mark-BW fuel assemblies in the core lead to the conclusion that first cycle calculations bound subsequent cycles of operation with FANP fuel.

### 7.2.4 Realistic Large Break LOCA Results

The analyses assume full-power operation at 2,893 MWt (plus uncertainties), a steam generator tube plugging level of 12 percent in all generators, a total peaking factor ($F_Q$) of 2.32, and a nuclear enthalpy rise factor ($F_{\Delta H}$) of 1.65. These analyses accommodate operation within specified ranges for sampled parameters: pressurizer pressure and level, accumulator pressure, temperature (containment temperature) and level, RCS average temperature, core flow, and containment pressure and temperature.

A set of fifty-nine calculations was performed for NAPS Unit 2 sampling the parameters listed in Table 7.2-1. The remainder of this section provides results from that analysis.

#### 7.2.4.1 NAPS Unit 1 Large Break LOCA Results

Text for this section will be provided in a subsequent revision.

### 7.2.4.2 NAPS Unit 2 Large Break LOCA Results

The limiting PCT case (2,032 °F) was Case 8. It is characterized in Table 7.2-9 and Table 7.2-10. The maximum oxidation (3.2 %) and total oxidation (0.071 %) results are also reported in Table 7.2-10. The fraction of total hydrogen generated was not directly calculated; however, it is conservatively bounded by the calculated total percent oxidation that is well below the 1 percent limit. A nominal 50/50 PCT case was identified as Case 37. The nominal PCT is 1,448 °F. This result can be used to quantify the relative conservatism in the limiting PCT case result. In this analysis, it is 584 °F.

The hot fuel rod results, event times and analysis plots for the limiting PCT case are shown in Table 7.2-10, Table 7.2-11, and in Figures 7.2-19 through 7.2-33, respectively. Figure 7.2-19 shows linear scatter plots of the important parameters sampled for the 59 calculations. Parameter labels appear to the left of each individual plot. These figures show the parameter ranges used in the analysis. Figures 7.2-20 and 7.2-21 show PCT scatter plots versus the time of PCT and versus break size from the 59 calculations. Figure 7.2-22 shows the maximum oxidation versus PCT for the 59 calculations. Figures 7.2-23 through 7.2-33 show important parameters from the S-RELAP5 limiting calculation. Figure 7.2-23 is the plot of PCT independent of elevation.

### 7.2.5 Large Break LOCA Core Geometry

The effect of fuel rod bowing on whole-core blockage is considered in the fuel assembly and fuel rod designs that minimize the potential for rod bowing. The minor adjustments of fuel pin pitch due to rod bowing do not alter the fuel assembly flow area substantially and the average sub-channel flow areas are preserved. Therefore, there is no detrimental effect of rod bow on LOCA calculations.

NAPS-specific calculations indicate that deformation of the fuel pin lattice in some core periphery fuel assemblies occurs from the combined mechanical LOCA and seismic loads (Reference 7-8, Section 3.3.3). The predicted deformations have a maximum impact of reducing the sub-channel flow area of one row of pins by 32 percent. (If the deformation is evenly spread over three rows of fuel pins the blockage per sub-channel reduces to 11 percent.) The assembly flow area is reduced by less than two percent and retains a coolable geometry.

Evaluations of the impact of this amount of flow area reduction on the LOCA performance of fuel pins in the peripheral assemblies were conducted with the following results: (1) The coolant flow within these assemblies is not substantially altered; and (2) The maximum cladding temperature during LOCA for the affected pins remain below 1,800 °F. This is less than the temperature at which significant metal-water reaction occurs. Hence, these grid deformations do not lead to conditions that interfere with core coolability; nor do they affect the reported PCT or metal-water oxidation results.

The consequences of thermal and mechanical deformation of the fuel assemblies in the core were assessed. The resultant deformed geometry maintains a coolable configuration. The conclusions rely on basic phenomena encountered during LOCA and are equally applicable to the Advanced

Mark-BW fuel and the current resident NAIF. Therefore, the coolable geometry requirements of 10CFR50.46 are met, and the core remains amenable to cooling.

### 7.2.6 Large Break LOCA Long-Term Cooling

Successful initial operation of the ECCS is shown by demonstrating that the core is quenched, and the cladding temperature is returned to near saturation. Thereafter, long-term cooling is achieved by the pumped injection systems. These systems are redundant and provide a continuous flow of cooling water to the core fuel assemblies so long as the coolant channels in the core remain open. For a cold leg break, the concentration of boric acid within the core can induce a crystalline precipitation that may prevent coolant flow from reaching portions of the core. This section evaluates the initial operation of the ECCS, considers the long-term supply of water to the core, and discusses the procedures to prevent the build-up of boric acid in the core.

#### 7.2.6.1 Initial Cladding Cooldown

The LOCA calculations in Section 7.2.4 provide a simulation of the hottest fuel pins through core quench. After quenching, core heat transfer is through pool nucleate boiling or forced convection to liquid, depending on the break location (cold leg breaks are in pool nucleate boiling and hot leg breaks are in forced convection to liquid). Either heat transfer mechanism is fully capable of maintaining the core within a few degrees of the coolant saturation temperature. Thus, within ten to fifteen minutes following a large break LOCA, the core is returned to an acceptably low temperature.

#### 7.2.6.2 Extended Coolant Supply

Once the core is cooled to low temperature, maintaining that condition relies upon the systems that are designed to provide a continuous supply of coolant to the core. Detailed descriptions of the plant systems and functions are provided in the NAPS UFSAR. Long-term core cooling with the ECCS is independent of the fuel design. Thus, the current licensing basis remains valid for Advanced Mark-BW fuel assemblies.

#### 7.2.6.3 Boric Acid Concentration

The long-term cooling mechanism for a hot leg break is forced convection to liquid. Once cooling is established, and a positive core flow is assured, boron precipitation is not an issue, and no further consideration is necessary. For cold leg breaks, there is no forced flow through the core. The liquid head balance between the core and the downcomer prevents ECCS water from entering the core at a rate faster than core boil-off. Extra injection simply flows out the break and spills to the containment floor. With no core flow, core boiling acts to concentrate boric acid adding to the potential for precipitation and core blockage. To eliminate boron precipitation and any accompanying core blockage, operator action is required to establish hot leg re-circulation (positive core flow).

In this mode, the ECCS is aligned to inject into the hot legs. In doing so, the hot leg injection provides a positive core flow capable of controlling the concentration of boric acid. The timing and effectiveness of the hot leg injection is established by demonstrating that the in-vessel concentration of boric acid is below solubility limits. There is no dependency on the fuel element design since concentrations depend on ECCS injection rate, RCS geometry, and core power level. Since the Framatome ANP fuel does not alter these factors, the current evaluation remains valid and is equally applicable to Advanced Mark-BW fuel. Emergency operating procedures provide guidance to address the boric acid precipitation issue and ensure that long-term cooling is maintained.

7.2.6.4 Adherence to Long-Term Cooling Criterion

Compliance with this criterion is demonstrated in the NAPS UFSAR. It is independent of fuel design. The initial phase of core cooling results in low clad and fuel temperatures. A pumped injection system, capable of re-circulation, is available and operated by the plants to provide extended coolant injection. The concentration of dissolved solids is limited to acceptable levels through the timely implementation of hot leg injection. Hence, long-term cooling is established and compliance to 10CFR50.46 demonstrated.

7.2.7 Large Break LOCA Conclusions

The analyses reported herein support operations at a power level of 2,893 MWt, a steam generator tube plugging level of 12 percent in each generator, a total peaking factor ($F_Q$) of 2.32 and a nuclear enthalpy rise factor ($F_{\Delta H}$) of 1.65. The analyses support peak rod average exposures of up to 62,000 MWd/mtU. The analyses applied no $K_Z$ restraint on axial peaking; that is, $K_Z$ is set equal to one for all core elevations. The impact of NAIF co-resident fuel on FANP Advanced Mark-BW fuel is included within the analyses—the analyses consider the initial core composition of both NAIF and Advanced Mark-BW fuel. The analysis of the Westinghouse fuel remains valid. The co-resident FANP fuel, being 2.5 psi (based on rated flow) more resistive than NAIF, will promote favorable flow diversion to NAIF, thereby improving its LBLOCA performance. Hence, the NAIF will be positively (lower clad temperature and metal-water oxidation) affected by the co-resident FANP fuel.

The limiting PCT for Unit 2 is 2,032 °F. Maximum oxidation thickness and hydrogen generation for both units are well within regulatory requirements. Discussions in Sections 7.2.5 and 7.2.6 demonstrate compliance with the coolable geometry and long-term cooling criteria.

Table 7.2-1: NAPS Units 1 and 2 Sampled RLBLOCA Parameters

| Phenomenological | |
|---|---|
| | Time in cycle (axial shape, rod properties, and burnup) |
| | Peaking factors |
| | Break type (guillotine versus split) |
| | Break size |
| | Critical flow discharge coefficients (break) |
| | Offsite power availability |
| | Decay heat |
| | Critical flow discharge coefficients (surgeline) |
| | Initial upper head temperature |
| | Film boiling heat transfer |
| | Dispersed film boiling heat transfer |
| | Critical heat flux |
| | Tmin (intersection of film and transition boiling) |
| | Initial stored energy |
| | Downcomer hot wall effects |
| | Steam generator interfacial drag |
| | Condensation interphase heat transfer |
| | Metal-water reaction |
| **Plant**[a] | |
| | Core power |
| | Initial flow rate |
| | Initial operating temperature |
| | Pressurizer pressure |
| | Pressurizer level |
| | Containment volume |
| | Containment temperature |
| | Accumulator pressure |
| | Accumulator system volume |

[a] Uncertainties for plant parameters are based on customer-provided plant-specific data.

Table 7.2-2: NAPS Units 1 and 2 Plant Parameter Values Supported by the RLBLOCA Analyses

| | Parameter Description | Parameter Value |
|---|---|---|
| **1.0** | **Plant Physical Description** | |
| | *1.1 Fuel* | |
| | a) Cladding outside diameter | 0.374 in |
| | b) Cladding inside diameter | 0.329 in |
| | c) Cladding thickness | 0.0225 in |
| | d) Pellet outside diameter | 0.3225 in |
| | e) Pellet density | 96% of theoretical |
| | f) Active fuel length | 144 in |
| | g) Maximum rod average exposure | ≤ 62,000 MWd/mtU |
| | *1.2 RCS* | |
| | a) Flow resistance | Analysis |
| | b) Pressurizer location | Analysis assumes location giving most limiting PCT |
| | c) Hot assembly location | Anywhere in core |
| | d) Hot assembly type | 17x17 |
| | e) SG tube plugging | ≤ 12% |
| **2.0** | **Plant Initial Operating Conditions** | |
| | *2.1 Reactor Power* | |
| | a) Core power | ≤ 2,893 MWt |
| | b) Maximum core peaking ($F_Q$) | ≤ 2.32[a] (normalized) |
| | c) Maximum pin radial peaking ($F_{\Delta H}$) | ≤ 1.65[b] (normalized) |
| | d) Maximum assembly radial peaking | < 1.587[c] (normalized) |
| | e) MTC | ≤ 0 at HFP |
| | f) HFP boron | Normal letdown |
| | *2.2 Fluid Conditions* | |
| | a) Loop flow | 110.5 Mlbm/hr ≤ M ≤ 118.2 Mlbm/hr |
| | b) RCS average temperature | 580.8 °F ≤ T ≤ 586.8 °F |
| | c) Upper head temperature | < Core Outlet Temperature |
| | d) Pressurizer pressure | 2,220 psia ≤ P ≤ 2,250 psia |
| | e) Pressurizer level | 64.5%[d] |
| | f) Accumulator pressure | 613.7 psia ≤ P ≤ 681.7 psia |
| | g) Accumulator system liquid volume | 954.8 $ft^3$ ≤ V ≤ 978.5 $ft^3$ |
| | h) Accumulator temperature | 86 °F ≤ T ≤ 120 °F (coupled to containment temperature) |

[a] Includes 5% measurement uncertainty and 3% engineering uncertainty.

[b] Includes 4% measurement uncertainty.

[c] Value equivalent to hot rod peaking factor without 4% uncertainty.

[d] Value reflects nominal operating point maintained by plant control system.

Table 7.2-2: NAPS Units 1 and 2 Plant Parameter Values Supported by the RLBLOCA Analyses (continued)

| | Parameter Description | Parameter Value |
|---|---|---|
| | i) Accumulator line resistance | As-built piping configuration |
| | j) Minimum ECCS boron | ≥ 2,200 ppm |
| **3.0** | **Accident Boundary Conditions** | |
| | a) Break location | Any RCS piping location |
| | b) Break type | Double-ended guillotine or split |
| | c) Break size per side (relative to cold leg pipe) | 0.05 ≤ A ≤ 0.5 full pipe area (split)<br>0.5 ≤ A ≤ 1.0 full pipe area (guillotine) |
| | d) Worst single failure | Loss of one LHSI and one HHSI |
| | e) Offsite power | On or Off |
| | f) Low-head safety injection flow | Flow network result injected at cold leg injection as flow versus pressure |
| | g) High-head safety injection flow | Flow network result injected at cold leg injection as flow versus pressure |
| | h) RWST temperature | ≤ 60 °F |
| | i) Safety injection delay | ≤ 13 seconds (with offsite power)<br>≤ 27 seconds (without offsite power) |
| | j) Containment pressure | Bounding current configuration |
| | k) Containment temperature | 86 °F ≤ T ≤ 120 °F |
| | l) Containment sprays | Minimum actuation time |

Table 7.2-3: NAPS Units 1 and 2 Statistical Distributions Used for Process Parameters

| Parameter | Operational Uncertainty Distribution | Parameter Range | Standard Deviation, σ | Uncertainty Included in Parameter Range Endpoints |
|---|---|---|---|---|
| | | | | |
| Core Power (%) | Gaussian | 100 | 1.1% | N/A |
| Initial Flow Rate (Mlbm/hr) | Uniform | 108.26 – 120.54 | 1.0% | 2σ |
| Initial Average Operating Temperature (°F) | Uniform | 576.8 – 590.8 | 2.0 °F | 2σ |
| Pressurizer Pressure (psia) | Uniform | 2,184 –2,286 | 18.0 psi | 2σ |
| Pressurizer Level (%) | Uniform | 54.5 – 74.5 | 5% | 2σ |
| Containment Volume (x $10^6$ $ft^3$) | Uniform | 1.825 – 2.087[a] | N/A | N/A |
| Containment Temperature (°F) | Uniform | 84.5 – 121.5 | 0.75 °F | 2σ |
| Accumulator Pressure (psia) | Uniform | 583.7 – 711.7 | 15.0 psi | 2σ |
| Accumulator System Volume ($ft^3$) | Uniform | 951.0 – 982.3 | 1.9 $ft^3$ | 2σ |

[a] Minimum value represents a lower bound, assuming minimum containment diameter, maximum installed equipment and provision for structural concrete; maximum value is gross volume of empty containment with nominal dimensions.

Table 7.2-4: Summary of Full Range of Applicability

[ ]

Table 7.2-5: Plant-Specific Compliance Measure for RLBLOCA Range of Applicability

[ ]

Table 7.2-6: Summary of Major Parameters for Limiting NAPS Unit 1 Transient

This table will be provided in a subsequent revision.

Table 7.2-7: Summary of Results for the NAPS Unit 1 Limiting PCT Case

This table will be provided in a subsequent revision.

Table 7.2-8: Calculated Event Times for the NAPS Unit 1 Limiting PCT Case

This table will be provided in a subsequent revision.

Table 7.2-9: Summary of Major Parameters for Limiting NAPS Unit 2 Transient

| | |
|---|---|
| Time (hrs) | 8,114 |
| Burnup (MWd/mtU) | 18,000 |
| Core Power (MWt) | 2,906 |
| Core Peaking ($F_Q$) | 2.295 |
| Radial Peak ($F_{\Delta H}$) | 1.65 |
| Local Peaking (Fl) | 1.05 |
| Break Type | DEGB |
| Break Size per Side ($ft^2$) | 2.59 (~63%) |
| Offsite Power Available | No |
| Decay Heat Multiplier | 1.0215 |

Table 7.2-10: Summary of Results for the NAPS Unit 2 Limiting PCT Case

| Case Number | 8 |
|---|---|
| PCT | |
| Temperature | 2,032 °F |
| Time | 92.3 seconds |
| Elevation | ~9.9 ft |
| Metal-Water Reaction | |
| % Oxidation Maximum | 3.2 % |
| % Total Oxidation | 0.071 % |
| Total Hydrogen | 1.15 lbm |

Table 7.2-11: Calculated Event Times for the NAPS Unit 2 Limiting PCT Case

| Event | Time (sec) |
|---|---|
| Begin Analysis | 0.0 |
| Break Opens | 0.0 |
| RCP Trip | 0.0 |
| SI ACTUATION SIGNAL Issued | 0.9 |
| Start of Broken Loop Accumulator Injection | 10 |
| Start of Intact Loop Accumulator Injection | 13 |
| End of Bypass | 26 |
| Start of HHSI | 28 |
| Start of LHSI | 28 |
| Beginning of Core Recovery (Beginning of Reflood) | 31 |
| Broken Loop Accumulator Empties | 37 |
| Intact Loop Accumulators Empties | 39, 39 |
| PCT Occurs (2,032 °F) | 92.3 |

Figure 7.2-1: NAPS Units 1 and 2 RLBLOCA Loop Noding Diagram

Figure 7.2-2: NAPS Unit 1 RLBLOCA RV Noding Diagram

Figure 7.2-3: NAPS Unit 2 RLBLOCA RV Noding Diagram

Figure 7.2-4: NAPS Unit 1 Scatter Plots of Operational Parameters

This figure will be provided in a subsequent revision.

Figure 7.2-5: NAPS Unit 1 PCT versus PCT Time Scatter Plot

This figure will be provided in a subsequent revision.

Figure 7.2-6: NAPS Unit 1 PCT versus Break Size per Side Scatter Plot

This figure will be provided in a subsequent revision.

Figure 7.2-7: NAPS Unit 1 Maximum Oxidation versus PCT Scatter Plot

This figure will be provided in a subsequent revision.

Figure 7.2-8: NAPS Unit 1 Peak Cladding Temperature for the Limiting Break (elevation independent)

This figure will be provided in a subsequent revision.

Figure 7.2-9: NAPS Unit 1 Break Flow for the Limiting Break

This figure will be provided in a subsequent revision.

Figure 7.2-10: NAPS Unit 1 Early Core Inlet Mass Flux for the Limiting Break

This figure will be provided in a subsequent revision.

Figure 7.2-11: NAPS Unit 1 Core Outlet Mass Flux for the Limiting Break

This figure will be provided in a subsequent revision.

Figure 7.2-12: NAPS Unit 1 Void Fraction at RCS Pumps for the Limiting Break

This figure will be provided in a subsequent revision.

Figure 7.2-13: NAPS Unit 1 ECCS Flows (includes Accumulator, HHSI and LHSI) for the Limiting Break

This figure will be provided in a subsequent revision.

Figure 7.2-14: NAPS Unit 1 System (Upper Plenum) Pressure for the Limiting Break

This figure will be provided in a subsequent revision.

Figure 7.2-15: NAPS Unit 1 Collapsed Liquid Level in the Downcomer for the Limiting Break

This figure will be provided in a subsequent revision.

Figure 7.2-16: NAPS Unit 1 Collapsed Liquid Level in the Lower Vessel for the Limiting Break

This figure will be provided in a subsequent revision.

Figure 7.2-17: NAPS Unit 1 Collapsed Liquid Level in the Core for the Limiting Break

This figure will be provided in a subsequent revision.

Figure 7.2-18: NAPS Unit 1 Containment and Loop Pressures for the Limiting Break

This figure will be provided in a subsequent revision.

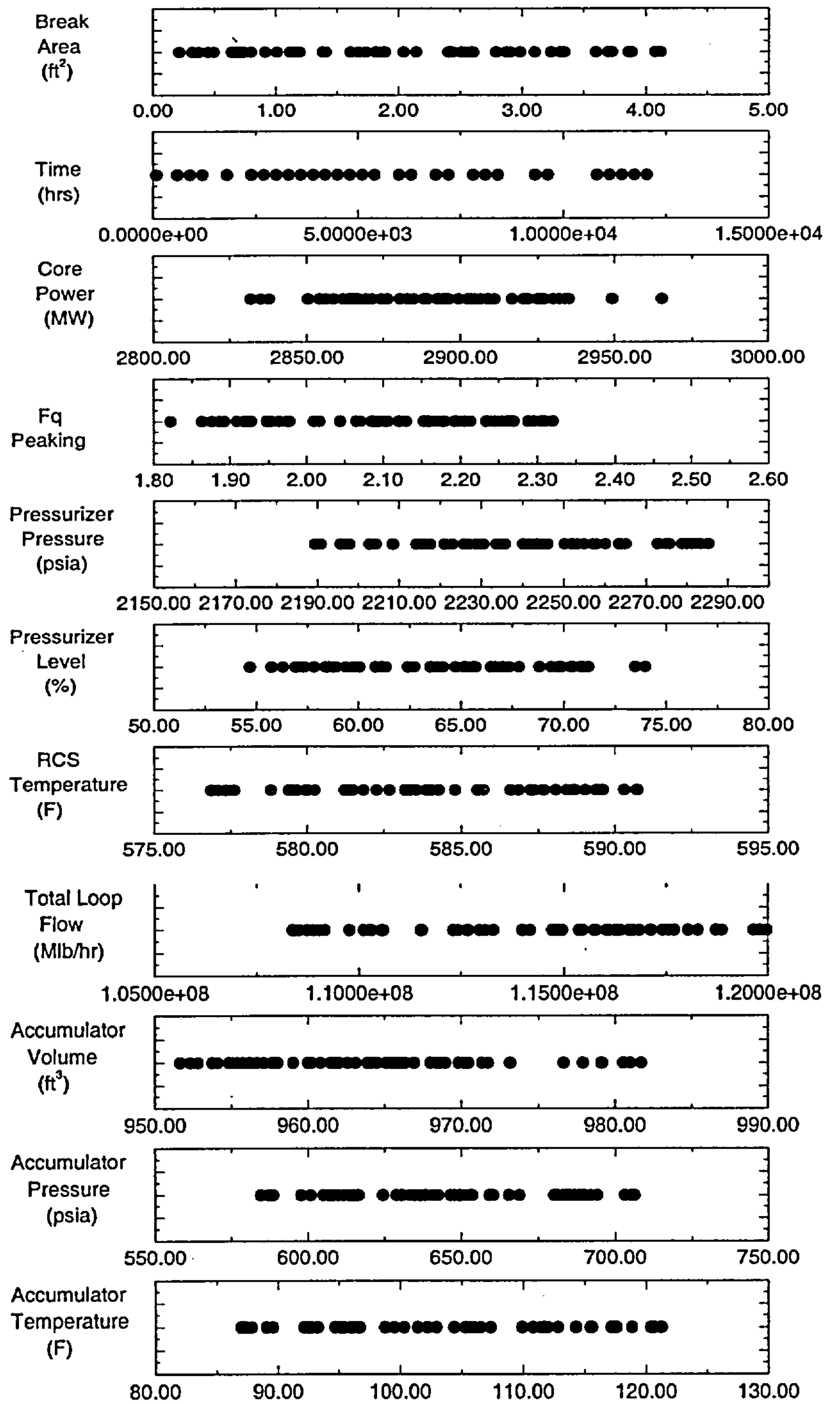


Figure 7.2-19: NAPS Unit 2 Scatter Plots of Operational Parameters

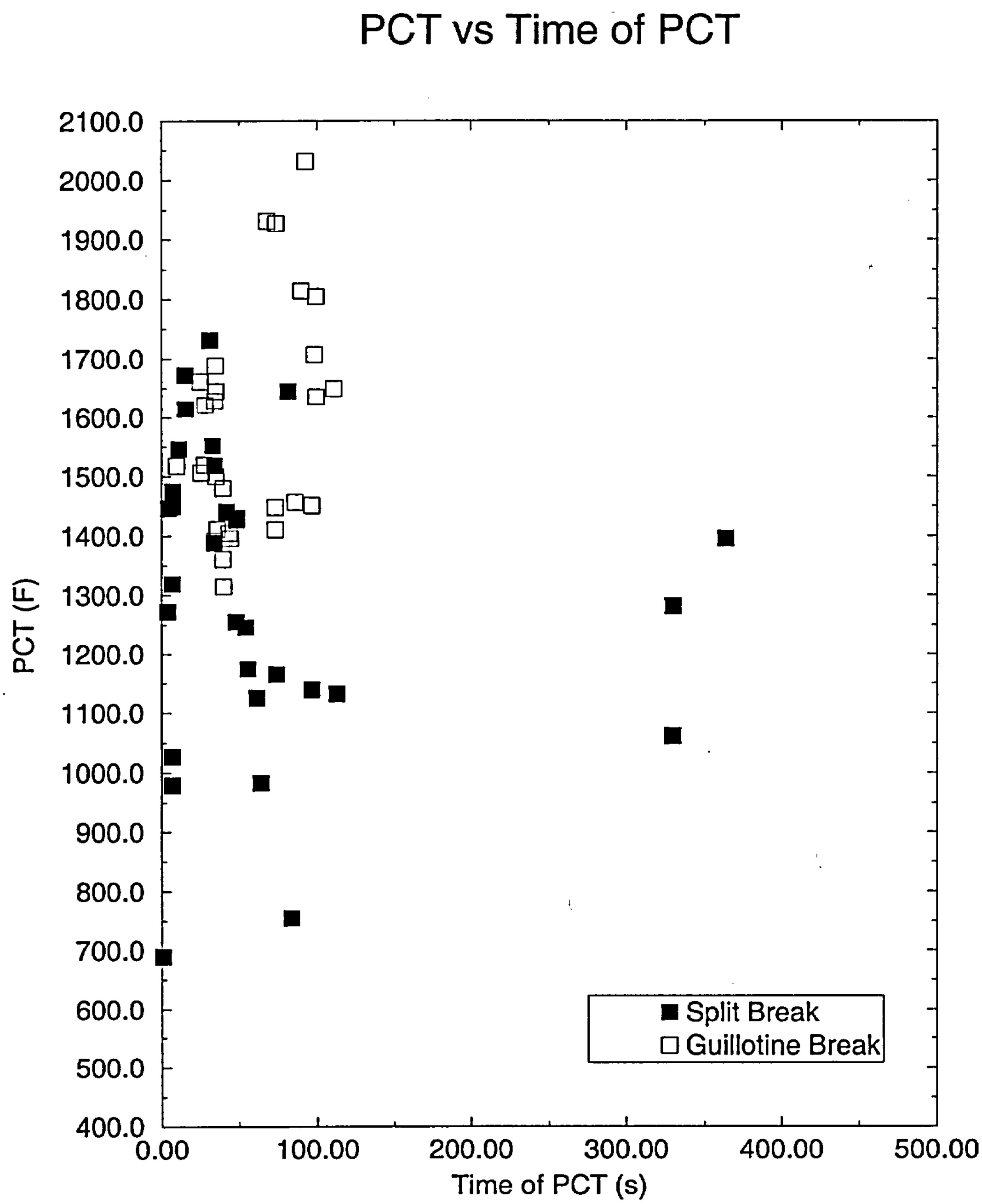


Figure 7.2-20: NAPS Unit 2 PCT versus PCT Time Scatter Plot

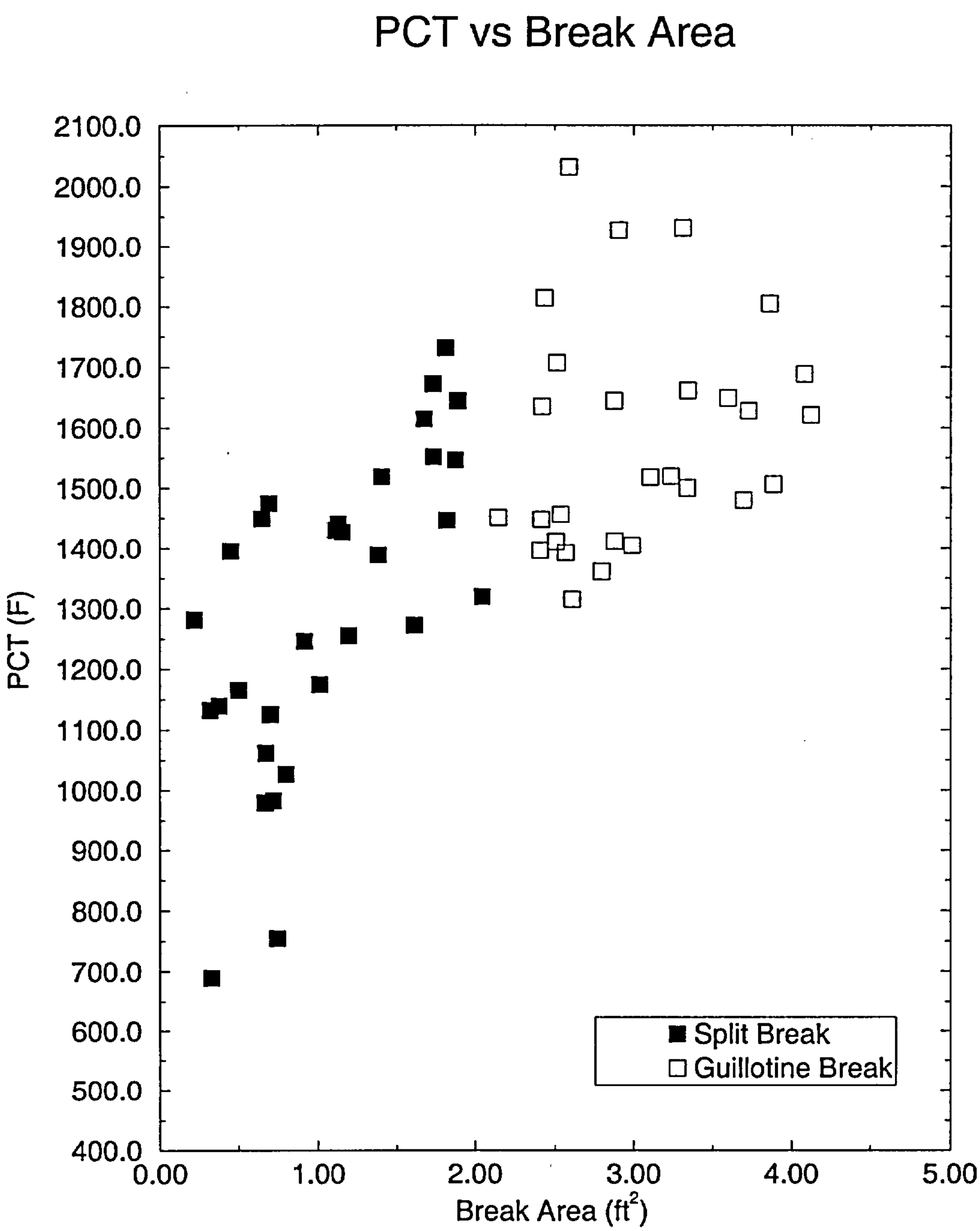


Figure 7.2-21: NAPS Unit 2 PCT versus Break Size per Side Scatter Plot

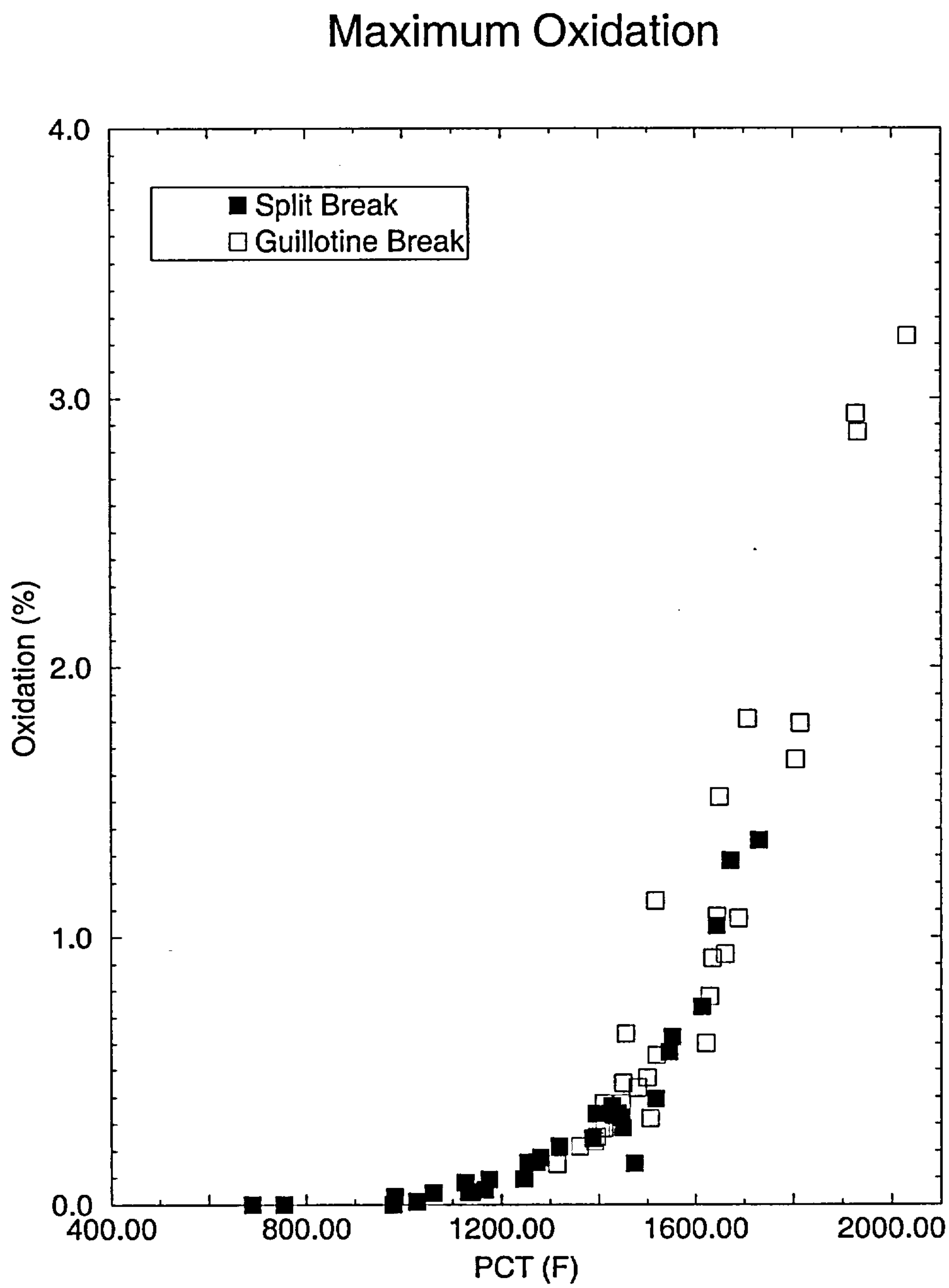


Figure 7.2-22: NAPS Unit 2 Maximum Oxidation versus PCT Scatter Plot

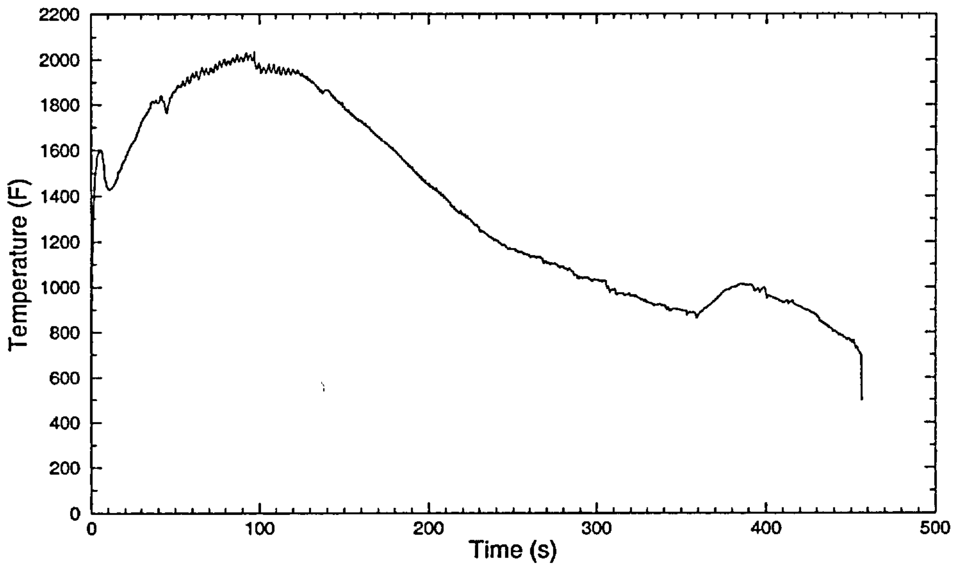


Figure 7.2-23: NAPS Unit 2 Peak Cladding Temperature for the Limiting Break (elevation independent)

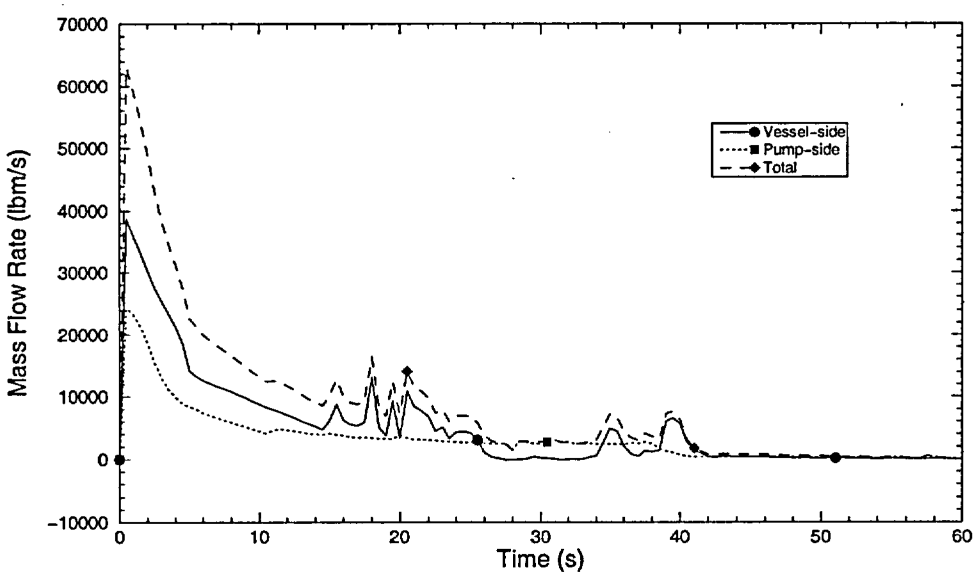


Figure 7.2-24: NAPS Unit 2 Break Flow for the Limiting Break

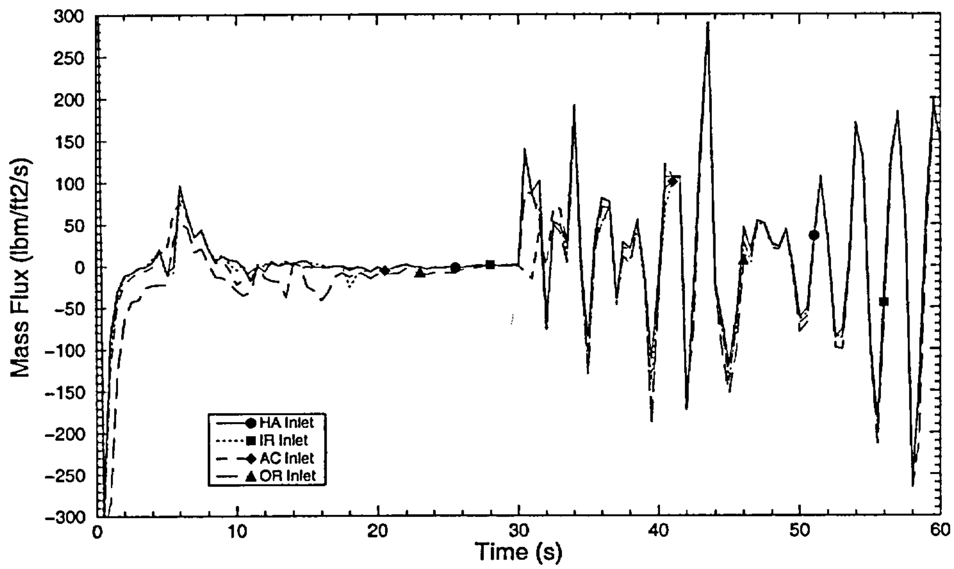


Figure 7.2-25: NAPS Unit 2 Early Core Inlet Mass Flux for the Limiting Break

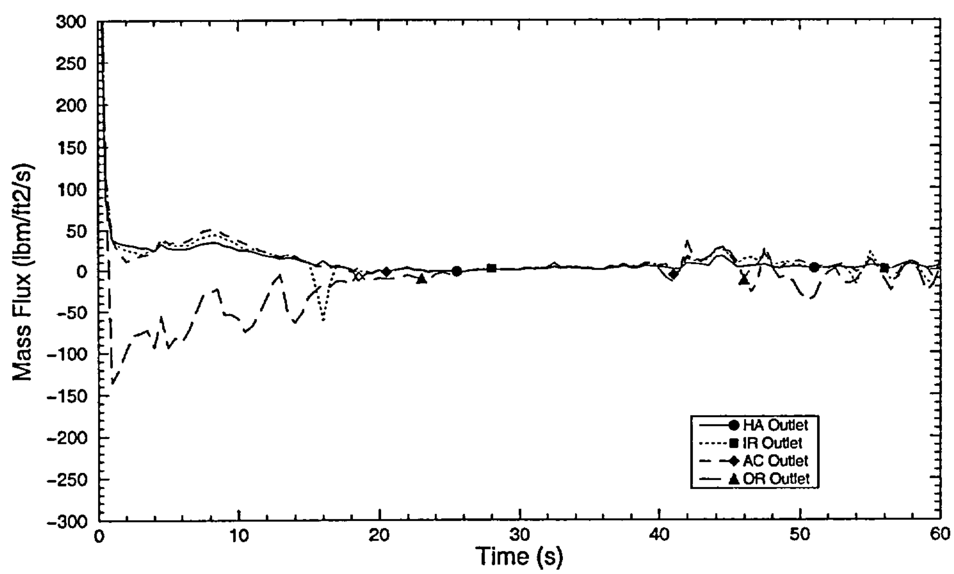


Figure 7.2-26: NAPS Unit 2 Core Outlet Mass Flux for the Limiting Break

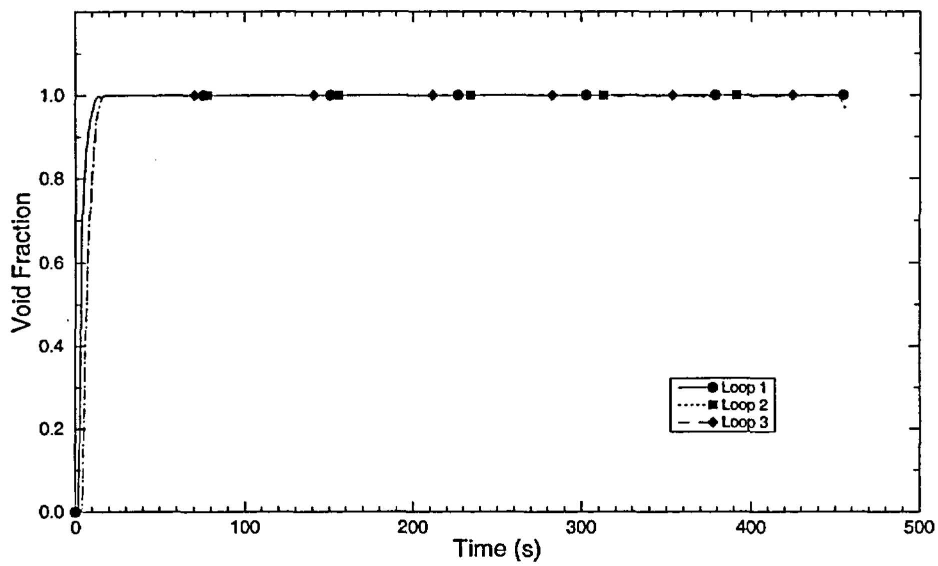


Figure 7.2-27: NAPS Unit 2 Void Fraction at RCS Pumps for the Limiting Break

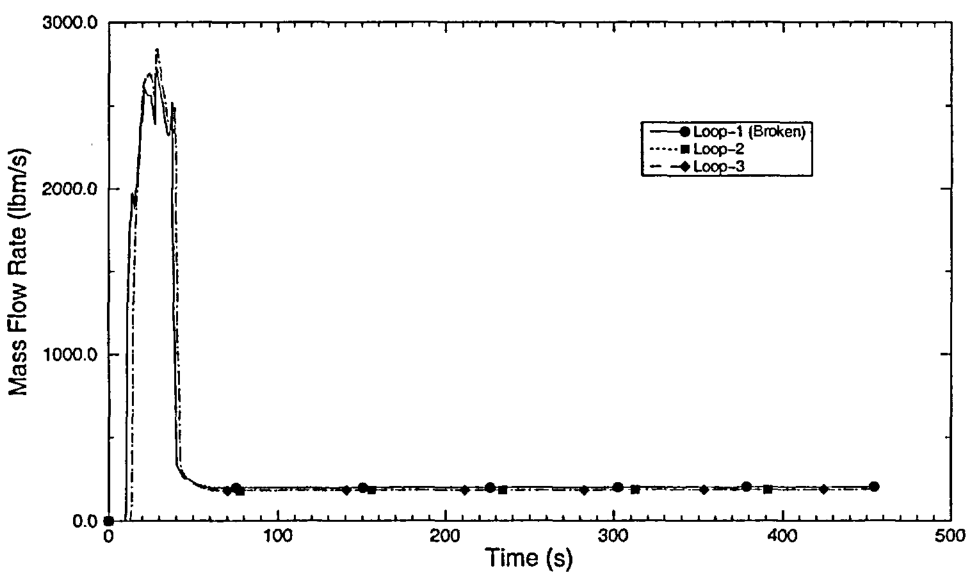


Figure 7.2-28: NAPS Unit 2 ECCS Flows (includes Accumulator, HHSI and LHSI) for the Limiting Break

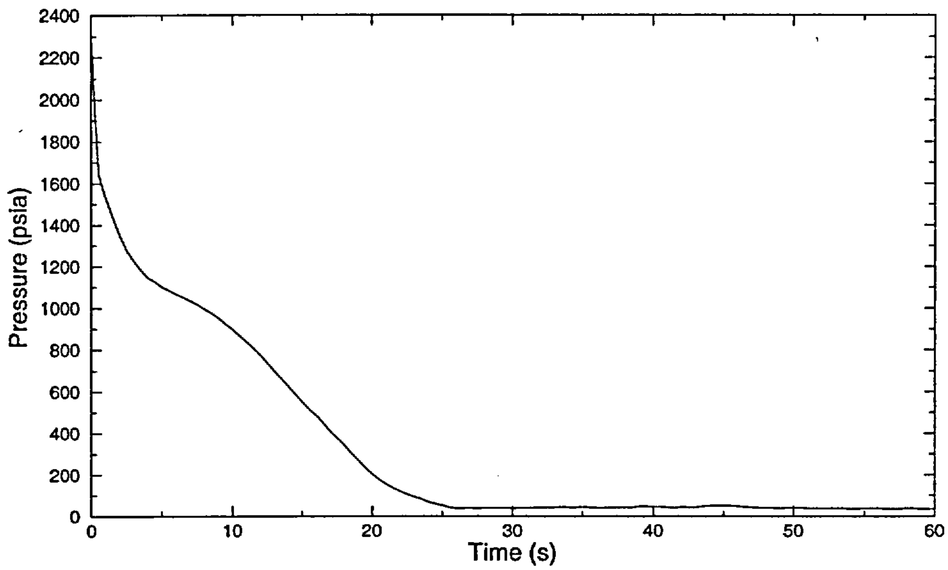


Figure 7.2-29: NAPS Unit 2 System (Upper Plenum) Pressure for the Limiting Break

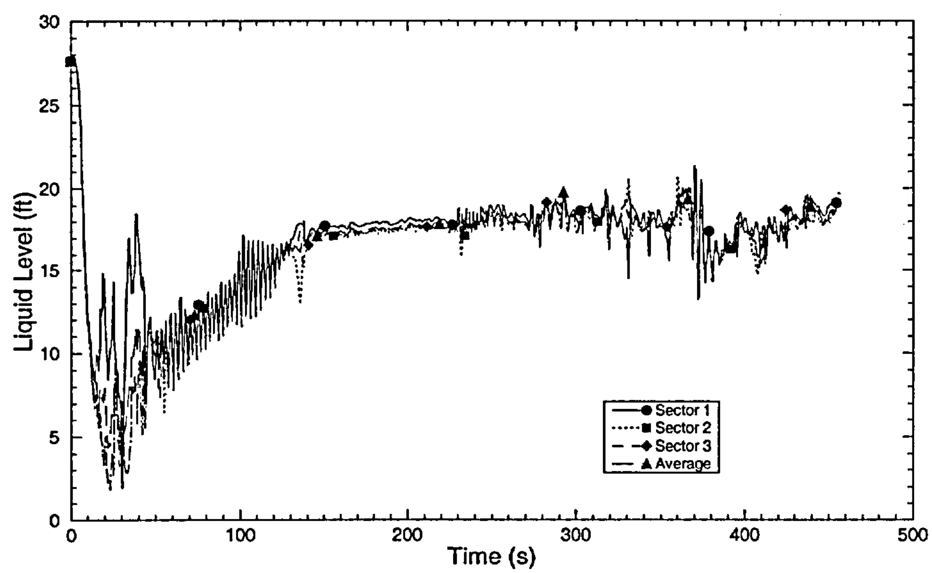


Figure 7.2-30: NAPS Unit 2 Collapsed Liquid Level in the Downcomer for the Limiting Break

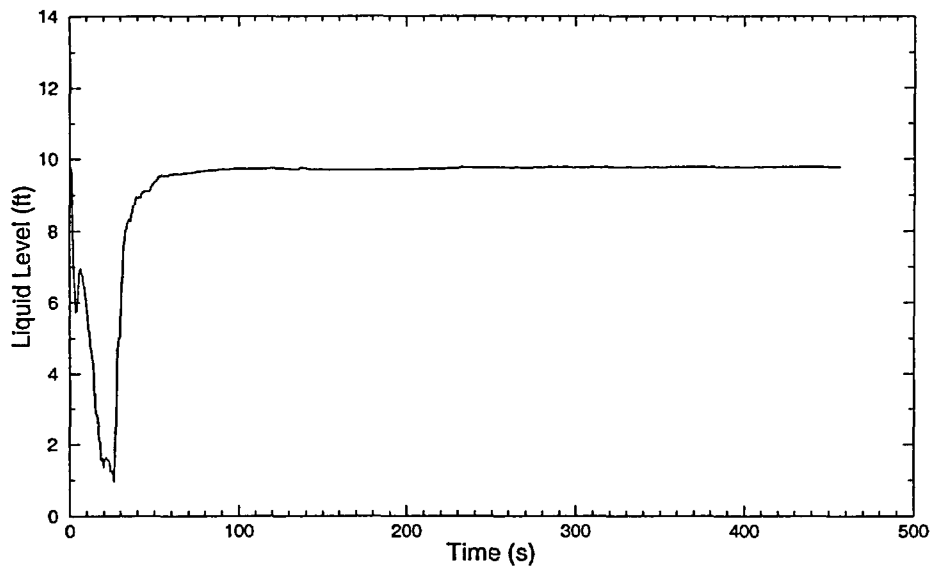


Figure 7.2-31: NAPS Unit 2 Collapsed Liquid Level in the Lower Vessel for the Limiting Break

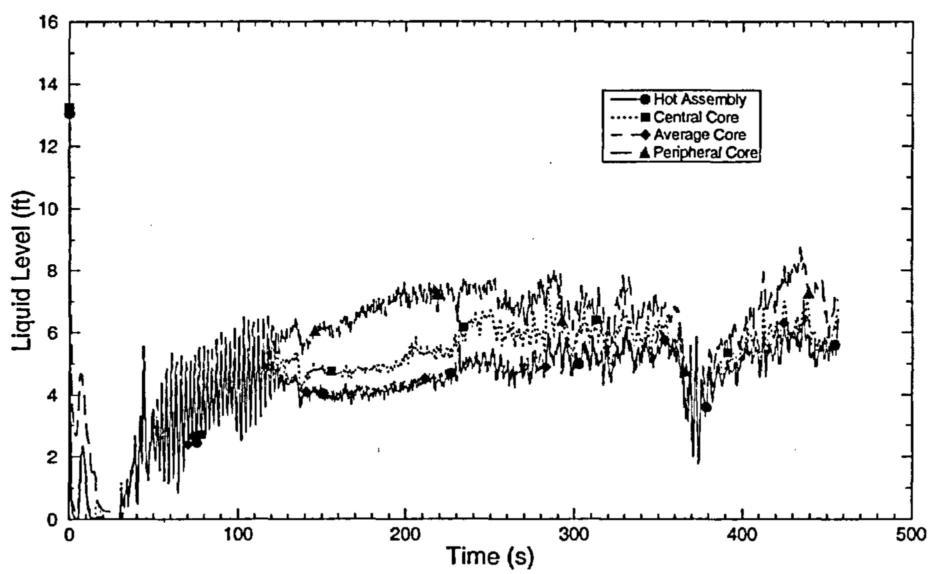


Figure 7.2-32: NAPS Unit 2 Collapsed Liquid Level in the Core for the Limiting Break

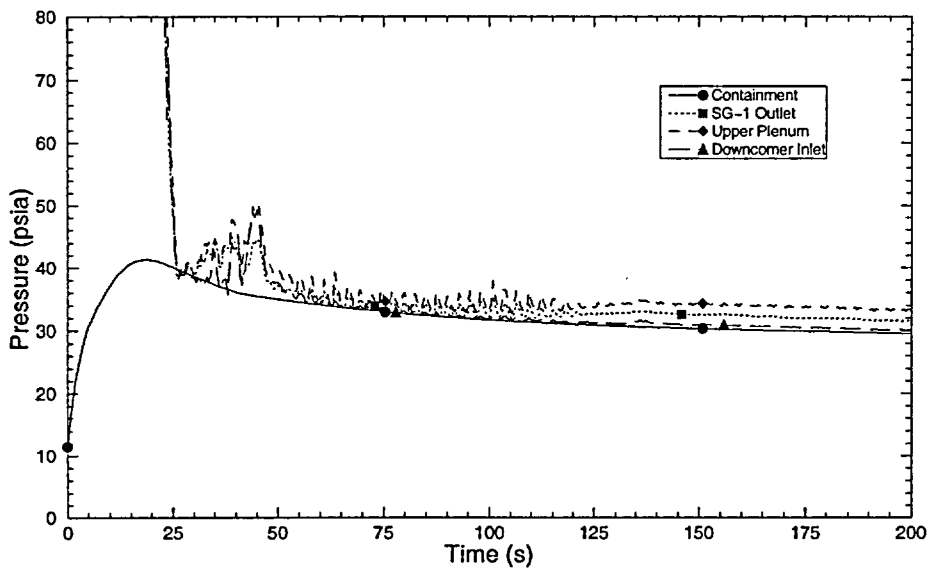


Figure 7.2-33: NAPS Unit 2 Containment and Loop Pressures for the Limiting Break

## 7.3 Small Break LOCA Analysis

Text for this section will be provided in a subsequent revision.

## 7.4 LOCA Summary

10CFR50.46 specifies that the ECCS for a commercial nuclear power plant must meet five criteria. The calculations and evaluations documented in this chapter demonstrate that the two NAPS units meet the required licensing criteria when operated with Advanced Mark-BW fuel. LOCA calculations performed in accordance with approved evaluation models (Reference 7-1 through 7-3) demonstrate compliance for breaks up to and including the double-ended severance of the largest primary coolant pipe. The co-residence of Advanced Mark-BW fuel and NAIF assemblies in the same fuel cycle is concluded to be of minor consequence and does not cause the calculated clad temperature of either assembly to approach the limits of 10CFR50.46.

Specifically, this report, in conjunction with Dominion's LOCA evaluations for NAIF, concludes that when the North Anna units are operated with Advanced Mark-BW fuel:

1. The calculated PCT for the limiting PCT case is less than 2,200 °F.

2. The maximum calculated local clad oxidation is less than 17 percent.

3. The maximum amount of core-wide oxidation does not exceed 1 percent of the fuel cladding.

4. The cladding remains amenable to cooling.

5. Long-term cooling is established and maintained after the LOCA.

Large break studies were performed for both units using the FANP RLBLOCA evaluation model (References 7-1 and 7-2). Tables 7.2-6 through 7.2-11 show the RLBLOCA results. The RLBLOCA analyses applied no $K_Z$ restraint on axial peaking, that is, $K_Z$ is set equal to one for all core elevations. The results demonstrate LBLOCA compliance with the five criteria of 10CFR50.46. The transition core was evaluated and no significant impact on either NAIF or Advanced Mark-BW fuel was identified.

7.5 References


7-1 EMF-2103(P) Revision 0, Realistic Large Break LOCA Methodology, FANP Richland, Inc., August 2001.

7-2 NRC Letter: Herbert N. Berkow, NRC, to James F. Mallay, FANP, "SAFETY EVALUATION ON FRAMATOME ANP TOPICAL REPORT EMF-2103(P), REVISION 0, "REALISTIC LARGE BREAK LOSS-OF-COOLANT ACCIDENT METHODOLOGY FOR PRESSURIZED WATER REACTORS" (TAC NO. MB7554)," April 9, 2003.

7-3 RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants, BAW-10168P-A, Revision 3, December 1996.

7-4 RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis, BAW-10164P-A, Revision 4, November 2002.

7-5 TACO3 - Fuel Pin Thermal Analysis Code, BAW-10162P-A, January 1990.

7-6 Technical Program Group, Quantifying Reactor Safety Margins, NUREG/CR-5249, EGG-2552, December 1989.

7-7 FANP Letter: NRC-02-062A(P), Responses to a Request for Additional Information on EMF-2103(P), Revision 0, "RLB LOCA Methodology for PWR's" (TAC NO. MB2865), 12/20/2002.

7-8 Dominion Letter: L. N. Hartz (Virginia Electric and Power Company) to Document Control Desk (NRC), "North Anna Power Station Units 1 and 2, Proposed Technical Specifications Changes and Exemption Request Use Framatome ANP Advanced Mark-BW Fuel," Serial No. 02-167, March 28, 2002.

7-9 FTI Letter: J. J. Kelly, FTI, to Document Control Desk, NRC, "A Request to Rescind FTI's Commitment to Analyze Top and Side SBLOCAs for $T_{hot}$ Recirculating Steam Generator Plants—BAW-10168P-A, Revision 3, Volume II, December 1996," FTI-98-3797, December 10, 1998.

7-10 NRC Letter: Stephen Dembek, NRC, to James F. Mallay, Framatome ANP, "BREAK ORIENTATION ANALYSIS FOR SMALL BREAK LOSS OF COOLANT ACCIDENT (TAC NO. MA9313)," August 2, 2001.

**NON-PROPRIETARY VERSION**

# Revisions to March 2002 Edition of Licensing Analysis Report

**Framatome Fuel Transition Program**
**Technical Specification Change**

## 4.0 Thermal-Hydraulic Evaluation

This section provides a description of the thermal-hydraulic analyses performed to demonstrate the acceptable performance of the Framatome Advanced Mark-BW fuel in full-core and mixed-core configurations with the resident NAIF fuel. The primary purpose of the thermal-hydraulic analysis is to demonstrate acceptable thermal performance that ensure fuel and clad integrity are maintained during normal operation and transients of moderate frequency. The design criteria that have been established to meet this goal are as follows:

(1) During Condition I and II events, there must be at least a 95 % probability with a 95 % confidence level that the hot pin will not experience a departure from nucleate boiling (DNB); and a 99.9% probability that DNB will not occur core-wide.

(2) During Condition I and II events, there must be at least a 95 % probability with a 95 % confidence level that no fuel rod will experience centerline melting.

The second purpose of the hydraulic evaluation is to ensure hydraulic compatibility with the resident NAIF design. Core pressure drop, hydraulic lift forces and crossflow velocities were assessed justifying acceptable mixed-core and full-core performance of the Advanced Mark-BW fuel assembly in the North Anna reactors.

The thermal-hydraulic analyses were performed using NRC-approved models and methods. The only exception is Framatome's mixed-core methodology that was used to demonstrate thermal-hydraulic compatibility of the Advanced Mark-BW fuel assembly with the resident fuel (Appendix A).

Section 4.1 describes the thermal-hydraulic models and inputs. Section 4.2 describes the Framatome statistical core design methodology and its application to North Anna. Section 4.3 describes the full core DNB analysis for the Framatome Advanced Mark-BW fuel. The results of the mixed-core DNB and hydraulic analyses are described in Section 4.4.

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

e. Containment integrity is maintained throughout the transient as demonstrated by engineering evaluation of the results of the RCS overpressurization analysis. Specifically, the RCS pressure boundary remains intact since it is not overpressurized, and mass and energy release to containment through the pressurizer safety valves and/or the pressurizer PORVs is bounded by that of the large break LOCA event.

### 6.3.4 Rod Ejection

This accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a rod cluster control assembly and drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. The description of this event and details of the analysis are unaltered from that presented in UFSAR Section 15.4.6, except as noted below. Certain other description is presented here for completeness.

These acceptance criteria for the rod ejection accident are stated in UFSAR Section 15.4.6:

1. Average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel,

2. Peak clad temperature at the hot spot below the temperature at which clad embrittlement may be expected (2,700°F),

3. Peak reactor coolant pressure less than that which would cause stresses to exceed the faulted condition stress limits, and

4. Fuel melting limited to less than 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion 1 above.

The pressure criterion in item 3 is bounded by other events. Criteria 1 and 3 are of the greatest interest for this accident and the analysis of record documented that these criteria were met.

An industry-sponsored effort under the technical direction of EPRI has investigated the behavior of irradiated cladding to reactivity initiated accidents (RIAs). The North Anna reanalysis has attempted to incorporate the insight from this effort, which is available as Reference 37. Reference 37 proposes for NRC review a set of revised regulatory criteria for use in the safety analysis of the PWR control rod ejection accident. The proposed fuel rod failure threshold is defined as the maximum radial average fuel enthalpy as a function of rod average burnup. These limits were developed based on the observations from reactivity insertion

From a neutronic standpoint, the NAIF and the Advanced Mark-BW fuel assemblies are almost identical. The structural materials within the active fuel region are similar in composition and weight. The slight differences in uranium loading will be modeled such that isotopic composition and burnup differences are properly calculated. Thus, the use of the Advanced Mark-BW assembly in conjunction with the Westinghouse NAIF assembly in the core does not adversely affect plant operation or neutronic parameters. Analyses have determined that core design models can predict core power distributions for the Advanced Mark-BW fuel to the same degree of accuracy as the Westinghouse fuel product.

Thermal-hydraulic analyses were performed to demonstrate the acceptable performance of the Framatome Advanced Mark-BW fuel in full-core and mixed-core configurations with the current NAIF fuel. The primary purpose of the thermal-hydraulic analysis is to demonstrate acceptable thermal performance that ensures fuel and clad integrity are maintained during normal operation and transients of moderate frequency. The second purpose of the hydraulic evaluation is to ensure hydraulic compatibility with the current NAIF design. Core pressure drop, hydraulic lift forces and crossflow velocities were assessed justifying acceptable mixed-core and full-core performance of the Advanced Mark-BW fuel assembly in the North Anna reactors. The thermal-hydraulic analyses were performed using NRC-approved models and methods. The only exception is Framatome's mixed-core methodology that was used to demonstrate thermal-hydraulic compatibility of the Advanced Mark-BW fuel assembly with the resident fuel (see Appendix A).

NSSS accident analyses reported in Chapter 15 of the UFSAR were evaluated for potential impact from the introduction of Advanced Mark-BW fuel. Three events were identified as requiring reanalysis: Locked Rotor/Sheared Shaft, Rod Ejection, and Complete Loss of Reactor Coolant Flow. The reanalyses of these events confirmed that all acceptance criteria will be met following introduction of the Advanced Mark-BW fuel.

A review of the long-term containment integrity shows that when the North Anna core contains Advanced Mark-BW fuel the existing analysis results remain bounding. The important aspects of the fuel change that potentially impact the analysis are the reactor coolant system average operating temperature, the core stored energy and fuel heat capacity, and the decay heat. Each aspect was reviewed and confirmed to be bounded by existing analyses.

These fuel design analyses and evaluations/review follow the content of the NRC approved topical report BAW-10172P, "Mark-BW Mechanical Design Report," as applied to the North Anna application and confirm that the Advanced Mark-BW fuel assembly maintains mechanical integrity when operated in North Anna, either as a full complement of Advanced Mark-BW assemblies, or in conjunction with the resident fuel assemblies.

29. Letter from Gordon E. Edison (U. S. Nuclear Regulatory Commission) to J. P. O'Hanlon (Virginia Electric and Power Company), "North Anna Units 1 and 2 – Issuance of Amendments Re: Demonstration Fuel Assemblies (TAC Nos. M96530 and M96531)," May 9, 1997.

30. DELETED.

31. Letter from Stephen Monarque (NRC) to David 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.

32. Engineering Calculation PM-821, "North Anna Spent Fuel Storage Calculations for 4.6 w/o LEU Fuel," Rev. 0, include Add A-D, December 3, 2001.

33. BAW-10054P, "Fuel Densification Report," Revision 2, May 1973.

34. BAW-10163P-A, "Core Operating Limits Methodology for Westinghouse Designed PWRs," Revision 0, June 1989.

35. EPRI Report NP3966-CCM, "CEPAN Method of Analyzing Creep Collapse of Oval Cladding, Volume 5: Evaluation of Interpellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods," April 1985.

36. VEP-FRD-41A, "Reactor System Transient Analyses Using the RETRAN Computer Code," May 1985.

37. "Topical Report on Reactivity Initiated Accident: Bases for RIA Fuel and Core Coolability Criteria," EPRI, Palo Alto, CA: 2002. 1002865.

38. VEP-NFE-2-A, "Vepco Evaluation of the Control Rod Ejection Transient," December 1984.

39. W. J. O'Donnell and B. F. Langer, "Fatigue Design Basis for Zircaloy Components," Nuclear Science and Engineering, Vol. 20, pp. 1-12, 1964.

40. BAW-10162P-A, "TACO3 Fuel Pin Analysis Computer Code," October 1989.

41. "ANSYS 5.6 Engineering Analysis System User's Manual, Volumes 1 and 2," SAS IP, Inc., November 1999.

42. Letter from USNRC to J. Mallay, "Acceptance for Referencing of Licensing Topical Report BAW-10199P, Addendum 2, 'Application of BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids," dated March 27, 2002.

Attachment 4

Framatome ANP Affidavit
Realistic Large Break LOCA Analysis Results – Unit 2

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

# AFFIDAVIT

COMMONWEALTH OF VIRGINIA )<br>
) ss.<br>
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 an attachment to a letter from Leslie N. Hartz to the NRC (serial number 03-313) concerning realistic LBLOCA results for the North Anna Power Station, Units 1 and 2. This attachment 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 29th
day of April, 2003.

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