VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

March 28, 2002

10 CFR 50.90

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

Serial No.	02-167
Docket Nos.	50-338
	50-339
License Nos.	NPF-4
	NPF-7

Gentlemen:

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

Pursuant to 10 CFR 50.90 and 50.12, Virginia Electric and Power Company (Dominion) requests: 1) an amendment to Facility Operating License Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, and 2) associated exemptions from 10 CFR 50.44 and 10 CFR 50.46. The amendments and exemptions will permit North Anna Units 1 and 2 to use Framatome ANP Advanced Mark-BW fuel, beginning with Cycle 17 of each unit. The Advanced Mark-BW fuel product is similar in design to the four lead test assemblies that have completed three cycles of irradiation in Unit 1. This fuel design has been evaluated by Framatome and Dominion for compatibility with the resident Westinghouse fuel and for compliance with specified acceptable fuel design limits. The attachments to this letter provide documentation of the design evaluation, in addition to proposed Technical Specifications changes and an exemption request associated with use of the Advanced Mark-BW fuel.

The Advanced Mark-BW design is an evolution of the Mark-BW design that is currently licensed for the Tennessee Valley Authority (TVA) Sequoyah Nuclear Plant Units 1 and 2. The current Westinghouse fuel product is referred to as North Anna Improved Fuel (NAIF) and is described in detail in Section 4.2 of the NAPS Updated Final Safety Analysis Report. The Advanced Mark-BW fuel is functionally equivalent to the existing NAIF fuel product. The Advanced Mark-BW fuel contains several advanced design features, including: the TRAPPER™ bottom nozzle (incorporating a debris filter), Mid-Span Mixing Grids (MSMGs), a floating intermediate grid design, a quick disconnect top nozzle, and use of the advanced zirconium-based alloy M5™ for the fuel assembly structural tubing, fuel rod cladding and grids.

Although alloy M5 is approved by the NRC for use at other utilities, the alloy does not conform with the specifications for either Zircaloy or ZIRLO. Therefore, exemptions to 10 CFR 50.44 and 10 CFR 50.46 are also required to support use of the Advanced Mark-BW fuel. The basis for the exemption from the requirements of these sections of the Code of Federal Regulations is included in Attachment 1.

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As discussed in Attachment 2, the design evaluations have concluded that the Advanced Mark-BW fuel will comply with all specified acceptable fuel design limits for operation in North Anna cores. Evaluations are underway to demonstrate the Advanced Mark-BW compliance with the ECCS requirements of 10 CFR 50.46 and will be reported in a subsequent transmittal in the 3rd quarter of 2002. Typical reload-specific core and fuel design evaluations will be conducted to confirm that these analyses remain bounding for operation within the proposed Technical Specifications limitations. The introduction of the Advanced Mark-BW fuel will not compromise the safe operation of the unit. The proposed Technical Specifications changes for use of the Advanced Mark-BW fuel design are provided in Attachment 3. It has also been determined that introduction of this fuel design does not constitute a significant hazard as defined in 10 CFR 50.92, as discussed in Attachment 4. In addition, the proposed changes have been determined to qualify for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9) as discussed in Attachment 3. The proposed changes and supporting design evaluations have been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Management Safety Review Committee.

As Attachment 2 contains information proprietary to Framatome Advanced Nuclear Power, it is supported by an affidavit (Attachment 5) signed by Framatome, the owner of the information, and a non-proprietary version of Attachment 2 (Attachment 6). In order to conform with the requirements of 10 CFR 2.790 concerning the protection of proprietary information, proprietary information is contained within brackets. Where the proprietary information has been deleted in the non-proprietary version, only the brackets will remain. Accordingly, it is requested that the information which is proprietary to Framatome be withheld from public disclosure in accordance with 10 CFR 2.790 of the Commission's regulations.

The initial reload batch of Advanced Mark-BW fuel is planned for North Anna Unit 1 Cycle 17, which is scheduled to begin operation in April 2003. To support the planned operational schedule for this cycle, we request approval of the proposed Technical Specifications changes and issuance of the necessary exemptions by January 31, 2003.

If you have any questions or require additional information on this, please contact us.

Very truly yours,

L. N. Hartz
Vice President - Nuclear Engineering

Attachments

Commitments made in this letter

1. Provide an evaluation to demonstrate the Advanced Mark-BW compliance with the ECCS requirements of 10 CFR 50.46 in the 3rd quarter of 2002.

cc: U.S. Nuclear Regulatory Commission (Attachments 1, 3, 4, and 6)
Region II
Sam Nunn Atlanta Federal Center
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Mr. J. E. Reasor, Jr. (Attachments 1, 3, 4, and 6)
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Bureau of Radiological Health
1500 East Main Street
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Richmond, VA 23218

Mr. M. J. Morgan (Attachments 1, 2, 3, and 4)
NRC Senior Resident Inspector
North Anna Power Station

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 28th day of March, 2002.

My Commission Expires: March 31, 2004.

Maggie McClure
Notary Public

(SEAL)

ATTACHMENT 1

10 CFR 50.44 and 10 CFR 50.46
Exemption Requests

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

REGULATORY BASIS AND REQUEST FOR SPECIFIC EXEMPTIONS

Virginia Electric and Power Company (Dominion) plans to refuel North Anna Units 1 and 2 with reloads of Framatome-ANP Advanced Mark-BW fuel, beginning with Cycle 17 of each unit. The Advanced Mark-BW fuel has several advanced features, including mid-span mixing grids, the advanced zirconium-based alloy M5 for the fuel rod cladding, a coarse mesh debris filter bottom nozzle, and a quick release top nozzle design. The fuel design is an evolution of the Framatome Mark-BW design, and is also similar to the resident Westinghouse fuel.

The fuel rods in the Advanced Mark-BWdesign have cladding fabricated from alloy M5. The NRC has now approved the use of this material for fuel rod cladding in reload batches of fuel (Reference 1). However, several sections of the Code of Federal Regulations continue to refer only to fuel with Zircaloy or ZIRLO cladding. Therefore, the NRC's approval for use of M5 fuel rod cladding specifically indicates that licensees should also submit exemption requests with regard to the provisions of 10 CFR 50.46, 10 CFR 50.44, and other applicable regulations that are relevant to particular fuel cladding materials.

In support of the proposed refueling of North Anna with Framatome ANP Advanced Mark-BW fuel, exemptions are hereby being requested to 10 CFR 50.46 and 10 CFR 50.44, which specifically refer to fuel with Zircaloy or ZIRLO cladding.

10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that: 1) the exemption is authorized by law, 2) the exemption will not result in an undue risk to public health and safety, 3) the exemption is consistent with the common defense and security, and 4) special circumstances, as defined in 10 CFR 50.12(a)(2) are present. The requested exemptions to allow the use of advanced zirconium alloys other than Zircaloy or ZIRLO for the fuel cladding material in the Framatome Advanced Mark-BW fuel for reloads at North Anna Units 1 and 2 satisfy these requirements as described below.

1. The requested exemption is authorized by law.

   Transition to an alternate, but equivalent fuel product is not precluded by law. The Framatome Advanced Mark-BW fuel to be irradiated at North Anna Units 1 and 2 contains cladding material that does not conform to the cladding material designations explicitly defined in 10 CFR 50.44 and 10 CFR 50.46 (i.e., Zircaloy or ZIRLO). However, the criteria of these sections will continue to be satisfied for the operation of the North Anna cores containing Advanced Mark-BW fuel.

2. The requested exemption does not present an undue risk to the public health and safety.

   The Advanced Mark-BW fuel has been evaluated for use in reloads for North Anna Units 1 and 2, to confirm that operation of the Advanced Mark-BW fuel product does not increase the probability of occurrence or the consequences of an accident at North Anna Units 1 and 2, and will not create the possibility for a new or different type of accident that could pose a risk to public health and safety. In addition, appropriate full-core and mixed core safety analyses have been performed to demonstrate that Advanced Mark-BW fuel does not present an undue risk to the public health and safety. Dominion will employ NRC approved methods for the reload design process for North Anna reload cores containing the Advanced Mark-BW fuel product.

3. The requested exemption will not endanger the common defense and security.

   The Advanced Mark-BW fuel is similar in design to normal reload fuel assemblies used at North Anna Units 1 and 2. The special nuclear material in this fuel product will continue to be handled and controlled in accordance with approved procedures. Use of Advanced Mark-BW fuel will not affect the operation of the North Anna Power Station or endanger the common defense and security.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.44 and 10 CFR 50.46.

   Pursuant to 10 CFR 50.12(a)(2), the NRC will not consider granting an exemption to the regulations unless special circumstances are present. The requested exemptions meet the special circumstances of paragraph (a)(2)(ii), in that application of these regulations in this particular circumstance is not necessary to achieve the underlying purpose of the regulations.

   - The underlying purpose of 10 CFR 50.46 is to ensure that nuclear power facilities have adequately demonstrated the cooling performance of their Emergency Core Cooling System (ECCS). The effectiveness of the ECCS at North Anna Units 1 and 2 will not be affected by the use of Advanced Mark-BW fuel. Although the fuel product incorporates cladding material other than those explicitly defined in 10 CFR 50.46, the criteria of this section will continue to be satisfied for the North Anna cores. Normal reload safety analyses will confirm that the safety analyses performed to support the use of Advanced Mark-BW fuel will remain applicable for the North Anna cores. Consequently, the use of the M5™ cladding in Advanced Mark-BW fuel will not have a detrimental impact on the performance of the North Anna cores under LOCA conditions.

- The intent of 10 CFR 50.44 is to ensure that there is an adequate means of controlling generated hydrogen following a LOCA. One source of the hydrogen produced in a post-LOCA scenario comes from a metal-water reaction. The Baker-Just equation was developed to assess the metal-water reaction rate for Zircaloy-4, but has also been confirmed to conservatively assess the metal-water reaction rates for Framatome's M5™ alloy. Therefore, the amount of hydrogen generated by metal-water reaction in this material will be within the design basis for North Anna Units 1 and 2.

Therefore, the intent of 10 CFR 50.44 and 10 CFR 50.46 will continue to be satisfied for the planned operation with Framatome Advanced Mark-BW fuel. Issuance of an exemption from the criteria of these regulations for the use of Advanced Mark-BW fuel in North Anna Units 1 and 2 will not compromise the safe operation of the reactors.

Reference 1: Letter from Stuart A. Richards (U. S. Nuclear Regulatory Commission) to T. A. Coleman (Framatome Cogema Fuels), "Revised Safety Evaluation (SE) for Topical Report BAW-10227P: 'Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel' (TAC No. M99903)," February 4, 2000.

ATTACHMENT 3

Marked-up and Proposed Technical Specifications Changes

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

Marked-up Technical Specifications Changes

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded.

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to the 95/95 DNBR criterion for the DNB correlations and methodologies specified in Section 5.6.5.

INSERT 1

2.1.1.2 ~~The peak fuel centerline temperature shall be maintained < 4700°F.~~

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

B 2.1 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

INSERT 2

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

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4.0 DESIGN FEATURES

4.1 Site Location

The North Anna Power Station is located in the north-central portion of Virginia in Louisa County and is approximately 40 miles north-northwest of Richmond, 36 miles east of Charlottesville; 22 miles southwest of Fredericksburg; and 70 miles southwest of Washington, D.C. The site is on a peninsula on the southern shore of Lake Anna at the end of State Route 700.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 157 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ~~or~~ ZIRLO or M5 fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core locations.

4.2.2 Control Rod Assemblies

The reactor core shall contain 48 control rod assemblies. The control material shall be silver indium cadmium, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.6 weight percent; R2

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) R4

a. (continued)

3. Moderator Temperature Coefficient,
4. Shutdown Bank Insertion Limits,
5. Control Bank Insertion Limits,
6. AXIAL FLUX DIFFERENCE limits, R11
7. Heat Flux Hot Channel Factor,
8. Nuclear Enthalpy Rise Hot Channel Factor,
9. Power Factor Multiplier, RAI 5.0-10 R4
10. Reactor Trip System Instrumentation - OTΔT and OPΔT Trip Parameters,
11. RCS Pressure, Temperature, and Flow DNB Limits, and RAI 5.0-10 R4
12. Boron Concentration.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. VEP-FRD-42-A, "Reload Nuclear Design Methodology."
2. WCAP-9220-P-A, "WESTINGHOUSE ECCS EVALUATION MODEL—1981 VERSION."
3. WCAP-9561-P-A, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS—SPECIAL REPORT: THIMBLE MODELING IN W ECCS EVALUATION MODEL."
4. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."
5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code."

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) R4

b. (continued)

6. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code."

7. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY—REFERENCE CORE REPORT."

8. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology."

9. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code."

10. VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."

11. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Functions." MB2073 MB2075 R11

12. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report."

Insert 3

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status. R4

TECHNICAL SPECIFICATIONS CHANGES INSERTS

Insert 1

The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup, for Westinghouse fuel and < 5173°F, decreasing by 65°F per 10,000 MWD/MTU of burnup, for Framatome fuel.

Insert 2

The maximum fuel centerline temperatures are given by the best-estimate relationships defined in SL 2.1.1.2 and are dependent upon whether Westinghouse or Framatome fuel is evaluated.

Insert 3

13. BAW-10168P-A, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants."
14. BAW-10164P-A, "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."
15. BAW-10171P-A, "REFLOD3B - Model for Multinode Core Reflooding Analysis."
16. BAW-10166P-A, "BEACH - A Computer Program for Reflood Heat Transfer During LOCA."
17. BAW-10095A, "CONTEMPT – Computer Program for Predicting Containment Pressure-Temperature Response to LOCA."
18. BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel."
19. BAW-10199P-A, "The BWU Critical Heat Flux Correlations."
20. BAW-10170P-A, "Statistical Core Design For Mixing Vane Cores."

Proposed Technical Specifications Changes

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded.

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to the 95/95 DNBR criterion for the DNB correlations and methodologies specified in Section 5.6.5.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup, for Westinghouse fuel and < 5173°F, decreasing by 65°F per 10,000 MWD/MTU of burnup, for Framatome fuel.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

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2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

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2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

B 2.1 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. The maximum fuel centerline temperatures are given by the best-estimate relationships defined in SL 2.1.1.2 and are dependent upon whether Westinghouse or Framatome fuel is evaluated. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

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BASES

BACKGROUND (continued)

weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and main steam safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System allowable values (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and flow, AFD, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the RPS and the main steam safety valves.

The SLs represent a design requirement for establishing the RPS trip allowable values identified previously (as indicated in the UFSAR, Ref. 2). LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The figure provided in the COLR shows the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below

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BASES

SAFETY LIMITS (continued)

melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS and main steam safety valves ensures that for variations in the THERMAL POWER, RCS pressure, RCS average temperature, RCS flow rate, and AFD that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The main steam safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Allowable values for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

4.0 DESIGN FEATURES

4.1 Site Location

The North Anna Power Station is located in the north-central portion of Virginia in Louisa County and is approximately 40 miles north-northwest of Richmond, 36 miles east of Charlottesville; 22 miles southwest of Fredericksburg; and 70 miles southwest of Washington, D.C. The site is on a peninsula on the southern shore of Lake Anna at the end of State Route 700.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 157 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO or M5 fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core locations.

4.2.2 Control Rod Assemblies

The reactor core shall contain 48 control rod assemblies. The control material shall be silver indium cadmium, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.6 weight percent;

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. (continued)

3. Moderator Temperature Coefficient,
4. Shutdown Bank Insertion Limits,
5. Control Bank Insertion Limits,
6. AXIAL FLUX DIFFERENCE limits,
7. Heat Flux Hot Channel Factor,
8. Nuclear Enthalpy Rise Hot Channel Factor,
9. Power Factor Multiplier,
10. Reactor Trip System Instrumentation - OTΔT and OPΔT Trip Parameters,
11. RCS Pressure, Temperature, and Flow DNB Limits, and
12. Boron Concentration.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. VEP-FRD-42-A, "Reload Nuclear Design Methodology."
2. WCAP-9220-P-A, "WESTINGHOUSE ECCS EVALUATION MODEL—1981 VERSION."
3. WCAP-9561-P-A, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS—SPECIAL REPORT: THIMBLE MODELING IN W ECCS EVALUATION MODEL."
4. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."
5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

b. (continued)

6. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code."

7. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY—REFERENCE CORE REPORT."

8. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology."

9. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code."

10. VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."

11. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function."

12. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report."

13. BAW-10168P-A, "RSG LOCA—BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants."

14. BAW-10164P-A, "RELAP5/MOD2-B&W—An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

15. BAW-10171P-A, "REFLOD3B—Model for Multinode Core Reflooding Analysis."

16. BAW-10166P-A, "BEACH—A Computer Program for Reflood Heat Transfer During LOCA."

17. BAW-10095A, "CONTEMPT—Computer Program for Predicting Containment Pressure-Temperature Response to LOCA."

18. BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel."

19. BAW-10199P-A, "The BWU Critical Heat Flux Correlations."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

b. (continued)

20. BAW-10170P-A, "Statistical Core Design For Mixing Vane Cores."

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Steam Generator Tube Inspection Report

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Nuclear Regulatory Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported to the NRC by March 1 of each year for the previous calender year. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections that fall into Category C-3 require prompt notification of the Commission pursuant to Section 50.72 to 10 CFR Part 50. A Licensee Event

(continued)

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

c. (continued)

Report shall be submitted pursuant to Section 50.73 to 10 CFR Part 50 and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

ATTACHMENT 4

Significant Hazards Consideration Determination

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

SIGNIFICANT HAZARDS CONSIDERATION

Virginia Electric and Power Company (Dominion) plans to refuel North Anna Units 1 and 2 with reloads of Framatome-ANP Advanced Mark-BW fuel, beginning with Cycle 17 of each unit. The Advanced Mark-BW fuel has several advanced features, including mid-span mixing grids, the advanced zirconium-based alloy M5 for the fuel rod cladding, a coarse mesh debris filter bottom nozzle, and a quick release top nozzle design. The fuel design is an evolution of the Framatome Mark-BW design, and is also similar to the resident Westinghouse fuel.

From the evaluation presented in this report, it is concluded that the operation with Advanced Mark-BW fuel does not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for this determination is delineated below:

1. The probability of occurrence or the consequences of an accident previously evaluated is not significantly increased. The Advanced Mark-BW fuel is very similar in design to the Westinghouse fuel that is being replaced in the core. The reload core designs for North Anna cycle will meet all applicable design criteria. Dominion will use the NRC-approved standard reload design models and methods to demonstrate that all applicable design criteria and all pertinent licensing basis criteria will be met. Evaluations will be performed as part of the cycle specific reload safety analysis to confirm that the existing safety analyses remain applicable for operation of the Framatome Advanced Mark-BW fuel. Operation of the Advanced Mark-BW fuel will not result in a measurable impact on normal operating plant releases, and will not increase the predicted radiological consequences of accidents postulated in the UFSAR. Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated is significantly increased.

2. The possibility for a new or different type of accident from any accident previously evaluated is not created. The Framatome Advanced Mark-BW fuel is very similar in design (both mechanical and composition of materials) to the resident Westinghouse fuel. The North Anna core in which the fuel operates will be designed to meet all applicable design criteria and ensure that all pertinent licensing basis criteria are met. Demonstrated adherence to these standards and criteria precludes new challenges to components and systems that could introduce a new type of accident. North Anna safety analyses have demonstrated in Section 6.0 of this report that the use of Advanced Mark-BW fuel is acceptable. All design and performance criteria will continue to be met and no new single failure mechanisms will be created. The use of the Advanced Mark-BW fuel does not involve any alteration to plant equipment or procedures which would introduce any new or unique operational modes or accident precursors. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3. The margin of safety is not significantly reduced. The operation of Advanced Mark-BW fuel does not change the performance requirements on any system or component

such that any design criteria will be exceeded. The normal limits on core operation defined in the North Anna Technical Specifications will remain applicable for the use of Advanced Mark-BW fuel. The reload core designs for the cycles in which the Advanced Mark-BW fuel will operate will specifically evaluate any pertinent differences between the Advanced Mark-BW fuel product and the current Westinghouse fuel product, including both the mechanical design differences and the past irradiation history. The use of Advanced Mark-BW fuel will be specifically evaluated during the reload design process using Dominion's reload design models and methods approved by the NRC. North Anna safety analyses have demonstrated in Section 6.0 of this report that the use of Advanced Mark-BW fuel is acceptable. Therefore, the margin of safety as defined in the Bases to the North Anna Units 1 and 2 Technical Specifications is not significantly reduced.

Based on the above information, the use of Framatome Advanced Mark-BW fuel at North Anna, and design of cores containing this fuel using Dominion reload design methodology previously approved by the NRC, will not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. It is concluded that the proposed use of the Advanced Mark-BW fuel meets the requirements of 10 CFR 50.92(c) and does not involve a significant hazards consideration.

ATTACHMENT 5

Framatome Advanced Nuclear Power Affidavit

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 ("FRA-ANP"), and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the information contained in the attachment to a letter from Framatome ANP to Dominion Generation of March 20, 2002 (BDC 02-216). This attachment, which contains information developed by Framatome ANP, is referred to herein as "Document." Some of the information contained in this Document has been classified by FRA-ANP as proprietary in accordance with the policies established by FRA-ANP 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 FRA-ANP 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 FRA-ANP to determine whether information should be classified as proprietary:

(a) The information reveals details of FRA-ANP'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 FRA-ANP.

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

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

7. In accordance with FRA-ANP'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 FRA-ANP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. FRA-ANP 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.

James F. Malley

SUBSCRIBED before me this 20th day of March, 2002.

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

ATTACHMENT 6

Non Proprietary Version of Discussion of Change

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

DISCUSSION OF CHANGES

INTRODUCTION

Virginia Electric and Power Company (Dominion), operator of the North Anna Power Station (North Anna), proposes to change the fuel supplier for both Units 1 and 2 from Westinghouse to Framatome ANP, Inc. [1] Framatome will deliver fuel assemblies of the Advanced Mark-BW design to Dominion, beginning with Cycle 17 for each unit. The first delivery is scheduled for Unit 1 in early 2003. The Advanced Mark-BW fuel design is compatible with previously licensed fuel operated at North Anna. This design is an evolution of that currently licensed for the Tennessee Valley Authority (TVA) Sequoyah Nuclear Plant Units 1 and 2. This report provides a description of the fuel product design and the analyses that support operation of the Advanced Mark-BW product for North Anna. It provides the necessary justification to demonstrate continued regulatory compliance for operation with full batches of the Advanced Mark-BW fuel.

The current North Anna fuel product is referred to as North Anna Improved Fuel (NAIF) and is described in detail in Section 4.2 of the North Anna Updated Final Safety Analysis Report (UFSAR). The Framatome fuel, denoted Advanced Mark-BW, is functionally equivalent to the existing NAIF fuel product. The Framatome supplied fuel assemblies will be very similar to the Framatome Advanced Mark-BW fuel assembly design that has previously been irradiated in other Westinghouse-designed reactors. The Advanced Mark-BW fuel contains several advanced design features, including: the TRAPPER™ bottom nozzle (incorporating a coarse mesh debris filter), Mid-Span Mixing Grids (MSMGs), a floating intermediate grid design, a quick disconnect top nozzle, and use of an advanced zirconium-based alloy (designated as M5™) for the fuel assembly structure. The fuel rod cladding in these assemblies will also be fabricated from M5™.

The proposed technical specification changes and exemptions to the Code of Federal Regulations have been reviewed, and have been determined to qualify for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Therefore, no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed technical specification changes and exemptions.

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

Use of the Advanced Mark-BW fuel design at North Anna will require several revisions to the existing plant Technical Specifications. These changes are largely administrative in nature, involving such areas as descriptive details of the fuel design or addition of references that support the Core Operating Limits Report (COLR). The specific proposed changes are provided below. These changes are noted with respect to the North Anna 1 & 2 Improved Technical Specifications as submitted for NRC staff review and approval in Reference 1, as

1 Framatome ANP, Inc. will be identified simply as "Framatome" for the remainder of this report.

supplemented in Reference 2.

TS 2.1.1, Reactor Core SLs

This specification is revised to state separate safety limits (SLs) to preclude centerline fuel melting that are applicable to the resident (Westinghouse fuel) and the Advanced Mark-BW fuel products. The proposed change presents a best-estimate fuel melt value for unirradiated fuel, with an adjustment to account for the effects of fuel burnup. The differences in melt values for the fuel types reflect inherent differences between vendor methodologies for characterizing fuel melt behavior. The revised specification reads as follows:

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup, for Westinghouse fuel and < 5173°F, decreasing by 65°F per 10,000 MWD/MTU of burnup, for Framatome fuel.

Bases, TS 2.1.1.2

The Bases for TS 2.1.1.2 are augmented to provide background information concerning the fuel centerline melt relationships presented in this specification. The following statement is added to the existing second paragraph of the Background section:

> The maximum fuel centerline temperatures are given by the best-estimate relationships defined in SL 2.1.1.2 and are dependent upon whether Westinghouse or Framatome fuel is evaluated.

TS 4.2.1, Fuel Assemblies

This section is revised to add the Framatome alloy M5 to the list of cladding materials that may be present in North Anna fuel assemblies. The revised specification reads as follows:

> The reactor shall contain 157 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO or M5 fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core locations.

TS 5.6.5b, CORE OPERATING LIMITS REPORT (COLR)

This section is revised to include modifications of existing references and additional references that reflect the proposed changes above. Most of the additional references describe the analytical methods used in determining core limits that are applicable to the Advanced Mark-BW fuel product. The following revisions and/or additions are proposed:

1. VEP-FRD-42-A, "Reload Nuclear Design Methodology."

13. BAW-10168P-A, “RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants.”

14. BAW-10164P-A, “RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis.”

15. BAW-10171P-A, “REFLOD3B - Model for Multinode Core Reflooding Analysis.”

16. BAW-10166P-A, “BEACH - A Computer Program for Reflood Heat Transfer During LOCA.”

17. BAW-10095A, “CONTEMPT – Computer Program for Predicting Containment Pressure-Temperature Response to LOCA.”

18. BAW-10227P-A, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel.”

19. BAW-10199P-A, “The BWU Critical Heat Flux Correlations.”

20. BAW-10170P-A, “Statistical Core Design For Mixing Vane Cores.”

<u>Reactor Trip System Changes</u>

The f(ΔI) reset function for the thermal overtemperature ΔT trip function requires two modifications. The first modification consists of a change in the value at which the negative end of the deadband begins. This value is changed from –44% to –35% axial flux difference (i.e., ΔI). The corresponding value for safety analyses is changed from –47% to –38% axial flux difference. No changes were required to the positive end of the deadband or to the negative and positive runback ramp-rates.

The second modification to the f(ΔI) reset function alters the maximum allowed penalty value obtained from the f(ΔI) reset function for highly top-skewed power distributions (positive ΔI). This change extends the range of the f(ΔI) reset function generator to accommodate axial flux differences between –50% and +50% (versus –50% and +28% currently). These changes are discussed in Section 4.5 of this report.

REQUEST FOR EXEMPTIONS

Several sections of the Code of Federal Regulations discuss the cladding material. Specifically, Title 10 CFR 50.46(a)(1)(i) states that:

> *"Each boiling and pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated."*

Section 10 CFR 50.46 goes on to delineate specifications for peak cladding temperature, maximum cladding oxidation maximum hydrogen generation, coolable geometry, and long-term cooling.

Also, 10 CFR 50.44 (a) states that

> *"Each boiling or pressurized light-water nuclear power reactor fueled with oxide pellets with cylindrical zircaloy or ZIRLO cladding, must, as provided in paragraphs (b) through (d) of this section, include means for control of hydrogen gas that may be generated, following a postulated loss-of-coolant accident (LOCA)..."*

Since 10 CFR 50.46 and 10 CFR 50.44 specifically refer to fuel with Zircaloy or ZIRLO cladding, the use of fuel with alloys that do not conform to either of these two designations will require exemptions from these sections of the code.

Title 10 CFR 50, Appendix K, paragraph I.A.5 states,

> *"The rate of energy release, hydrogen generation, and cladding oxidation from the metal water reaction shall be calculated using the Baker-Just equation."*

Since the Baker-Just equation was originally developed for the use of Zircaloy cladding, the use of the Advanced Mark-BW fuel assemblies with M5™ advanced alloy will require an exemption from this section of the code. Framatome has conducted high temperature oxidation testing to demonstrate that the Baker-Just equation can be used to conservatively predict the metal-water reaction rates for M5™ (Reference 3). These test results demonstrate generically that this section of 10 CFR 50 Appendix K is applicable to M5™. Since this result is documented in the approved topical report for M5™ (Reference 3), it is unnecessary to obtain an exemption from 10 CFR 50 Appendix K for irradiation of the Advanced Mark-BW fuel at North Anna.

Exemptions to 10 CFR 50.46(a)(1)(i) and 10 CFR 50.44 (a) are therefore requested because the specific wording regarding cladding materials for which these regulations are applicable does not accommodate use of M5™. Nevertheless, Framatome test results indicate the intent of these regulations will still be satisfied for operation of the Advanced Mark-BW fuel at North Anna containing fuel rods with M5™. Specifically:

- The underlying purpose of 10 CFR 50.46 is to ensure that nuclear power facilities demonstrate adequate performance for their ECCS. This is demonstrated by performing detailed analyses of the physical phenomena for LOCA events, and evaluating the results with respect to the acceptance criteria in 10 CFR 50.46(b). Reference 3 provides the basis for concluding that the criteria of 10 CFR 50.46(b) apply to M5™. The effectiveness of the ECCS at North Anna will be confirmed by plant-specific analyses that model the behavior of the Advanced Mark-BW fuel.

- The intent of 10 CFR 50.44 is to ensure that there is an adequate means of controlling generated hydrogen. One source of the hydrogen produced in a post-LOCA scenario comes from the metal-water reaction. The Baker-Just equation was developed to assess the metal-water reaction rate for Zircaloy-4, but has also been confirmed to conservatively assess the metal-water reaction rates for M5™. Therefore, the amount of hydrogen generated by metal-water reaction in these materials will be within the design basis for North Anna, and existing plant specific analyses for the total hydrogen generation following a LOCA will remain applicable for a core containing Advanced Mark-BW fuel.

Therefore, the intent of 10 CFR 50.46, 10 CFR 50.44, and 10 CFR Part 50 Appendix K will continue to be satisfied for the planned operation of the Advanced Mark-BW fuel at North Anna, without compromising the safe operation of the reactors.

SAFETY SIGNIFICANCE SUMMARY

The introduction of reload batches of Framatome Advanced Mark-BW fuel has been evaluated to confirm that it can be successfully operated in North Anna Units 1 and 2. The key areas evaluated include mechanical, thermal/hydraulic and accident behavior of the fuel product. This evaluation, performed by Framatome and Dominion, has demonstrated that all fuel assembly and fuel rod design criteria are met for operation of either full cores of Advanced Mark-BW or mixed cores with the current Westinghouse fuel product. The Advanced Mark-BW fuel is functionally equivalent to the existing fuel, particularly in terms of core neutronic behavior. The assessment has demonstrated that the normal Dominion reload evaluation methodology and tools can be used to model either fuel type. This evaluation has identified changes to the overtemperature ΔT protection system setpoint that are necessary to ensure that adequate thermal margin exists for the Advanced Mark-BW fuel under all potential core power distributions. A similar change has been incorporated in prior transitions to this fuel type at other Westinghouse plants. The evaluation results have confirmed that operation of the Advanced Mark-BW fuel at North Anna within current Technical Specifications limits (as modified) will ensure that the specified acceptable fuel design limits are met. A cycle specific nuclear design evaluation will be performed to demonstrate that cores containing the Advanced Mark-BW fuel will meet all applicable design criteria, and will not adversely impact plant operation. This evaluation is conducted as part of the routine Dominion reload evaluation process.

EVALUATION OF USE OF FRAMATOME ANP ADVANCED MARK-BW FUEL AT NORTH ANNA POWER STATION

DOMINION GENERATION
MARCH 2002

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1.0 Fuel Assembly Design Description

Figure 1.0-1 shows the Advanced Mark-BW fuel assembly. The individual components and features are described below. Table 1.0-1 lists key parameters of the Advanced Mark-BW fuel assemblies, and the North Anna Improved Fuel (NAIF) designs.

1.1 Quick Connect/Disconnect Top Nozzle

The Advanced Mark-BW fuel assembly design incorporates a reconstitutable, quick disconnect top nozzle assembly. The primary distinguishing features of the Advanced Mark-BW top nozzle assembly includes:

- Quick Disconnect Guide Thimble Attachment Features
- Three-Leaf Holddown Spring System
- Low Pressure Drop Nozzle Structure

A mechanical joint provides the structural connection between the top nozzle and the guide thimbles. The quick disconnect (QD) feature allows for rapid removal and reattachment of the top nozzle with no loose parts.

The Advanced Mark-BW top nozzle assembly incorporates four sets of 3-leaf springs made of Inconel-718 alloy fastened to the nozzle with Inconel-718 clamp screws. During operation the springs prevent fuel assembly lift due to hydraulic forces. The upper leaf has an extended tang that engages a cutout in the top plate of the nozzle. This arrangement assures spring leaf retention in the unlikely event of a spring failure.

The Advanced Mark-BW top nozzle assembly is a box-like structure made of stainless steel. The flow hole pattern provides increased flow area over traditional designs yielding a reduced pressure drop, while satisfying the same strength requirements.

1.2 Guide Thimble and Instrumentation

The Advanced Mark-BW guide and instrument thimbles are fabricated from M5™. The Advanced Mark-BW guide thimble has two inner diameters. The larger diameter at the top provides a relatively large annular clearance that permits rapid insertion of the rod cluster control assembly (RCCA) during a reactor trip and accommodates coolant flow during normal operation. A reduced diameter section, at the lower end of the tube provides a dashpot action that decelerates the control rods near the end of the control rod travel during a reactor trip. This deceleration limits the magnitude of the RCCA impact loads on the top nozzle. Four (4) small holes located just above the dashpot allow both outflow of the water during RCCA insertion and coolant flow to control components during operation.

The quick disconnect sleeve is attached to the upper end of the guide thimble tube for connection to the Top Nozzle (see Section 1.1). An M5™ lower end plug is welded onto the end of the guide thimble dashpot section. The lower end plug is internally threaded for engagement with the guide thimble bolt which connects the guide thimble to the bottom nozzle. A small flow hole in the guide thimble bolt provides flow through the reduced diameter section.

1.3 Spacer Grids

The Advanced Mark-BW design incorporates both intermediate and end spacer grid assemblies. The end grids use low cobalt precipitation hardened Inconel-718. This material ensures proper grip of the fuel rod through the licensed burnup. The six intermediate spacer grids in the active fuel region are made of M5™ with inherently low growth and good corrosion performance. In addition, the Advanced Mark-BW fuel assemblies utilize two types of intermediate spacer grid assemblies: vaned and vaneless. Five intermediate vaned (mixing) grids are used in the high heat flux region of the fuel assembly to promote mixing of the coolant. The vaned-grid incorporates mixing vanes in the strip, projecting from the trailing (upper) edges into the coolant stream. DNB performance is discussed further in Section 4.

For each grid type there are various features press-formed into the grid strip. For a given fuel rod cell, a combination of springs (softstops) and dimples (hardstops) acting in two orthogonal planes support each rod. All spring and dimple edges are bent inward (i.e. coined) to avoid scratching of fuel rods during loading. Tight control of dimple and spring heights ensures a constant, uniform rod pitch and fuel rod restraint load.

Each guide and instrumentation thimble cell features saddles and scallops facilitating loading and support of the thimbles. A weld, performed at each strip intersection on both faces of the assembled grid, secures the strips. Grid strip height and thickness are optimized to meet crush and impact strength, pressure drop, and geometry requirements.

The Advanced Mark-BW outer strip design of the intermediate and end grids incorporates many handling enhancement features, which include:

- Welded, reinforced guide vane
- Dimpled, reinforced outer strip
- Column-structured corner

To facilitate fuel assembly loading and unloading of the core, the outer grid strips have generous lead-in vanes that aid in guiding the grids and fuel assemblies past projecting surfaces. To strengthen the lead-in vanes, a welded tab-slot joint connects the guide vane to the inner strip. The outer strips also have press-formed stiffening dimples that provide added strength to resist tearing. The recess of the stiffening dimple is also used as a weld land for the inner and outer strip connection, eliminating any exposed edges. The outer strip corner joint is a welded, lapped joint carefully dressed to remove weld buildup and minimize distortion. The outer grid corner also incorporates a structural support column, which increases the corner strength. The grid corner strength is designed to exceed normal handling equipment limits.

The Advanced Mark-BW intermediate spacer grid restraint system allows for floating grid assemblies, which permits a limited amount of upward motion of the grids. The intermediate spacer grids are not rigidly attached to the guide thimbles, but are allowed to follow the fuel rods as they grow due to irradiation until burnup effects have significantly relaxed the M5™ spacer grids. To ensure axial alignment of spacer grids with adjacent fuel assemblies, the design incorporates stops on selected guide thimbles that limit grid movement after irradiation relaxation of the intermediate M5™ grids. The stops are short sleeves or ferrules attached to the guide thimble above each intermediate grid. Eight (8) restraining guide thimbles, in addition to the center cell, are utilized for the intermediate grid restraint system to provide sufficient structural margins. The sleeve below the grid on the instrument sheath (center cell) is attached with two sets of swages or dimples to provide enough strength to prevent the grid from moving downward during fuel assembly handling.

The top and bottom end grid restraint systems employ short stainless steel sleeves that are attached to weld tabs at the guide thimble locations. On the upper end grid, these sleeves are attached to the top side of the grid, and on the bottom end grids the sleeves are attached to the lower side of the grid.

The top end grid sleeves seat against the bottom surface of the quick disconnect (QD) sleeve. The QD sleeves restrain the grid as the fuel rods slip through due to irradiation growth.

For the bottom end grid connection, mechanical crimping of the end grid sleeves into circular grooves in the guide thimble bottom end plugs attaches the grid to the guide thimble assembly.

The Advanced Mark-BW intermediate and end grids incorporate keying windows which allow 100% of the fuel rod cells to be opened, or "keyed", during fuel rod insertion. The keying process comprises thin keys, inserted through the keying windows, which are rotated to restrain the soft stop springs. This process is utilized to minimize fuel rod scratches, cell hardstop/softstop damage, and fuel assembly residual stresses. The keys are removed after fuel rod insertion to restore the grid's grip force on the fuel rods.

1.4 Mid-Span Mixing Grids (MSMG)

The Advanced Mark-BW design also incorporates mid-span mixing grid assemblies (MSMGs). Three (3) MSMGs are incorporated onto each fuel assembly, one at each mid-span between the upper four (4) intermediate vaned grids. The MSMGs provide additional flow mixing in the high heat flux region for improved DNB performance.

Constructed from M5™, the individual strips are slotted and assembled in an egg-crate fashion and welded at each of the grid strip intersections, the same as for the intermediate grid design. Stops formed in each of the four cell walls prevent the fuel rods from contacting the mixing vanes but impose no grip force (or slip load) onto the rods; thus, these are designated "non-contacting" grids. The outer strips incorporate a wrap-around corner design to improve the corner handling interface.

The outer strip design precludes hang-up or damage during handling due to its large lead-in feature. A reduced grid envelope eliminates mechanical interaction with adjacent fuel assemblies during transition fuel cycles.

The MSMGs use the same mixing vane design and pattern as utilized on the Advanced Mark-BW intermediate vaned grid.

The MSMGs are attached to the guide thimbles at the sixteen (16) outside corner guide thimble locations. These guide thimble locations are different than the eight (8) restraining guide thimble locations for the floating intermediate grids in order to help distribute the hydraulic loads. The MSMGs are rigidly attached to the guide thimbles, as opposed to the floating grid concept, since they are non-contacting (i.e. no axial support from the fuel rods).

1.5 Debris Filter Bottom Nozzle

The Advanced Mark-BW incorporates the TRAPPER™ bottom nozzle, which is designed with debris-resistant features.

The stainless steel bottom nozzle consists of a frame of deep ribs connecting the guide thimble locations and conventional legs that interface with the reactor internals. The frame distributes the primary loads on the fuel assembly through the bottom nozzle. A high strength A-286 alloy filter plate is attached to the top of the frame. Upon skeleton assembly, the guide thimble lower end plugs serve to clamp the filter plate to the structural frame at internal locations. The filter plate serves two functions. First, it provides the axial restraint for fuel rods, which are seated on the filter plate, by distributing these loads to the structural frame. Secondly, it provides a very effective barrier to debris while maintaining acceptable pressure drop.

1.6 Fuel Rods

The Advanced Mark-BW fuel rod assembly design features M5™ cladding, which significantly increases protection from corrosion associated with long cycles, high temperatures, and high burnup.

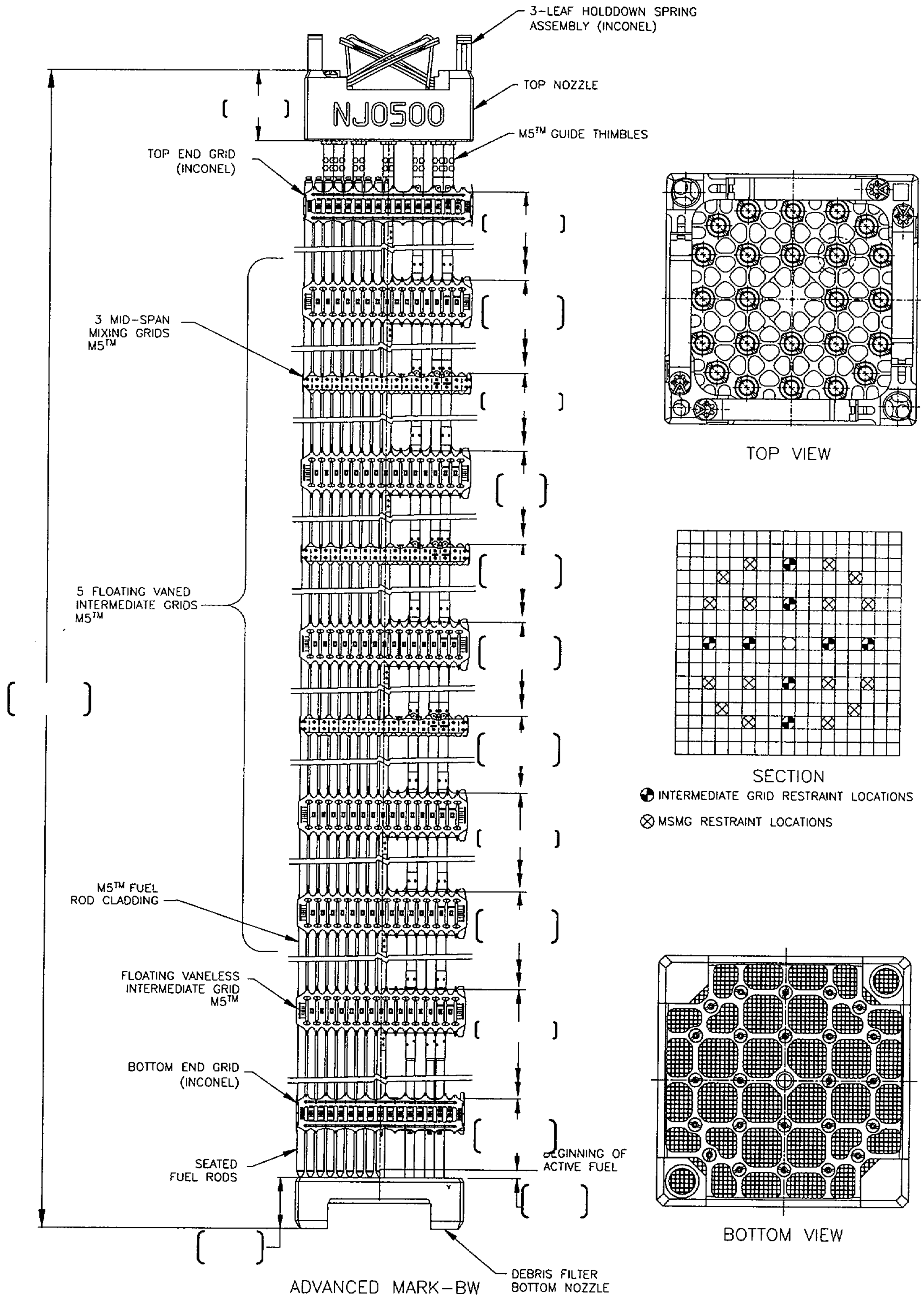
The fuel rod design consists of UO_2 pellets contained in a seamless M5™ tube with M5™ end caps welded at each end. The design utilizes a 144.0 inch fuel stack length. The fuel pellets have a diameter of [ ] inches. The fuel rod cladding has a 0.374 inch outside diameter and a [ ] inch wall thickness. This configuration leaves a small clearance between the inside diameter of the cladding and the outside diameter of the fuel pellets. The fuel rod utilizes one stainless steel spring in the upper plenum to prevent the formation of fuel stack gaps during shipping and handling, while also allowing for the expansion of the fuel stack during operation. The fuel stack rests on the lower end cap. The lower end cap is made from M5™ and has a bullet nose shape to provide a smooth flow transition in addition to facilitating reinsertion of the rods into the assembly if any rods are removed after the assemblies have been irradiated (e.g., during fuel examination programs). The upper end cap is also made of M5™ and has a grippable top hat shape that allows for the removal of the fuel rods from the fuel assembly if necessary.

The fuel pellets are a sintered ceramic, comprised of high density UO_2. The fuel pellets are cylindrically shaped with a spherical dish at each end. The corners of the pellets have an outward land taper (chamfer) that eases the loading of the pellets into the cladding. The dish and taper geometry also reduces the tendency for the pellets to assume an hourglass shape during operation. The nominal design density of the pellets is 96% Theoretical Density (TD).

Table 1.0-1: Comparison of Advanced Mark-BW and W NAIF

Parameter	Advanced Mark-BW Design	W NAIF Design
Fuel Assembly Length, in.	[ ]	159.8
Fuel Rod Length, in.	[ ]	152.60
Assembly Envelope, in.	8.425	8.426
Fuel Rod Pitch, in.	0.496	0.496
Number of Fuel Rods/Ass'y	264	264
Number of Guide Thimbles/Ass'y	24	24
Number of Instrumentation Tube/Ass'y	1	1
Fuel Tube Material	M5™	ZIRLO™
Fuel Rod Clad O.D., in.	0.374	0.374
Guide Thimble Material	M5™	ZIRLO™
Inner Diameter of Guide Thimbles (upper part), in.	[ ]	0.442
Outer Diameter of Guide Thimbles (upper part), in.	[ ]	0.474
Inner Diameter of Guide Thimbles (lower part), in.	[ ]	0.397
Outer Diameter of Guide Thimbles (lower part), in.	[ ]	0.430
Inner Diameter of Instrument Guide Thimbles, in.	[ ]	0.442
Outer Diameter of Instrument Guide Thimbles, in.	[ ]	0.474
Composition of Grids		
End Grids	2 Inc.-718	2 Inc. –718
Mixing Vane Grids	5 M5™	6 ZIRLO™
Non-Mixing Grids	1 M5™	N/A
Mid-Span Mixing Grids	3 M5™	N/A

Figure 1.0-1: Advanced Mark-BW Fuel Assembly Outline



2.0 Advanced Materials (M5™) and Post Irradiation Examination (PIE)

2.1 M5™ Introduction

M5™ is the advanced zirconium alloy employed by Framatome in the fabrication of fuel rod cladding and fuel assembly structural material. It is a proprietary variant of the Zr1Nb alloy. The use of M5™ provides performance improvements in these areas: corrosion, hydrogen pickup, axial growth, and diametral creep. Detailed evaluations of M5™ behavior have been performed by Framatome and approved for reload applications by NRC, as documented in Reference 3.

2.2 Fuel Assembly Testing

A comprehensive test program summarized in Table 2.3-1 was conducted to characterize and verify the structural and hydraulic performance of the Advanced Mark-BW fuel assembly design. The Advanced Mark-BW design is also supported by extensive testing of the standard Mark-BW design, which is described in detail in Reference 4. In addition, all structural and functional testing has been verified by in-reactor operation of over 2400 Mark-BW fuel assemblies.

Table 2.3-1: Advanced Mark-BW Test Summary

TEST	INFORMATION OBTAINED
FA Prototype Static Axial Compression Test	- FA axial stiffness under compression - FA stability - GT load distribution -GT stresses
FA Prototype Static Lateral Bending Test	- FA lateral stiffness - GT stresses
FA Prototype Natural Frequency & Mode Shape Test (Shaker)	- FA first six natural frequencies and mode shapes - FA damping
FA Prototype Lateral Pluck w/o Impact Test	- FA frequency and damping versus displacement amplitude
FA Prototype Lateral Pluck w/ Impact Test	- FA spacer grid internal stiffness and damping - FA spacer grid impact force versus displacement
FA Prototype Axial Drop Test	- FA impact force versus displacement - FA impact force versus impact velocity - GT stresses
FA Prototype Axial Tension Test	- FA axial stiffness under tension - GT load distribution - GT stresses
FA Spacer Grid Static Crush Test	- SG static crush load to cause failure - SG elastic spring rate - SG failure mode - SG crush and recovery height
FA Spacer Grid Dynamic Crush Test	- SG dynamic crush load to cause failure - SG damping - SG post-buckling behavior
FA HD Spring Compression Test	- HD spring load/deflection characteristic - Max. HD spring deflection - Max./Min. HD loads
FA ΔP Test FA Component ΔP Test	- FA pressure drop - Grid - Nozzle
FA Prototype Life and Wear Test	- FA 1000 hour endurance - corrosion & wear - RCCA drop times - Endurance under RCCA Stepping/Stroking
FA Flow-Induced Vibration Test	- Flow-induced behavior of prototype X1 and Mark-BW fuel assemblies
Bottom Nozzle Test	- BN pressure drop - BN debris filtering effectiveness - BN mechanical strength

3.0 Mechanical Design Evaluation

This section presents the structural evaluation to ensure the Advanced Mark-BW fuel assemblies meet all applicable structural criteria to maintain safe plant operation with a coolable geometry under all plant design conditions. The Advanced Mark-BW fuel assembly is evaluated to ensure acceptable operation under the loading associated with the normal operation, seismic and loss-of-coolant-accident (LOCA) events, and shipping and handling events. Current state-of-the-art methods are used in the structural analyses. Methodologies for the fuel assembly faulted structural evaluations are described in References 5 and 6. These topical reports have received NRC approval for referencing in licensing applications. The fuel assembly structural evaluation is based on the Standard Review Plan, NUREG-0800 and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

The scope of this evaluation includes both Advanced Mark-BW fuel assemblies and resident Westinghouse fuel assemblies to demonstrate that both fuel types will continue to meet licensing basis requirements following the introduction of Advanced Mark-BW. The design of the Advanced Mark-BW fuel assembly is compatible with resident fuel assemblies and all reactor internals and all equipment for normal handling. This evaluation demonstrates that the Advanced Mark-BW design can operate in either mixed cores with the resident NAIF fuel or full cores and satisfy the licensing basis requirements for all reactor internal components. Detailed calculations of relevant interface loads were performed to support these conclusions.

3.1 LTA Experience

The Advanced Mark-BW LTA program was a cooperative effort with Dominion to thoroughly test the design prior to batch implementation. Four Mark-BW LTAs were inserted in North Anna Unit 1 in 1997. The LTAs completed three cycles of operation with no cladding leakage with a peak pin burnup of approximately 56,000 MWD/MTU.

Figure 3.1-1 provides the core locations and corresponding maximum fuel rod burnup for the LTAs for each irradiation cycle. The LTAs were operated for two cycles in typical interior core locations and for the third cycle on the core periphery. The core periphery is known to create a hydraulic environment that has resulted in flow-induced vibration failures in resident fuel assemblies on several occasions. In-mast sipping during the fuel off-load confirmed that all lead test assemblies were free of cladding failure following the third irradiation cycle. RCCA insertion data was obtained prior to the second irradiation cycle in which the LTAs operated in core locations with RCCAs. Measured RCCA drop times to dashpot entry were well within the acceptance criterion. In addition, no LTA handling or operational problems occurred.

3.2 Fuel Assembly Compatibility

The Advanced Mark-BW fuel assemblies are designed for full compatibility with the North Anna mechanical interfaces including:

- Core Internals
- Control Components
- Resident Fuel
- Shipping and Handling

Component dimensional analyses were performed for the Advanced Mark-BW fuel assemblies in each of the interface areas above to ensure the functional compatibility of the fuel assemblies in the North Anna reactor environment. Direct measurements of Westinghouse standard LOPAR and Vantage 5H 17x17 fuel assemblies (made in support of other Mark-BW fuel assembly reload contracts) were also used as input for the compatibility analyses. The compatibility evaluation also involved a detailed dimensional comparison between Advanced Mark-BW fuel and Westinghouse resident NAIF fuel.

3.3 Structural Integrity

3.3.1 Normal Operation

The evaluations performed to verify the structural integrity of the Advanced Mark-BW fuel assembly components are presented in the following sections. Table 3.3-1 summarizes the design limits and margins for each of the major Advanced Mark-BW structural components under normal operating loads.

3.3.1.1 Fuel Assembly Holddown Springs

The design bases for the Advanced Mark-BW fuel holddown springs require that the springs be capable of maintaining fuel assembly contact with the lower core plate during normal operating conditions. During a pump overspeed condition, the fuel assembly should not cause the springs to deflect to the solid state nor produce any permanent set.

The Advanced Mark-BW holddown springs were analyzed to show that the holddown springs can accommodate irradiation growth of the fuel assembly and the differential thermal expansion between the fuel assembly and the core internals. The fuel assembly lift evaluation was performed by comparing the holddown force provided by the leaf springs with that of the North Anna hydraulic forces at both normal operating conditions and at the pump overspeed condition. The analysis results, which bounds the transition core configurations, confirmed that the Advanced Mark-BW assembly meets the required design criteria.

The Advanced Mark-BW holddown spring stress calculations show that the springs are structurally adequate under all static and fatigue loading conditions.

Furthermore, in the unlikely event of spring failure, the top nozzle provides positive retention of the holddown springs. The operational performance of the holddown springs has been proven in the McGuire, Catawba and Sequoyah plants.

The clamp screws, which mount the holddown springs on the top plate of the top nozzle, were also analyzed for normal operation and fatigue loading to determine their structural adequacy. The reactor coolant system design transients used for analyzing fatigue failure were taken from the North Anna UFSAR. Table 3.3-1 summarizes the transients and number of cycles used in the fatigue analysis. The analysis indicated that the clamp screws are structurally sound for all loading conditions.

An evaluation of lift forces on the resident NAIF fuel for a mixed-core configuration was performed to confirm that the design criteria as specified in UFSAR Section 4.4.2.7 continue to be met following introduction of the Advanced Mark-BW fuel. This evaluation assumed a conservative arrangement of one NAIF assembly in a core of Advanced Mark-BW fuel, using forces provided by Framatome. This approach thus incorporates the same hydraulic conditions and conservatisms as the Advanced Mark-BW evaluation. The assessment results confirmed that the NAIF fuel will continue to meet the required criteria.

3.3.1.2 Spacer Grids

The design bases for fuel assembly spacer grids require that no crushing deformations occur for normal operation and Operational Basis Earthquake (OBE) conditions. The grids must also maintain sufficient geometry to ensure control rod insertability for Safe Shutdown Earthquake (SSE) conditions. Grids must provide adequate support to maintain the fuel rods in a coolable configuration under all conditions, including Safe Shutdown Earthquake (SSE) and Loss of Coolant (LOCA) conditions. Any calculated grid deformation must be evaluated to confirm that a coolable geometry can be maintained following a LOCA. Spacer grids were evaluated and found to have positive margin to their elastic limit for all normal operating and OBE conditions.

3.3.1.3 Top and Bottom Nozzles

The top and bottom nozzle design bases follow those outlined in Section III of the ASME Boiler and Pressure Vessel Code.

Finite-element analyses of the top nozzle grillage and the bottom nozzle grillage using ANSYS (Reference 41) were performed to show that the designs are more than adequate to withstand the normal operating loads. The loads used for these analyses were from EOL shutdown condition, since this is the condition at which the holddown force is a maximum. At the operating condition temperature, a conservative scram load was applied to the grillage in addition to the holddown force. For the bottom nozzle, in addition to the scram load applied to the top nozzle, the weight of the fuel assembly was considered when analyzing the structural integrity of the grillage. Results of these analyses met the design basis requirements.

3.3.1.4 Guide Thimble

The design bases for the guide thimble state that no buckling of the thimbles shall occur during normal operation or any transient condition under which control rod insertion is required. In addition, the primary and primary plus secondary stresses shall be lower than the material allowable stresses. Guide thimble buckling was analyzed for normal operating conditions, including mechanical design flow rate, pump overspeed and Rod Cluster Control Assembly (RCCA) scram loading conditions.

The following load cases were analyzed:

1) 100% full power mechanical design flow rate
2) 120% full power mechanical design flow rate (pump overspeed condition)
3) 100% full power mechanical design flow rate with an upper bound scram load of [ ].

Results of these analyses met the design basis requirements.

3.3.1.5 Connections

The ferrule to guide thimble interface was tested to determine the stiffness and strength of the interface. The results of this test, coupled with the results of the guide thimble buckling test were used in the evaluation of the floating intermediate and upper end grid restraint system. The evaluation showed that sufficient margin exists for the ferrule to grid interface under all conditions, both operating and handling. Testing of the connection indicates that the dimple will provide adequate strength under all conditions.

The performance of the guide thimble upper connections, such as the QD sleeve swage and the QD Sleeve-to-End Grid Sleeve interface, and the guide thimble lower connections, such as the End Grid Sleeve-to-Plug crimp and the guide thimble bolt, are ensured through material selection as well as testing and/or analysis of the connection.

3.3.2 Control Rod Drop Times

The design bases for the fuel assembly states that the fuel assembly shall not experience any permanent deformation during either a Condition I or II event that would cause the control component drop time to increase beyond the allowable limits. The maximum allowable control rod drop time specified in the North Anna Technical Specifications is 2.7 seconds, measured from the beginning of decay of stationary gripper coil voltage until the control rod enters the dashpot region of the guide thimbles.

Comparison of the guide thimble diameters for the resident Westinghouse fuel to those of the Framatome Advanced Mark-BW fuel shows that the guide thimble diameter for the resident NAIF fuel design (with either Zircaloy-4 or ZIRLO™ guide thimbles) is smaller. The guide thimble diameter for the Advanced Mark-BW is directly comparable to the Westinghouse LOPAR fuel design. The existing maximum allowable control rod drop time limit is applicable for the Advanced Mark-BW fuel.

3.3.3 Seismic and LOCA Evaluation

A mixed core analysis of the Advanced Mark-BW and resident NAIF fuel was performed to evaluate response to combined seismic and LOCA loads. The acceptance criteria for this analysis are that the fuel assembly shall maintain structural integrity and a coolable core geometry (Reference 6). In the accident analyses, the horizontal effect (LOCA and seismic) and the vertical effect (LOCA) are investigated separately. The analysis involved the development of a horizontal model representing a row of assemblies located on a symmetry axis of the core and a vertical model of the fuel assembly. Only the LOCA effect was analyzed in the vertical direction, as the seismic excitation in this direction will not cause fuel assembly liftoff.

3.3.3.1 Transition Core Horizontal Seismic and LOCA Loads

The SSE and LOCA time history motions of the upper grid plate, lower grid plate and core barrel upper core plate elevation were applied to the reactor core model. The fuel assembly deflection and grid impact force responses were determined by the CASAC computer program, using the general procedure outlined in Reference 5.

The design basis LOCA time histories for core and reactor internals evaluations use "leak-before-break" (LBB) methodology, which has been incorporated into the North Anna licensing basis (UFSAR Section 3.6.2.4). The displacements provided are those associated with a worst case attached pipe break for branch lines attached to the main RCS piping. The cold leg and hot leg data correspond to the accumulator line and pressurizer surge line breaks, respectively. Separate case data was used for Unit 1 (upflow configuration) and Unit 2 (downflow configuration). These displacement data represent the worst case branch line breaks as calculated by Westinghouse for the reactor vessel internals upflow modification.

A mixed core bounding analysis of both Westinghouse NAIF and Advanced Mark-BW was performed under seismic and LOCA events, to demonstrate acceptable performance of both fuel designs. Three possible mixed core configurations were selected to account for the potential core locations where the Advanced Mark-BW may be loaded. These core patterns are shown in Figures 3.3-1, 3.3-2 and 3.3-3. The full core Westinghouse fuel configuration was also analyzed to establish a baseline loading. Dominion provided design input properties that characterized the NAIF fuel assemblies for use in the Framatome analysis. The results of the transition core analyses were compared with the faulted condition analysis results of the Advanced Mark-BW full core configuration.

Fuel assembly models were combined to represent the row configurations in the core. The shortest row in the core has 3 assemblies and the longest has 15 assemblies. Row models with 3 to 15 assemblies were created. The impact forces under a horizontal LOCA loading were calculated for the three mixed core configurations. The maximum grid impact forces for each of the load cases occurred [ ].

The maximum grid impact forces for the SSE conditions occurred [

]. The grid impact forces for the LOCA plus SSE loading condition [

-]

The predicted deformations are evaluated in Section 7.0 to confirm that a coolable geometry is maintained. Results of this evaluation will be reported as part of the LOCA evaluation of Section 7.0 to be included in a subsequent submittal.

3.3.3.2 Vertical LOCA Analysis

A vertical analysis was performed to verify the guide thimble structural integrity in order to allow for control rod insertion during a LOCA. The Reference 6 vertical LOCA method in conjunction with the general-purpose finite-element program ANSYS was used in the analysis.

The guide thimble [ ] is the limiting criterion for the vertical LOCA condition. For conservatism, a load factor of [ ] was used on the guide thimble load to account for unequal loading due to external factors, fabrication differences and inherent design factors. The analysis results confirmed that the forces on the guide thimble were well below conservatively calculated allowable loads. The guide thimble [ ] limit calculated in the analysis is conservative because [

]

The guide thimble stresses resulting from the fuel assembly deflection and axial load were calculated. The results of the analysis also showed that the fuel assembly does not impact the upper core plate during the LOCA. All of the calculated forces are well below conservatively calculated allowable loads for the guide thimbles and fuel rods.

3.4 Fuel Rod Design

A series of analyses have been performed for the Advanced Mark-BW M5™ fuel rod design to confirm its in-reactor mechanical performance. The areas that are analyzed include:

- Cladding Stress
- Cladding Strain
- Cladding Fatigue
- Creep Collapse
- Fuel Rod Growth
- Corrosion
- Shipping and Handling
- Fretting Wear
- Rod Internal Pressure
- Linear Heat Rate to Melt

The calculations support the use of the Advanced Mark-BW M5™ fuel rod assemblies to a peak pin burn-up of [ ] (excepting Rod Internal Pressure and Linear Heat Rate to Melt as the TACO3 code is only licensed to 60,000 MWD/MTU). The 60,000 MWD/MTU limit is equivalent to the peak pin burnup limit for North Anna.

3.4.1 Fuel Rod Cladding Stress

The fuel rod cladding was analyzed for the stresses induced during operation. The ASME pressure vessel stress intensity limits were used as guidelines along with the approved methodology of Reference 3. Conservative values are used for cladding thickness, oxide layer buildup, external pressure, internal fuel rod pressure, differential temperature and unirradiated cladding yield strength. [

]

[

] Based on these results, the Advanced Mark-BW M5™ clad fuel rods will not be adversely impacted by any stresses resulting from operation in the North Anna units.

The limits for the fuel rod stress analysis are based on ASME terminology. Stress level intensities are calculated in accordance with the ASME Code, which includes both normal and shear stress effects. These stress intensities were compared to Sm. Sm is equal to [

]. The limits are as follows:

- Pm < [ ] in compression and < Sm in tension
- Pm + Pb < [ ]
- Pm + Pb + Pl < [ ]
- Pm + Pb + Pl + Q < [ ]

Where the types of stresses analyzed are classified as follows:

1. Pressure Stresses - These are primary membrane stresses (Pm) due to the external and internal pressure on the fuel rod cladding.

2. Flow Induced Vibration - These are longitudinal primary membrane bending stresses (Pb) due to vibration of the fuel rod. The vibration is caused by coolant flow around the fuel rod.

3. Ovality - These are primary membrane bending (Pb) stresses due to external and internal pressure on the fuel rod cladding that is oval. This does not include the stresses resulting from creep ovalization into an axial gap.

4. Thermal Stresses - These are secondary stresses (Q) that arise from the temperature gradient across the fuel rod during reactor operation.

5. Fuel Rod Growth Stresses - These secondary stresses (Q) are due to the fuel rod slipping through the spacer grids. These may be due to the fuel assembly expanding more than the fuel rod due to heat-up, or they may be due to fuel rod growth from irradiation.

6. Three-point Spacer Grid Stop Stresses - These are bending stresses due to the grid stop loads against the fuel rod cladding.

7. Fuel Rod Spacer Grid Interaction - These are localized stresses (Pl) due to contact between the fuel rod cladding and the spacer grid stops.

The analysis results demonstrate that margin exists to the allowable limits for all of the types of stresses analyzed.

3.4.2 Fuel Rod Cladding Strain

The Advanced Mark-BW fuel rod was analyzed to determine the maximum transient the fuel rod cladding could experience before the transient strain limit of 1% is exceeded. Transient strain limit is defined based upon cladding circumferential changes before and after a linear heat rate (LHR) transient. The analysis was conducted using the TACO3 fuel rod thermal analysis code (Reference 40). The M5™ materials have relatively high creep strength compared to Zircaloy-4; the creep rate of the M5™ material is approximately [ ] of the creep rate of Zircaloy-4.

The formula for determining the transient strain is:

$$\varepsilon_{transient} = \frac{(\text{Pellet O.D.})_{transient} - (\text{Pellet O.D.})_{o}}{(\text{Pellet O.D.})_{o}} \text{ X } 100\% \leq 1.00\%$$

The calculated allowable linear heat rates required to satisfy the 1% cladding strain criterion are typically not limiting for plant operation, and are much greater than the maximum transient the fuel rod is expected to experience.

3.4.3 Fuel Rod Fatigue Usage

The fuel rod was analyzed for the total fatigue usage factor using the approved methodology and the procedures outlined in the ASME Code. A maximum fatigue usage factor of 0.9 is allowed. Testing has been conducted by Framatome in France to determine the fatigue performance of M5™ cladding. These tests have shown [

] of Reference 39 is used to determine the fatigue performance of the M5 cladding material. For the fatigue analysis, a fuel rod life of 8 calendar years is used. This fuel rod life bounds the planned exposure of the fuel at North Anna. Based on vessel life of 40 years, the fuel cladding will experience 20% (8/40 years) of the transients the reactor pressure vessel will experience. All possible condition I & II events expected and one condition III event were analyzed to determine the total fatigue usage factor experienced by the fuel rod cladding. Conservative inputs are assumed for cladding thickness, oxide layer buildup, external pressure, internal fuel rod pressure and differential temperature across the cladding.

[

]

3.4.4 Fuel Rod Cladding Creep Collapse

The fuel rods were analyzed for creep collapse using approved methods outlined in Reference 7. The acceptance criterion is that the predicted creep collapse life of the fuel rod must exceed the maximum expected in-core life. The design analysis assumes that the fuel rod will fail due to creep collapse when either of the following happens:

1) [ ] or
2) [
]

The following conservatisms were used in determining creep collapse life of the fuel rod:

[

-]

[

]

The methodology described above was used to simulate the creep collapse lifetime of the Advanced Mark-BW fuel rod. The fuel rod creep collapse lifetime is greater than the design burnup.

3.4.5 Fuel Rod Cladding Corrosion

As discussed in Section 4.0, M5™ fuel rod cladding exhibits a strong resistance to corrosion. From previous irradiation experience with this cladding type, the corrosion has been found to be less than one half the corrosion of low-tin Zircaloy claddings. For the present application, a corrosion prediction based on the present database of M5™ corrosion measurements under the operating conditions at the North Anna reactor shows that an upper limit on cladding corrosion for the M5™ claddings will be [ ] versus an upper limit requirement of 100 μm. The hydrogen pick-up rates of the M5™ cladding have been found to be approximately [ ].

At this corrosion level, the maximum hydrogen content of the M5™ cladding at [ ] is approximately [ ] ppm. The upper limit for hydrogen pick-up is 710 ppm. This level of corrosion and associated hydriding will not adversely affect the structural integrity of the fuel rod during its design lifetime.

3.4.6 Fuel Rod Shipping and Handling Loads

The fuel rod is designed to withstand a [ ]g axial loading during the shipment and the handling of the fuel assembly without gaps forming between pellets in the fuel stack. This design condition is achieved with the usage of a stainless steel spring in the upper plenum of the fuel rod. This spring has an approximate free length of [ ] inches, a wire diameter of [ ] inches, and an outside diameter of [ ] inches. The spring is designed to maintain pre-load on the fuel stack, which prohibits the formation of gaps within the fuel stack. Fuel rod retention within the fuel assembly structure is discussed in Section 3.3.1.2.

3.4.7 Fuel Rod Fretting Wear

The Advanced Mark-BW fuel rod fretting wear performance has been verified based on the proven performance of the standard Mark-BW, the successful 3 cycle operation of the Advanced Mark-BW Lead Test Assemblies in North Unit 1, out-of-core Life and Wear and Flow-Induced Vibration Testing, and analytical benchmarks and evaluations. These efforts were described in Section 2.3.

The Advanced Mark-BW LTA program was a cooperative effort with Dominion that sought to thoroughly test the design prior to batch implementation. Four LTAs were inserted in the core of North Anna Unit 1 in 1997. The LTAs successfully completed three cycles of operation with leak-free performance with a peak pin burnup of ~56,000 MWD/MTU.

Section 2.3 discussed wear and flow induced vibration (FIV) testing results. The wear results are applicable to the M5™ spacer grids given that the grid design and the mechanical properties between the RXA Zircaloy-4 and M5™ materials are almost identical, including grid spring load deflection characteristics, fuel rod contact, elastic modulus and yield strength, which are key parameters for relative wear performance.

3.4.8 Fuel Rod Growth

Growth allowance evaluations were performed for the Advanced Mark-BW fuel assembly. The axial gaps between the top nozzle and reactor upper core plate and between the top nozzle grillage and fuel rods were conservatively analyzed to show that these gaps allow sufficient margin to accommodate the fuel assembly and fuel rod growth to maximize design burnup. The analysis was conducted using the latest irradiation growth models for alloy M5™ guide thimbles and fuel rods based on PIE data for the Framatome fuel designs.

The minimum fuel assembly/reactor core plate gap at end of life for an assumed fuel assembly burnup of [ ] MWD/MTU was determined to be [ ] inch at worst case (cold) conditions. A highly conservative maximum fuel assembly growth was used, particularly considering the low LTA fuel assembly growth. The minimum fuel rod shoulder gap at end of life (EOL) for an assumed rod average burnup of [ ] MWD/MTU was predicted to be [ ] inch at worst-case (hot) conditions using highly conservative methods. For the fuel rod growth evaluations, worst case was considered to be maximum fuel rod growth and minimum (no) guide thimble growth.

3.4.9 Fuel Rod Internal Pressure

The analysis of maximum fuel rod internal pressure was determined using a bounding pin power envelope and axial flux shapes provided by Dominion. The pin power history and axial flux shapes (steady state and transient) were generated using Framatome approved methodologies with Dominion's NRC approved codes. The rod power envelope used bounds the planned cycle designs for the transition and full batch implementation of the Advanced Mark-BW fuel assemblies. The rod powers for Advanced Mark-BW fuel will be evaluated in each fuel cycle to confirm acceptable rod internal pressure performance.

The internal pin pressure predicted with the conservative design pin power envelope was [ ] psia at [ ] MWD/MTU. The fuel rod internal pressure for the Advanced Mark-BW fuel rod design remains below the [ ] psia criterion for operation above system pressure.

3.4.10 Linear Heat Rate to Melt

The Framatome fuel melt limit methodology (Reference 40) has shown that the peak linear power for prevention of centerline melt is 21.9 kW/ft.

Table 3.2-1: Advanced Mark-BW and Resident Westinghouse Fuel Dimension Comparison (in)

Dimension Description	**Advanced Mark-BW**	**W Resident**
Fuel Assembly		
Bottom Nozzle to Top Grid	[ ]	[ ]
Bottom Nozzle to Grid 2	[ ]	[ ]
Bottom Nozzle to Grid 3	[ ]	[ ]
Bottom Nozzle to Grid 4	[ ]	[ ]
Bottom Nozzle to Grid 5	[ ]	[ ]
Bottom Nozzle to Grid 6	[ ]	[ ]
Bottom Nozzle to Grid 7	[ ]	[ ]
Bottom Nozzle to Bottom Grid	[ ]	[ ]
Bottom Nozzle to Spring Clamp	[ ]	[ ]
Top Nozzle		
Top Nozzle Pin Hole Pitch	[ ]	[ ]
Top Nozzle Pin Hole Diameter	[ ]	[ ]
Top of Clamp to Bottom of Enclosure	[ ]	[ ]
Top of Clamp to Top of Grillage	[ ]	[ ]
Top of pad to Align. Pin Relief	[ ]	[ ]
Grillage Thickness	[ ]	[ ]
Bottom Nozzle		
Bottom Nozzle Pin Hole Pitch	[ ]	[ ]
Bottom Nozzle Pin Hole Diameter	[ ]	[ ]
Bottom Nozzle Height	[ ]	[ ]
Grillage Thickness	[ ]	[ ]
Bottom of Leg to Pin Relief	[ ]	[ ]
Bottom of Leg to Bottom of Grillage	[ ]	[ ]

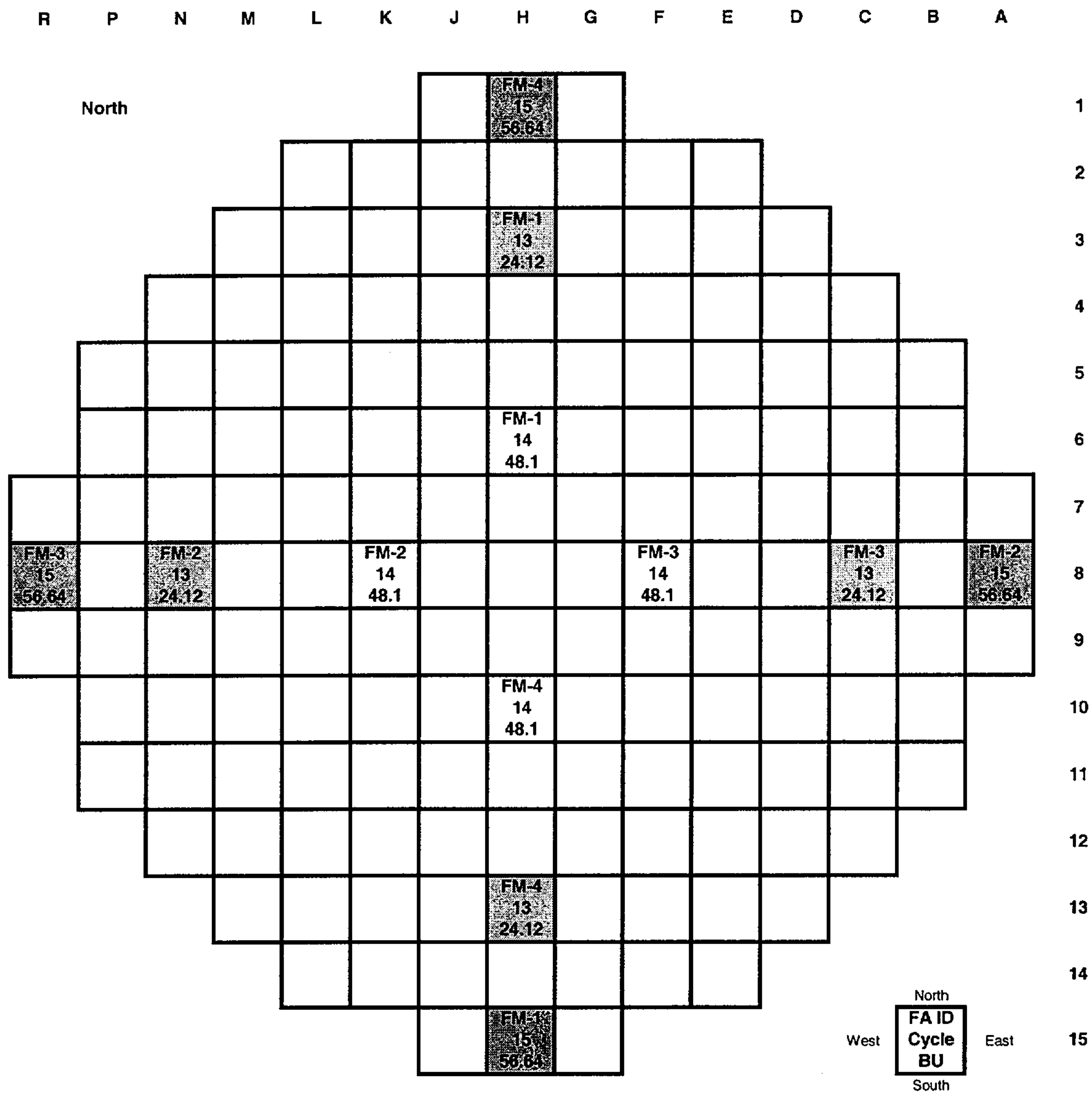
NOTE: (1) Measurements taken from seating surface of bottom nozzle to top of keying window.
(2) Measurements taken from seating surface of bottom nozzle to slot in outer strip.

Table 3.3-1: Summary of Reactor Coolant System Design Transients

Event Description		Anticipated Life-time Occurrences
Normal Conditions		
1.	Heatup and cooldown at 100° F/hr	200 (each)
2.	Unit loading and unloading at 5% of full power/min	18,300 (each)
3.	Step load increase and decrease of 10% of full power	2,000 (each)
4.	Large step load decrease	200
5.	Steady state fluctuations	infinite
Upset Conditions		
1.	Loss of load, without immediate turbine or reactor trip	80
2.	Loss of power (blackout with natural circulation in the reactor coolant system)	40
3.	Loss of flow (partial loss of flow, one pump only)	80
4.	Reactor trip from full power	400
5.	Spray actuation with a differential temperature > 320° F ≤ 560° F	10
6.	Operational Basis Earthquake Reactor Vessel	200 cycles

In accordance with the ASME Nuclear Power Plant Component Code, faulted conditions are not included in the fatigue evaluation.

Figure 3.1-1: North Anna Unit 1 LTA Core Positions*



*Burnups shown in the figure are rod-averaged values for the lead rod. Third cycle values are estimated.

Figure 3.3-1: Core Loading – All Advanced Mark-BW Fuel Configuration Structural Analysis

Figure 3.3-2: Core Loading – Mixed Core Configuration Number 1 - Structural Analysis

Figure 3.3-3: Core Loading – Mixed Core Configuration Number 2 - Structural Analysis

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 exceptions are as follows:

(1) A description of Framatome's mixed-core methodology that was used to demonstrate thermal-hydraulic compatibility of the Advanced Mark-BW fuel assembly with the resident fuel is presented in Appendix A.

(2) An addendum to the BWU-Z CHF topical report justifying the enhanced CHF performance of the MSMGs is under review by the NRC (see Section 4.2.3).

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.

4.1 Design Comparison

The current fuel in North Anna Units 1 and 2 is the North Anna Improved Fuel (NAIF) design, which is a Westinghouse 17x17 VANTAGE-5H design, into which additional debris resistance features and ZIRLO™ fuel rod cladding and skeleton components have subsequently been incorporated. The Framatome Advanced Mark-BW fuel assemblies include a coarse-mesh bottom nozzle for debris resistance, M5™ alloy fuel rod cladding and skeleton, a non-mixing vane grid as the bottommost intermediate grid, and three additional mid-span mixing grids (MSMGs) between the top four intermediate mixing vane grids. The MSMGs, which provide enhanced thermal performance, are not currently utilized on the resident fuel.

4.2 Thermal-Hydraulic Core Models and Inputs

To perform the various thermal-hydraulic analyses needed to license the Advanced Mark-BW design, Framatome uses the LYNXT thermal-hydraulic analysis code. LYNXT, a single-pass code, employs crossflow methodologies to evaluate subchannel thermal-hydraulic conditions for both steady-state and transient conditions. A more complete description of LYNXT is provided in Reference 8.

4.2.1 Summary of Thermal-Hydraulic Analysis Design Parameters

A summary of general core conditions used in the North Anna thermal-hydraulic analyses is provided in Table 4.1-1. The analyses herein are performed at a rated thermal power (RTP) of 2,942.2 MWt which incorporates a measurement uncertainty reduction (MUR) uprate of 1.7% from the current licensed power level of 2,893 MWt.

4.2.2 LYNXT Modeling

Four LYNXT models of the North Anna core were used for the thermal-hydraulic evaluations. An eighth-core 12-channel model was used for most of the full-core DNB analyses and an [ ] model was used for the hydraulic analyses. The details of these two models are discussed in Appendix A (see Figures A-1 and A-4). Both models are essentially the same as those used in licensing calculations for the Sequoyah Nuclear Plant (Reference 9) and for the Framatome LTA project at North Anna (Reference 10).

The 12-channel model was extended for the main steamline break (MSLB) licensing analyses in which additional detail is required. The inlet temperature distribution is not uniform across the core for the MSLB. [ ] In the North Anna MSLB cases, [ ]. Figure A-2 represents this model, including these core location identifiers.

In addition to these models, an [ ] LYNXT model was developed for the thermal hydraulic evaluation of the Advanced Mark-BW and NAIF in mixed-core configurations. The details of this model are discussed in Appendix A (see Figure A-3). A detailed discussion of the mixed-core methodology used for the thermal margin and hydraulic evaluation is presented in Appendix A. The results from application of the mixed-core methodology are discussed in Sections 4.3 and 4.4.

The Advanced Mark-BW is similar in design to the NAIF 17x17 fuel design, with the only significant differences being that the first grid on the Advanced Mark-BW is a non-mixing grid, the Advanced Mark-BW upper guide thimble OD is [ ] inch as opposed to 0.474 inch on the NAIF design and the mid-span mixing grids (MSMG) are incorporated on the Advanced Mark-BW. From a thermal-hydraulic perspective, the two designs are equivalent with the exception of the hydraulic form loss coefficients for the spacer grids and nozzles. Therefore, the thermal-hydraulic core models described in this section are applicable to the NAIF, when the appropriate form loss coefficients are incorporated into the model.

4.2.3 DNB Correlations

Framatome has utilized the results from the approved BWU CHF correlations for the DNB analysis of the Advanced Mark-BW fuel assembly. The BWU family of CHF correlations consists of three correlations that use the same basic equation form but are fit to different data bases. BWU-N is applicable to non-mixing vane grids. BWU-I is the basic mixing vane correlation. BWU-Z is the enhanced mixing vane correlation approved for the Advanced Mark-BW fuel assembly design. The BWU-N and BWU-Z CHF correlations were used as the licensing basis for the Advanced Mark-BW fuel assembly.

The applicable CHF correlation for DNB analysis of the Advanced Mark-BW fuel assembly in the non-mixing (vaneless-grid) region of the fuel assembly is the BWU-N CHF correlation documented in Reference 11. The non-mixing region of the fuel assembly extends from the beginning of the heated length to the leading edge of the first vaned mixing grid.

The applicable CHF correlation for analysis of the Advanced Mark-BW fuel assembly in the mixing region, but below the mid-span mixing grids, is BWU-Z documented in Reference 11. The BWU-Z correlation is used above the mid-span mixing grids with an enhancement factor. The data base for the BWU-Z correlation extends its range of application with improved margin in the annular (middle) and low flow regimes at low pressure, mass velocity and high quality compared to the previous NRC approved CHF correlation BWCMV-A (Reference 12) used for the Mark-BW. Similar to the BWCMV-A correlation, BWU-Z uses a design specific equivalent grid spacing factor.

Improved CHF performance, beyond 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 (Reference 13) to the BWU-Z CHF topical report is under review 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.

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, Reference 15) are used. Framatome justifies the extension of the correlations to the NAIF on the fact that their databases include CHF data representative of the configuration of the Westinghouse NAIF fuel design (Reference 15). Therefore, the correlation applies 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 dimensional velocity profiles downstream of an intermediate spacer grid. The data measurements span over 22 inches of the assembly length representing the grid spacing within a Mark-BW assembly. The results showed a decrease in lateral or crossflow velocities as flow progressed downstream of the grid. The incorporation of mid-span mixing grids in the Advanced Mark-BW design decreases the length between grids and results in an improvement in the span average value of the thermal diffusion coefficient.

4.2.6 Inlet Flow Maldistribution

The thermal-hydraulic analyses impose a five percent reduction in inlet flow to the hot assembly.

4.2.7 Engineering Hot Channel Factors

Engineering hot channel factors are factors that are used to account for the effects of manufacturing variations on the maximum linear heat generation rate and enthalpy rise.

The local heat flux engineering hot channel factor, F_Q^E is used in the evaluation of the maximum linear heat generation rate. This factor is determined by statistically combining manufacturing variances for pellet enrichment and weight and has a value of 1.03 at the 95% probability level with 95% confidence. As discussed in References 14 and 15, relatively small heat flux spikes such as those represented by F_Q^E, have negligible effect on DNB. Therefore this factor is not used in DNBR calculations.

The average pin power factor, $F_{\Delta H}^E$, accounts for the effects of variations in fuel stack weight, enrichment, fuel rod diameter, and pin pitch on hot pin average power. This factor, which has a value of 1.03, is combined statistically with other uncertainties to establish the statistical design limit (SDL) DNBR used with the statistical core design method (discussed in Section 4.4.2).

Since $F_{\Delta H}^E$ is incorporated into the statistical design limit (SDL), this factor is not included in the LYNXT model used for SCD analyses (Section 4.4.2). For non-SCD analyses, $F_{\Delta H}^E$ is incorporated into the LYNXT model as a multiplier on the hot pin average power.

The Framatome methodology for determining the overall peaking factor uncertainty is described in Reference 34. The overall peaking factor uncertainty (F_{QU}) is a statistically combined factor that includes the effect of nuclear calculational uncertainty (F_{NU}), local engineering hot channel factor for fuel (F_{QE-F}), local engineering hot channel factor for lumped burnable poison (F_{QE-P}), rod bow (F_{Q-RB}), and assembly bow (F_{Q-AB}). The overall uncertainty so obtained is less than the current maximum uncertainty factor of 1.0815 (F_{NU} x F_{QE}) used in the Dominion methodology for evaluation of the total peaking factor, F_Q (UFSAR Section 4.3.2.2.6). Dominion will continue to employ the conservative uncertainty factor of 1.0815 (F_{QU}). This value will be used in the calculation of the total peaking factor (F_Q) for centerline fuel melt, transient cladding strain, and LOCA.

4.2.8 Fuel Rod Bowing

As discussed in Section 4.1.1.7 of BAW-10172P (Reference 4), the Mark-BW Zircaloy-4 fuel design has features that make its fuel rod bow performance similar to that of other Framatome fuel designs. In BAW-10186P (Reference 16), Framatome presented new data that extended the rod bow database for Framatome fuel (Zircaloy) to 58,300 MWD/MTU. The topical report concluded that the rod bow correlations from BAW-10147PA-R1 (Reference 17) are applicable at extended burnups and apply to the Mark-BW. [

]

The Advanced Mark-BW fuel contains several advanced design features, including the use of M5™ for the fuel assembly structural tubing and fuel rod cladding. Fuel rod bow is driven by the irradiation growth of the fuel rods and friction with the supporting guide structure. As discussed in Section 3.9 of Reference 3, M5™ has lower growth than Zircaloy-4. Therefore, the performance and penalties established for the Mark-BW fuel design are conservatively applied to the Advanced Mark-BW design.

4.2.9 Active Fuel Stack Height

The active fuel stack height varies during reactor operation due to the combined effects of fuel densification, swelling, and thermal expansion. Densification, which acts to shrink the stack, occurs predominantly at low fuel burnup values, while swelling, which increases stack height, predominates at higher burnups. For high density fuel, such as the Advanced Mark-BW fuel design to be used in the North Anna core, stack shrinkage due to densification is less than the increase caused by thermal expansion of the fuel pellets upon initial heatup. Therefore stack shrinkage is not considered in thermal-hydraulic analysis models. The active fuel height in these models is conservatively assumed to be the nominal initial stack height. This is consistent with the method used to analyze Mark-B fuel that is discussed in Reference 18.

4.2.10 Spike Densification Peaking Factor

The spike densification peaking factor (Reference 33) is used to account for the increased peaking due to inter-pellet gap formation caused by fuel densification as described in North Anna UFSAR Section 4.3.2.2.5. This factor currently is not applied in design analyses (e.g., LOCA linear heat rate limits and centerline fuel melt (CFM) limits) for Advanced Mark-BW fuel. Justification for this treatment is provided below.

Two changes were made in the Mark-BW and Advanced Mark-BW fuel rod designs to mitigate or prevent the formation of large gaps and to prevent cladding creep collapse. The fuel fabrication was modified to produce fuel that undergoes a smaller amount of densification and the fuel rods were pre-pressurized. These changes have reduced gap sizes to an order of magnitude smaller than those observed in the non-pressurized, highly densified fuel rods. In order to determine the effect of inter-pellet gaps on power peaking, Framatome ANP addressed the size and distribution of gaps, and the effect of gaps on power peaking (Reference 34).

The analyses demonstrated that with conservative methods the peaking factor increase due to spike densification for Mark-BW fuel is negligible (Reference 34) and is overly conservative in light of the thermal expansion characteristics of the Mark-BW fuel designs. Two other factors support not including an explicit peaking factor due to inter-pellet gap formation. First, the crucial time for power peaking is early in the life of the fuel rod. The fuel rods measured after a single cycle of irradiation showed no gaps in the eight rods examined. Since no gaps were present, no additional peaking increase occurred in these rods due to axial gaps at the time in life that is a major concern for power peaking. Second, an EPRI report (Reference 35) utilizing axial gap data from three PWR fuel suppliers (including Framatome ANP) obtained similar results and reached a similar conclusion concerning axial gap induced power peaking. Therefore, a penalty for densification to augment calculated power peaking is not applied to the Advanced Mark-BW fuel assembly design.

4.2.11 Core Power Distributions

The reference core axial power distribution for the Advanced Mark-BW fuel is the 1.55 cosine. This power distribution is used for the current North Anna core thermal-hydraulic design analyses.

The nuclear enthalpy rise hot-channel factor, $F_{\Delta H}{}^{N}$, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power. The design value $F_{\Delta H}{}^{N}$ for normal operation is defined in the plant Technical Specifications and is evaluated for its impact on thermal and hydraulic design criteria for each reload core. For power levels below 100%, the radial power distribution is scaled by the part-power multiplier:

$F_{\Delta H}^{N} \leq CFDH * [1 + PFDH(1\text{-}P)]$

where CFDH = the limit at Rated Thermal Power specified in the Core Operating Limits Report,
P = Thermal Power / Rated Thermal Power
PFDH = the Power Factor Multiplier specified in the Core Operating Limits Report

The Power Factor Multiplier is equal to 0.3.

The ultimate goal of the thermal-hydraulic analysis is to support an $F_{\Delta H}^{N}$ limit of 1.587 for reload cores that include the Advanced Mark-BW fuel. For such a limit, the reload core would be designed on a best estimate basis to meet a peak $F_{\Delta H}^{N}$ of 1.526. For statistical analysis, a 4% calculational uncertainty is added to obtain a value of 1.587 for $F_{\Delta H}^{N}$, which is used in the safety analysis, and an additional 4% measurement uncertainty is included in the statistical DNB limit. For non-statistical DNB analysis, the full 8% uncertainty is added to 1.526 to derive a $F_{\Delta H}^{N}$ of 1.65, which is then used in the safety analysis.

The first batch of the Advanced Mark-BW will be implemented assuming the current Technical Specifications $F_{\Delta H}^{N}$ limit of 1.490. Most likely, this limit will be increased in a first step from 1.490 to 1.538, and in a second step from 1.538 to 1.587. However, a one-step change is not precluded at this time.

The analyses for the Advanced Mark-BW are performed to support the ultimate goal for an $F_{\Delta H}^{N}$ of 1.587. As will be discussed in Section 4.4.3, with the exception of the DNB calculations for the loss of flow and locked rotor transient statepoints, the results support this goal. The DNB calculations for the loss of flow and locked rotor transient statepoints support an $F_{\Delta H}^{N}$ limit of 1.538.

The mixed-core effect for the NAIF fuel is calculated in Section 4.4.1 using the current Technical Specification value of 1.490 for $F_{\Delta H}^{N}$.

4.2.12 Core Bypass Flow

The main hydraulic impact of the Advanced Mark-BW is a small increase in the pressure drop across the core. This increase in pressure drop acts to increase the flow through the various flow paths around the core (i.e., core bypass flow paths). The evaluation demonstrated that the component bypass flow through the Advanced Mark-BW guide thimbles was conservatively evaluated assuming a full core of Advanced Mark-BW fuel assemblies at conditions representative of North Anna.

The change in core bypass flow due to the Advanced Mark-BW fuel assembly design was assessed. A bounding nominal (i.e., no uncertainties) bypass flow was calculated to be less than 5.0% for a core loading pattern with no core inserts (e.g., discrete poison assemblies, secondary sources) located in the thimble guide tubes. The purpose for this calculation is to cover the potential use of an integral poison product. The minimum bypass flow was calculated to be greater than 3.0% for a full core of Advanced Mark-BW fuel with a core loading pattern with 1500 core inserts (e.g., discrete poison assemblies, secondary sources) located in the thimble guide tubes.

A bypass flow of 5.5% is specified for use in the statistical DNB analysis for the Advanced Mark-BW. This value represents a bounding value for the nominal bypass flow for a core configuration with all Advanced Mark-BW fuel assemblies and no core inserts. A design core bypass flow of 6.5% is specified for non-statistical DNB applications and other deterministic NSSS evaluations. A minimum bypass flow of 3.0% is specified for the lift force calculations.

4.2.13 Reactor Coolant Flow Rate

Assessments of core pressure drop are performed using the LYNXT thermal-hydraulics code (Section 4.3.1). Due to grid design differences and the presence of the MSMGs, the pressure drop is higher for a Framatome Advanced Mark-BW fuel assembly than for the resident Westinghouse fuel design (NAIF). It is estimated that there will be a small decrease in RCS flow in the first batch of Framatome fuel and a slightly greater decrease in RCS flow once a full core of Framatome fuel is achieved. Due to the current large RCS flow margins in North Anna, this effect represents a small but acceptable impact on the RCS flow rate.

4.3 Hydraulic Compatibility

The process of evaluating the pressure drop, hydraulic loads, and cross flow velocities in mixed-core configurations uses the Framatome mixed-core methodology outlined in Appendix A. The calculations are performed using the LYNXT computer code and the [ ] LYNXT model.

The mixed-core analyses for pressure drop, hydraulic load, and cross flow velocities consider four configurations:

(1) Full-core of Advanced Mark-BW
(2) Full-core of resident NAIF
(3) Limiting configuration for Advanced Mark-BW
(4) Limiting configuration for resident NAIF

The full-core configurations define the baseline characteristics of each assembly used in the quantification of the penalty/benefit incurred for mixed-core configurations. In DNBR evaluations, the limiting configuration is one that minimizes the flow into the limiting fuel assembly. In pressure drop, hydraulic loads, and crossflow velocity evaluations, the limiting configuration for each assembly type is the configuration that maximizes the flow in the limiting fuel assembly. Hence, the limiting configuration for the lower-pressure drop fuel assembly type (NAIF) is a single lower-pressure drop assembly (NAIF) in the center location of the core, with the higher-pressure drop assemblies (Advanced Mark-BW) comprising the rest of the core. The limiting configuration for the higher-pressure drop fuel assembly type (Advanced Mark-BW) is a full-core configuration of the high pressure drop assemblies (Advanced Mark-BW).

The core operational conditions for the analyses consist of a cold (low inlet temperature) zero power, hot (near nominal inlet temperature) zero power, hot full-power, hot overpower, normal flow, mechanical design flow, and high flow (pump overspeed). Pressure drop cases are performed at nominal conditions to define the baseline operating characteristics of the assembly type. The hydraulic loads are generated across a wide range of operating conditions since the mechanical behavior of the hold down springs is dependent on the operating condition of the core, particularly the temperature. Cross flow velocities are calculated during the evaluation of the nominal pressure drop and the hydraulic loads and thus cover the same operational conditions.

4.3.1 Mixed-Core Nominal Pressure Drop Results

Nominal pressure drop evaluations are determined using the [ ] LYNXT model. Figure 4-1 shows the axial pressure drop profiles for each of the core configurations. The pressure drops are for the fuel assembly only and do not include the effects of the upper and lower core support plates. As expected, the maximum pressure drop for any Advanced Mark-BW is calculated for a full-core configuration and the maximum pressure drop for any NAIF is calculated for a mixed-core with one NAIF in the center location and the remainder of the core being Advanced Mark-BW assemblies.

4.3.2 Mixed-Core Hydraulic Load Results

The hydraulic load evaluations are determined using the [ ]. The hydraulic loads do not include the fuel assembly weight, buoyancy forces, or spring hold down forces. The maximum hydraulic load for any Advanced Mark-BW is calculated for a full-core configuration and the maximum hydraulic load for any NAIF is calculated for a mixed core configuration with one NAIF in the center location and the remainder of the core being Advanced Mark-BW assemblies. The results of these analyses are used in the evaluation of the fuel assembly holddown springs in Section 3.3.1.1.

4.3.3 Mixed-Core Cross Flow Velocities

A design requirement of the Advanced Mark-BW fuel assemblies is that they shall not cause inter-assembly cross flow velocities to [ ] for a mixed-core configuration. Cross flow velocities are checked for compliance in pressure drop and hydraulic load calculations. The maximum 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. 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 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). The exceptions to this rule were the nominal, rod urgent fail, and main steamline break statepoints which were evaluated with $F_{\Delta H}{}^{N}$ values that were equal to a bounding reload set of values. These are non-limiting conditions and it is sufficient to demonstrate acceptable performance with this bounding set. In addition, the loss of flow and locked rotor statepoints were evaluated with $F_{\Delta H}{}^{N}$ values of 1.538 and 1.587.

In general, the statepoint conditions which were defined to evaluate the Advanced Mark-BW were used to evaluate the NAIF in the transition-core analysis, except that the maximum radial peaking is based on a full-power $F_{\Delta H}{}^{N}$ of 1.490 (1.55 with uncertainty). This is the current licensing basis for the NAIF. The radial peaking factors for the nominal, the rod urgent failure, and the MSLB statepoints were identical to those specified for the Advanced Mark-BW statepoints.

4.4.2 Statistical Core Design (SCD)

The LYNXT computer code was employed with Framatome's Statistical Core Design (SCD) technique to assess thermal margin for the Advanced Mark-BW fuel design. The SCD method is not specific to a fuel design and applies to any Framatome analysis code with any CHF correlation. The NRC has approved Framatome's SCD methodology for licensing of Framatome fuel in Westinghouse-designed reactors (Reference 19).

The Framatome SCD approach uses a statistical combination of uncertainties technique. In the SCD method, described in Reference 19, the uncertainties on a group of input variables are subjected to a statistical analysis and an overall DNBR uncertainty is established. This uncertainty is then used to establish a DNBR design limit known as the Statistical Design Limit (SDL). All variables treated in the development of the SDL are then input to the thermal-hydraulic analysis computer codes at their nominal level for subsequent analyses for which the SCD is applicable. For added flexibility, margin is added to the SDL. This added margin defines an analysis limit termed the Thermal Design Limit (TDL). Once the TDL has been established, the calculated DNBR at a specific core state is compared to the TDL to determine if the DNB protection criterion is met.

For the planned insertion of the Advanced Mark-BW fuel assemblies into North Anna, the plant specific variables listed in Table 4.1 were used to determine Statistical Design Limits for the BWU-N and BWU-Z CHF correlations. The ranges and uncertainties of these variables are consistent with those used for the implementation of the Virginia Electric and Power Company Statistical DNBR Evaluation Methodology (Reference 20) for North Anna (Reference 21). Hot pin and core-wide SDL values were calculated for the BWU-N and BWU-Z CHF correlations at the statepoints defined in Section 4.4.1.

The resulting SDL values are 1.61 for BWU-N correlation and 1.31 for the BWU-Z correlation. A common TDL of 1.70 has been defined for application of both correlations for the initial implementation of the Advanced Mark-BW fuel. The retained thermal margin made available by using this TDL is calculated using the following formula.

$$\text{Retained Thermal Margin (\%)} = \frac{\text{TDL - SDL}}{\text{TDL}} \times 100$$

The resulting retained thermal margin values are 5.3% for the BWU-N correlation and 22.8% for the BWU-Z correlation. This retained thermal margin provides flexibility in the fuel cycle design. Examples of tradeoffs that might be assessed against the retained margin include mixed-core effects and penalties for input uncertainties greater than those considered in the SDL development.

4.4.3 Full-Core SCD DNB Analysis for Advanced Mark-BW

Full-core SCD DNB calculations were performed for the applicable statepoints defined in Section 4.4.1 using the LYNXT computer code and the 12-channel LYNXT model. The results of the calculations demonstrate that the minimum DNBR values are equal to or greater than a TDL of 1.70 (Section 4.4.2) for the safety limit lines, limiting axial flux shapes, misaligned rod, rod withdrawal at power, and rod urgent failure all statepoints with an $F_{\Delta H}^{N}$ of 1.587 and for the loss of flow and locked rotor with an $F_{\Delta H}^{N}$ of 1.538. For the loss of flow and locked rotor with an $F_{\Delta H}^{N}$ of 1.587, the minimum DNBR values exceed 1.610 and would be supported by a reduced TDL associated with a second batch application in which the mixed-core effect was reduced.

4.4.4 Mixed-Core DNB Analysis

The process of evaluating DNB in mixed-core configurations uses the Framatome mixed-core methodology outlined in Appendix A. The DNB calculations are performed using the LYNXT computer code and the [ ] LYNXT model. The DNBR results from mixed-core configurations are compared to calculations performed at identical conditions with either a full-core Advanced Mark-BW model or a full-core NAIF model. The mixed-core penalty is equal to the largest differential value. That is,

$$\text{Bounding Penalty} = \frac{\text{Full Core DNBR - Mixed Core DNBR}}{\text{Full Core DNBR}} \times 100\%$$

Both the NAIF and the Advanced Mark-BW assemblies are examined for potential mixed-core penalties.

The various patterns used to determine the mixed-core penalty for the Advanced

Mark-BW are shown schematically in Figure 4-2. The patterns consist of a range of configurations: [

]

[

]

The core configuration used to calculate the relative thermal performance of the NAIF consists of a single Westinghouse assembly in a core of Advanced Mark-BW. The single NAIF is located in the core center. This is the most conservative mixed-core configuration for NAIF. Only this core configuration is considered for NAIF since the mixed-core penalty for the NAIF is small.

The results from the calculations indicate that mixed-core penalties of [ ] and [ ] be applied to the Advanced Mark-BW when it is being inserted into an NAIF core for the first and second transition core applications, respectively. The mixed-core penalty applied to the NAIF fuel for DNB is [ ].

4.4.5 Deterministic DNBR Calculations

For the deterministic analyses, uncertainties on the statepoint conditions and model inputs are treated explicitly in the LYNXT input as opposed to incorporating them into the SDL for SCD-based applications. The deterministic MDNBRs are compared to the applicable CHF correlation design limit and not the TDL as in the case with SCD. Statepoints are analyzed deterministically if the conditions fall outside the ranges of the SCD uncertainty propagation RSM or if the conditions fall outside the ranges of the applicable CHF correlation.

The three deterministic cases for North Anna Units 1 and 2 are as follows:

1. Rod Withdrawal From Subcritical (RSWC)
2. Main Steam Line Break/High Flow (MSLB/HF)

3. Main Steam Line Break/Low Flow (MSLB/LF)

The DNBR analyses for the RSWC are based on the 12-channel model in Figure A-1. The two MSLB DNBR analyses are based on a [ ] model, [ ] (core location identifiers, see Figure A-1) [ ] is shown in Figure A-2.

DNBR results were calculated for the full-core Advanced Mark-BW, Advanced Mark-BW PLB in a NAIF core, full-core NAIF, and NAIF PLB in an Advanced Mark-BW core. None of the other mixed-core configurations in Figure 6-2 were evaluated since these four configurations were most limiting and these events were non-limiting DNBR transients. The BWU-N/BWU-I CHF correlations were used for DNBR analyses of the NAIF PLB since the MSLB conditions on pressure and mass flux are well below the W-3 CHF correlation limit. [

]

4.5 Overpower ΔT (OPΔT) and Overtemperature ΔT (OTΔT) Reactor Trip Functions

The thermal overpower ΔT and overtemperature ΔT trip functions are discussed in the North Anna UFSAR and Technical Specifications. Westinghouse in Reference 22 describes the analytical methods used to derive limiting safety-system settings for these trip functions. In Westinghouse-designed reactors, these trip functions are designed to provide protection against fuel centerline melting (by limiting the linear heat generation rate) and departure from nucleate boiling (DNB) and to assure that the vessel temperature rise (ΔT) is proportional to core power during postulated transients (Condition II events). Included in each trip is an $f(\Delta I)$ reset function, which imposes a penalty on the allowable temperature rise in the event of adverse axial power shapes that may occur during a transient.

The bases for these trip functions are the core thermal limit lines (i.e., reactor safety limit lines), axial offset envelopes, and other reactor coolant system and plant parameters. The core thermal limit lines are defined as the most limiting of vessel exit boiling, hot channel exit quality, and the core DNB considerations. The axial offset envelopes are a parameterization of DNB performance versus axial flux difference (ΔI). The statepoint conditions from Section 4.4.1 represent the DNB portion of the currently licensed core thermal limit lines modified for the 1.7% MUR uprate (Section 4.2.1) and a bypass flow of 5.5% (Section 4.2.9). Both these statepoint conditions and the 1.7% MUR uprate represent a change from the current plant parameters. Therefore, the thermal overpower ΔT and overtemperature ΔT trip functions were reevaluated to determine whether any changes to their setpoint were required for the implementation of the Advanced Mark-BW at North Anna.

The thermal overpower ΔT and overtemperature ΔT trip functions were evaluated using the approved Westinghouse methodology described in Reference 22. The evaluation demonstrated that the currently licensed core thermal limit lines remain bounding and the K_1, K_2, K_3, K_4, and K_6 constants for the thermal overpower ΔT and overtemperature ΔT trip functions continue to provide bounding protection for the implementation of the Advanced Mark-BW at North Anna.

The $f(\Delta I)$ reset function for the thermal overtemperature ΔT trip function requires two modifications. The first modification consists of a change in the value at which the negative end of the deadband begins. This value is changed from –44% to –35% axial flux difference (i.e., ΔI). The corresponding value for safety analyses is changed from –47% to –38% axial flux difference. No changes were required to the positive end of the deadband or to the negative and positive runback ramp-rates.

The second modification to the $f(\Delta I)$ reset function alters the maximum allowed penalty value obtained from the $f(\Delta I)$ reset function for highly top-skewed power distributions (positive ΔI). This change extends the range of the $f(\Delta I)$ reset function generator to accommodate axial flux differences between –50% and +50% (versus –50% and +28% currently). This change is necessary to reflect the Reference 22 power distribution validation methodology as applied to the Advanced Mark-BW in North Anna cores without benefit of an accumulated history of reload power distribution thermal performance data. It is expected that such reload power data would confirm the non-limiting nature of the highly top-skewed power distributions. Similar sensitivity studies have shown that these highly top-skewed power distributions are significantly non-limiting for Westinghouse fuel.

These changes will be reflected in the COLR and Precautions, Limitations and Setpoints (PLS) document at North Anna. With the planned approval of the Improved Technical Specifications at North Anna, the cycle specific COLR for each unit will be modified to include the change in the value at which the negative end of the deadband begins. The Precautions, Limitations and Setpoints document will be modified to reflect both the change in the value at which the negative end of the deadband begins and the change to extend the range of the $f(\Delta I)$ reset function generator to accommodate axial flux differences between -50% and +50%.

Table 4.1-1: Thermal-Hydraulic Analysis Design Parameters

Design Parameter	Value
Reactor Coolant System:	
Rated Thermal Power, MWt	2942.2 [(a)]
Heat Generated In Fuel, %	97.4
Nominal System Pressure, psia	2250
Minimum DNBR at nominal conditions	3.0922 [(b)]
Minimum DNBR for design transients	1.70
DNB Correlation	BWU
Core Configuration:	
Number of Fuel Assemblies	157
Fuel Assembly Type	17x17
Number of Fuel Rods Per Assembly	264
Number of Control Clusters	48
Number of Absorber Rods per Control Cluster	24
Stack Height, in	144
Fuel Rod Outer Diameter	0.374
Assembly Flow Area, sq-in	38.7
Coolant Flow	
Minimum Measured Flow, gpm	295,000 [(c)]
Flow Fraction Effective for Heat Transfer	0.945 [(c)]
Lower Bounding Design Flow, gpm	289,100 [(d)]
Mechanical Design Flow, gpm	315,600
Hot Channel Core Inlet Flow Factor	0.95
Core Inlet Velocity, ft/sec	14.71
Inlet Mass Flux, Mlb/hr-sq-ft	2.47
Vessel Mass Flow Rate, Mlb/hr	110.38
Core Pressure Drop (nozzle to nozzle), psi	27.5
Coolant Temperatures (nominal at 100%RTP)	
Nominal inlet, °F	553.7
Average rise in vessel, °F	66.2
Average rise in core, °F	65.9
Average in vessel, °F	586.8
Average in core, °F	588.5
Vessel outlet, °F	619.9
Core outlet, °F	623.3

Table 4.1-1 (continued): Thermal-Hydraulic Analysis Design Parameters

<u>Heat Transfer</u>	
Average heat flux, Btu/hr-sq-ft^2	206,200
Average thermal output, kW/ft	5.76
Maximum thermal output for normal operation, kW/ft	12.62[e]
Maximum thermal output at maximum overpower, kW/ft trip point (118% power)	21.9

a. These parameters are for a core rated thermal power of 2942.2 MWt, which bounds the current licensed maximum core power of 2893 MWt.
b. DNB calculation based on full-power $F_{\Delta H}^{N}$ value of 1.49 (corresponds to 1.55 design value with 4% measurement uncertainty).
c. Value used in DNB analyses for statistical DNB events.
d. Value used in DNB analyses for non-statistical DNB events.
e. This parameter value is associated with an F_Q value of 2.19.

Table 4.2-1: Statistical Core Design Application Summary

North Anna Plant Specific Uncertainties

Plant Uncertainties

Variable	Uncertainty	Distribution
Core Power	$\pm$2.2% at 2σ	Normal
Core Flow	$\pm$2.0% at 2σ	Uniform
Core Pressure	$\pm$36 psi at 2σ	Uniform
Core Inlet Temperature	$\pm$3.7°F at 2σ	Uniform
Measured $F_{\Delta H}^{N}$	$\pm$4.0% at 2σ	Normal
Core Bypass Flow	$\pm$1.0% at 2σ	Uniform

Framatome Analysis Uncertainties

Variable	Uncertainty	Distribution
Hot Channel Factor	[	]
Bundle Spacing	[	]
Axial Peaking Factor	[	]
Axial Peak Location	[	]
DNB Correlation	[	]
	[	]
LYNXT Code Uncertainty	[	]
Response Surface Model	[	]
(RSM) to LYNXT Fit	[	]

Figure 4-1 - Axial Pressure Drop Profiles [for Various Core Configurations].

Figure 4-2 – [Various One-Eighth Core Symmetric Loading Patterns for North Anna Units 1 and 2.]

5.0 Neutronic Performance

Consistent with References 27 and 28, a nuclear design evaluation will be performed for each North Anna cycle to demonstrate that reload cores will meet all applicable design criteria. This evaluation will be performed under the normal reload design process and schedule, and will be documented in the cycle specific Reload Safety Evaluation.

The physical differences between the Advanced Mark-BW fuel assembly and the resident Westinghouse fuel are small. Cycle specific neutronic calculations will account for the effects of the M5™ material used for fuel rod cladding, guide thimbles, instrument tubes and grids. These effects have been successfully modeled for the Framatome LTAs that have experienced three cycles of irradiation in North Anna Unit 1. The Dominion nuclear core design tools can model the Advanced Mark-BW fuel to the same accuracy as the current Westinghouse fuel. This demonstration was accomplished by comparing the predicted and measured power distributions (axial and radial) for the specific core location containing the Framatome LTAs.

The minor changes in the Advanced Mark-BW fuel features that would affect key safety analysis parameters are within the modeling capability of Dominion safety and core design analysis codes. Such minor changes include: small change in nominal fuel density, use of M5™ cladding (versus ZIRLO™), and the inclusion of Mid-Span Mixing Grids (MSMGs) in the Advanced Mark-BW fuel. These changes are within the scope of similar plant and fuel design changes that Dominion has successfully analyzed and implemented during operation of the North Anna and Surry plants.

As a result of the general physical similarity to the resident Westinghouse fuel designs, the Advanced Mark-BW fuel has essentially the same neutronic behavior as the resident fuel assemblies. On an equal enrichment basis, the lead test assemblies irradiated in Unit 1 initially exhibited reactivity similar to the resident Westinghouse fuel. Due to the higher uranium loading (primarily the result of a higher nominal fuel density), the rate of reactivity depletion would be slightly smaller for the Advanced Mark-BW design than for the majority of the fuel in the North Anna core. This difference is explicitly modeled in the cycle specific neutronic calculations and does not adversely impact plant operation.

Changes to the neutronic model inputs necessary to model the physical differences between the Advanced Mark-BW assemblies and the resident Westinghouse fuel assemblies are similar to those used for previous Westinghouse fuel product changes, and are of a smaller magnitude than was necessary for many of the Westinghouse fuel product changes. The core reactivity coefficients and nuclear performance for the three North Anna cores containing the Framatome LTAs were not noticeably different from recent reload cores consisting of all Westinghouse fuel, confirming the applicability of Dominion's standard reload core design models and methods to cores containing Framatome assemblies.

6.0 Non-LOCA Safety Evaluations

The performance of the Advanced Mark-BW fuel product under postulated non-LOCA accident conditions was evaluated by Dominion. For the implementation of the Framatome Advanced Mark-BW fuel product at North Anna, all accident analyses were reviewed for potential impact upon the NSSS predictions. The Advanced Mark-BW fuel assembly has a slightly larger pressure drop than NAIF fuel, due to the use of mid-span mixing grids. The increase in pressure drop has a small impact on RCS flow reduction transients such as the loss of reactor coolant flow and the reactor coolant pump locked rotor/sheared shaft. In addition to the larger pressure drop, the Advanced Mark-BW has slightly different fuel thermal properties for safety analysis design inputs than the Westinghouse NAIF. These changes and the incorporation of M5™ cladding properties impact the rod heatup calculations for the control rod ejection transient. Therefore, the non-LOCA transients selected for reanalysis were the loss of reactor coolant flow, the reactor coolant pump locked rotor/sheared shaft, and the control rod ejection.

The implementation of the Framatome Advanced Mark-BW also has an impact on the calculated DNBR results for the Chapter 15 accident analyses. Each Chapter 15 accident was reviewed and the statepoints were selected for DNB analysis. The DNB analyses for the loss of flow and locked rotor are discussed with the results of their NSSS reanalyses in Sections 6.3.1 and 6.3.2, respectively. The remaining DNB evaluations are discussed in Section 6.2.2.

The accident analyses and DNBR calculations for the implementation of the Framatome ANP Advanced Mark-BW fuel product consider the effects of a small power uprate based on reduction in power calorimetric uncertainty. This approach was taken to accommodate future implementation of a design change referred to as a measurement uncertainty reduction (MUR) uprate. Existing transient analyses accommodate a core power calorimetric uncertainty of 2%. A reduced uncertainty can be achieved by installation of more accurate feedwater flow and feedwater temperature instrumentation. This reduction in uncertainty can then be used to uprate the rated thermal power (RTP) of the core. The accident and DNBR calculations discussed herein consider a 1.7% power uprate. The DNB analyses explicitly include a 1.7% power uprate by assuming a initial core power level of 2942.2 MWt (2893 MWt x 1.017). The RCS and MSS overpressurization analyses include a 2% allowance for core power calorimetric uncertainty with an initial core power of 2893 MWt. This is equivalent to the application of a 0.3% allowance with a core power 2942.2 MWt. The results from this method of analysis can be used to support a MUR uprate of up to 1.7%.

6.1 Assessment of Impact Upon NSSS Modeling Design Inputs

The system response to transient events is analyzed with the RETRAN system code (Reference 36). Separate RETRAN models are used to represent both a full core of NAIF and a full core of Advanced Mark-BW. The following issues were evaluated for inclusion as changes to the model: plant conditions, trip reactivity - control rod drop time, core stored energy, and clad and fuel thermal properties.

6.1.1 Nominal Plant Conditions

The nominal plant conditions for the Framatome Advanced Mark-BW implementation are as follows: RCS flowrate of 295,000 gpm, bypass flow of 5.5%, and a rated thermal power (RTP) of 2942.2 MWt. The rated core power includes an increase of 1.7% above the current Technical Specification RTP of 2893 MWt. These conditions are utilized for statistical DNB evaluations.

For non-statistical DNB applications and other deterministic NSSS evaluations, the design plant conditions for the Framatome Advanced Mark-BW implementation are as follows: RCS flowrate of 289,100 gpm, bypass flow of 6.5%, and a rated thermal power (RTP) of 2893 MWt with a 2% allowance for core power calorimetric uncertainty. As stated in Section 6.0, this is equivalent to the application of a 0.3% allowance with a core power 2942.2 MWt. The results from this method of analysis can be used to support a MUR uprate of up to 1.7%.

6.1.2 Trip Reactivity - Control Rod Drop Time

The negative reactivity insertion following a reactor trip is a function of the acceleration of the rod cluster control assemblies and the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry, approximately 85% of the rod cluster travel. The maximum allowable control rod drop time specified in the North Anna Technical Specifications is 2.7 seconds, measured from the beginning of decay of stationary gripper coil voltage until the control rod enters the dashpot region of the guide thimbles.

Comparison of the guide thimble diameters for the resident Westinghouse fuel to those of the Framatome Advanced Mark-BW fuel shows that the guide thimble diameter for the resident NAIF fuel design (with either Zircaloy-4 or ZIRLO™ guide thimbles) is smaller. The guide thimble diameter for the Advanced Mark-BW is directly comparable to the Westinghouse LOPAR fuel design.

In addition drop time data from the Framatome LTA program has been evaluated. RCCA drop times were typical of Framatome 17x17 fuel assembly designs and were well within the acceptance criterion. The increase in the core pressure drop from the introduction of the Advanced Mark-BW fuel will produce a slight but negligible increase in the measured control rod drop times. The RETRAN model includes a bounding time for control rod insertion. No changes are required to the RETRAN model to accommodate this effect. Therefore, transients which might be affected by changes in trip reactivity are not adversely impacted by the change to Advanced Mark-BW fuel.

6.1.3 Clad and Fuel Thermal Properties

Changes to the clad and fuel thermal conductivity, the heat capacity and the initial fuel temperature have been incorporated into the RETRAN model for the Advanced Mark-BW fuel. The changes to these parameters affect the core stored energy (Section 6.1.4) and the rod ejection accident. The rod ejection accident has been reanalyzed to account for variations that changes in the fuel or clad thermal properties may have on the results of this event (Section 6.3.4). The modeling of the very rapid power increase associated with this transient and the use of a hot spot model makes this transient particularly sensitive to these parameters.

6.1.4 Core Stored Energy

The stored energy in the core at the beginning of a system transient analysis impacts the transient results. Models used with transient analysis codes such as RETRAN use the average core fuel temperature as an initial condition. Considering the number of fuel pins in the reactor core, it is appropriate to use a nominal fuel average temperature in the calculation of initial core average temperature for quantifying the initial core heat content.

With the physical dimensions of the fuel from both vendors being so similar, the amount of initial energy stored in the core is then a function of the fuel average temperature and the fuel heat capacity. A comparison of core stored energy between the Westinghouse and Framatome fuel products concluded that the change in stored energy in the fuel between hot zero power conditions and hot full power conditions is slightly less for the Framatome fuel than that assumed in the current model for Westinghouse fuel. Therefore, transients which might be affected by changes in core stored energy, such as the loss of normal feedwater and loss of offsite power events, are not adversely impacted by the change from the Westinghouse fuel to the Framatome fuel.

These key design inputs for the Advanced Mark-BW fuel were reviewed for potential differences and impact upon the long-term containment integrity analysis: changes in the flow characteristics past the fuel, core stored energy and the decay heat. The localized flow diversion (for mixed-cores) or greater total core pressure drop (mixed and full-Advanced Mark-BW cores) will have an insignificant effect upon the mass and energy releases. Since the core stored energy is slightly reduced, the existing analysis values will remain bounding. The decay heat will be a function of the total core power, which is assumed to be 102% of 2893 MWt in the existing analysis. The existing mass and energy and containment integrity analyses will remain bounding for operation with the Advanced Mark-BW fuel.

6.2 Assessment for Accident Events Not Reanalyzed

6.2.1 NSSS Transient Effects

The assessment of Section 6.1 indicates that most key design inputs that influence NSSS transient behavior either remain bounding or are unaffected by the introduction of the Advanced Mark-BW fuel. It is concluded from that assessment that the accident analyses for each of the events listed below do not require reanalysis.

- Uncontrolled RCCA Withdrawal from Subcritical
- Uncontrolled RCCA Withdrawal at Power
- RCCA Misalignment
- Uncontrolled Boron Dilution
- Startup of an Inactive RC Loop
- Loss of External Electrical Load and/or Turbine Trip
- Loss of Normal Feedwater
- Loss of Offsite Power to the Station Auxiliaries
- Excessive Heat Removal Due to Feedwater System Malfunctions
- Excessive Load Increase Incident
- Accidental Depressurization of the RCS
- Accidental Depressurization of the Main Steam System and Steamline Ruptures
- Spurious Operation of the Safety Injection system at Power
- Major Rupture of a Main Feedwater Line
- Containment Pressure Analysis (Steamline Breaks and LOCAs)

The current analyses of record for these accidents remain bounding for cores with the current Westinghouse fuel Product, the Framatome fuel product and the transition cores with a mixture of the two fuel products.

6.2.2 Core Thermal (DNB) Effects

As stated in Section 6.0, the implementation of the Framatome Advanced Mark-BW also has an impact on the calculated DNBR results for the Chapter 15 accident analyses.

Even though the NSSS transient simulation for many events is unaffected by introducing the Advanced Mark-BW fuel, the detailed core thermal behavior for a number of such events was investigated. This was accomplished by performing detailed thermal analysis (DNB calculations) at specifically defined 'statepoints' representative of the limiting conditions that may occur during these events. The following procedure was used to obtain the necessary statepoints for the DNB calculations. Statepoints were considered from the UFSAR Chapter 15 events including the core thermal limits (CTLs), axial offset envelopes (AOs), rod withdrawal from power (RWAP), rod withdrawal from subcritical (RWSC), control rod misalignment, MSLB, LOFA, and LOCROT events. These various limits and events provide sensitivity of DNB performance to the following: (a) power level (including the impact of the part-power multiplier on the allowable hot rod power, $F_{\Delta H}^{N}$), pressure and temperature (CTLs); (b) axial power shapes (AOs); (c) elevated hot rod power (misaligned rod); (d) low flow (LOFA and LOCROT), and (e) non-statistical DNB events (RWSC and MSLB). The limiting statepoint information was then adjusted to be consistent with the design conditions selected for the Advanced Mark-BW implementation (Section 6.1).

This information was then provided to Framatome along with the set of more than 150 statepoint conditions that was defined by Dominion for use in the full-core and transition-core analysis to demonstrate acceptable DNB performance (Section 4.4.1). The DNB analyses for the loss of flow and locked rotor are discussed with the results of their NSSS reanalyses in Sections 6.3.2 and 6.3.3, respectively. The remaining evaluations are discussed below.

DNB analyses were conducted for this set of statepoint conditions using the analytical tools discussed in Section 4. The LYNXT computer code was used with the BWU DNB correlations to calculate a minimum DNBR for the Advanced Mark-BW fuel at each of the statepoint conditions. The minimum DNBR was demonstrated for all cases to be greater than the applicable DNBR design limit for each statepoint (Section 4).

6.3 Accidents Reanalyzed

6.3.1 Computer Codes and Models

The transients are analyzed using three computer codes. The system response to the three-pump flow coastdown is analyzed with the RETRAN system code and models (Reference 36). Separate RETRAN models are developed to represent both a full core of NAIF and a full core of Advanced Mark-BW. The RETRAN calculations determine primary systems parameters such as the loop and core flows, the time of reactor trip, the nuclear power transient, and the primary coolant system temperatures and pressures during the event. For the DNB events, these data are then used in a detailed thermal-hydraulic computation to compute the minimum DNBR values. The COBRA computer code is used with the WRB-1 DNB correlation to calculate a minimum transient DNBR for the resident NAIF fuel. The LYNXT computer code is used with the BWU DNB correlations to calculate a minimum DNBR for the Advanced Mark-BW fuel at the limiting statepoint condition.

6.3.2 Loss of Reactor Coolant Flow

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly. The description of this event and details of the analysis are unaltered from that presented in UFSAR Section 15.3.4, except as noted below. Certain other description is presented here for completeness. The following reactor trips provide necessary protection against a loss-of-coolant-flow accident:

1. Undervoltage or underfrequency on reactor coolant pump power supply buses.
2. Loss of reactor coolant loop flow.
3. Pump circuit breaker opening.

For loss of flow events initiated by a loss of voltage to the RCP bus, reactor protection is provided by the RCP bus undervoltage reactor trip. For loss of flow events initiated by a decrease in RCP bus frequency, reactor protection is provided by the RCP bus underfrequency reactor trip. The low coolant flow trip provides protection from all other loss of flow events and serves as a backup for the previously discussed functions. These reactor trips are adequate to ensure that limiting DNB ratios are maintained above the design limit for any loss of flow event.

Method of Analysis

The two limiting cases that were analyzed are as follows:

1. Loss of three out of three RCPs from a power level of 2942.2 MWt, due to an undervoltage condition.

2. Loss of three out of three RCPs from a power level of 2942.2 MWt, due to a frequency decay condition (-5 Hz per second).

Partial loss of flow scenarios remain nonlimiting as described in the UFSAR and were not reanalyzed. The analysis considered full cores of North Anna Improved Fuel (NAIF) and Advanced Mark-BW fuel.

The transient analysis utilizes the RETRAN transient analysis code (Reference 36). These transient data are then used in a detailed thermal-hydraulic computation to compute the minimum DNBR values. The COBRA computer code is used with the WRB-1 DNB correlation to calculate a minimum transient DNBR for the resident NAIF fuel. The LYNXT computer code is used with the BWU DNB correlations to calculate a minimum DNBR for the Advanced Mark-BW fuel at the limiting statepoint condition.

The initial operating conditions for power, pressure, flow, and RCS temperature are assumed to be at their nominal value which is consistent with the statistical treatment of key analysis parameters (References 19, 20).

Results

Both the underfrequency and the undervoltage trip events were analyzed. The transient responses of pressurizer pressure, nuclear power, heat flux, mass flow rate, and core inlet temperature versus time are plotted in Figures 6-13 through 6-17 for the undervoltage case and 6-18 through 6-22 for the underfrequency case. The minimum DNBRs for the two accidents showed significant margin to the design DNBR limit for the transient analysis of the NAIF with an $F_{\Delta H}^{N}$ of 1.490. The DNBR values resulting from the limiting statepoints provided in the analysis were higher than the required value for the Advanced Mark-BW fuel with an $F_{\Delta H}^{N}$ of 1.538.

Conclusions

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the limit value during the transient, and thus there is no clad damage or release of fission products to the reactor coolant system.

6.3.3 Locked Rotor

The Locked Rotor/Sheared Shaft (LOCROT) events are characterized by the rapid loss of forced circulation in one Reactor Coolant System (RCS) loop. A Locked Rotor event is defined as the seizure of a Reactor Coolant Pump (RCP) motor due to a mechanical failure. The Sheared Shaft event is defined as the separation of the RCP impeller from the motor due to the severance of the impeller shaft. For both the Locked Rotor and the Sheared Shaft events, the postulated RCP failure causes the reactor coolant flow rate to decrease more rapidly than a normal RCP coastdown. The description of this event and details of the analysis are unaltered from that presented in UFSAR Section 15.4.4, except as noted below. Certain other description is presented here for completeness.

Maintaining the fuel cladding integrity is a primary safety criterion for the Locked Rotor/Sheared Shaft event, although integrity may not be maintained for all fuel rods. Therefore, maintaining the RCS as a fission product barrier becomes more significant. Specifically, RCS integrity may be challenged as a result of the volumetric expansion of the fluid caused by the heating of the RCS fluid. Operation of the pressurizer sprays and Power Operated Relief Valves (PORVs) can help limit the impact of the subsequent pressure increase, but cannot counteract the volumetric expansion of the RCS fluid. In general, the short duration of the locked rotor event acts in concert with the functioning of the pressurizer safety valves (PSVs), to prevent excessive RCS pressurization. Thus, timely actuation of the Reactor Protection System is also required to help limit the RCS pressure response.

Method of Analysis – General

To cover all applicable phases of plant operation, Locked Rotor and Sheared Shaft events during Modes 1 through 5, as defined in the Technical Specifications, are considered. A transient analysis is only required for the Locked Rotor and Sheared Shaft event at full power with manual rod control. The results of Locked Rotor or Sheared Shaft event analyses at any of the remaining operating conditions are bounded by those of the full power manual rod control case. Except where otherwise noted, the following assumptions are made in the Locked Rotor/Sheared Shaft transient analysis:

1. The DNB analysis employs a statistical treatment of key analysis uncertainties; the transient cases are assumed to initiate from nominal thermal/hydraulic conditions, with an assumed power level of 2942.2 MWt.

2. The main steam and RCS overpressurization analyses employ a deterministic treatment of key analysis uncertainties (102% power; nominal T_{avg} +4°F; nominal pressurizer pressure +30 psi; and RCS thermal design flow).

3. The DNB, RCS overpressurization, and main steam system overpressurization analyses consider full cores of North Anna Improved Fuel (NAIF), Westinghouse standard (STD) 17x17 fuel, and Advanced Mark-BW fuel.

Transient Analysis for DNB and Fuel Cladding Integrity

The transient analysis for DNB and fuel cladding integrity considerations utilizes the RETRAN transient analysis code (Reference 36). These transient data are then used in a detailed thermal-hydraulic computation to compute the minimum DNBR values. The COBRA computer code is used with the WRB-1 DNB correlation to calculate a minimum transient DNBR for the resident NAIF fuel. The LYNXT computer code is used with the BWU DNB correlations to calculate a minimum DNBR for the Advanced Mark-BW fuel at the limiting statepoint condition.

The transient analysis for DNB is performed to determine the number of fuel pins that experience DNB as a result of a Locked Rotor or Sheared Shaft event. A fuel pin is assumed to fail if the predicted MDNBR is less than the statistical DNBR design limit. The Locked Rotor DNB event scenario is therefore designed to produce the most limiting DNB response. From an analytical perspective, this goal is achieved by choosing initial conditions and analysis assumptions that will maximize coolant temperature and the power-to-flow ratio, and minimize pressure during the event. The MDNBRs for the two accidents showed significant margin to the design DNBR limit for the transient analysis of the NAIF with an $F_{\Delta H}^{N}$ of 1.490. The DNBR values resulting from the limiting statepoints provided in the analysis were higher than the required value for the Advanced Mark-BW fuel with an $F_{\Delta H}^{N}$ of 1.538. The analysis demonstrates that the fraction of fuel failure for this event is less than that which has been demonstrated to provide acceptable dose consequences. Therefore, there is no effect upon dose results.

Transient Analysis for RCS and Main Steam Overpressurization

The transient analysis for RCS and main steam overpressurization considerations utilizes the RETRAN transient analysis code (Reference 36). The transient analysis for overpressurization considerations verifies that the peak RCS pressure (intact cold leg pump exit pressure) and peak main steam pressure (intact loop steam generator pressure) remain below 110% of RCS and main steam design pressure (2750 psia and 1210 psia, respectively). The Locked Rotor overpressurization event scenario is designed to produce the most limiting overpressurization response. From an analytical perspective, this goal is achieved by choosing initial conditions and analysis assumptions that will minimize RCS energy removal and minimize core coolant expansion during the transient.

Figures 6-1 through 6-7 show the transient response for the Locked Rotor event (limiting RCS overpressure analysis). Figure 6-8 has been extracted from the MSS overpressure analysis. RCS and MSS overpressure criteria were met by the analysis.

Conclusions

The following conclusions are applicable to the analyzed scenarios:

a. A coolable core geometry is maintained throughout the transient, since the Departure from Nucleate Boiling Ratio (DNBR) transient analysis demonstrates that limited fuel failure due to the onset of DNB is predicted to occur.

b. Acceptable offsite dose consequences are ensured, since the analysis demonstrates that the fraction of fuel rods predicted to experience Departure from Nucleate Boiling (DNB) is less than that which provides acceptable offsite dose analysis results.

c. Reactor Coolant System (RCS) integrity is maintained throughout the transient as demonstrated by analysis of transient RCS pressure. Specifically, the maximum RCS pressure, which occurred in the intact cold leg pump exit, remained below 2750 psia throughout the transient.

d. Main Steam System (MSS) integrity is maintained throughout the transient as demonstrated by analysis of transient MSS pressure. Specifically, the maximum main steam pressure, which occurred in the intact loop steam generator, remained below 1210 psia throughout the transient.

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 in draft publication form (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 accident experiments performed on test rods in conjunction with detailed fuel rod behavior analyses. The resulting limit has a constant threshold of 170 cal/gm below a rod burnup of 36,000 MWD/MTU, reducing to approximately 125 cal/gm at 60,000 MWD/MTU. Therefore, this analysis also compares the results to the following recently proposed limits on HZP cases consistent with the recommendations of Reference 37:

1. Average hot spot fuel enthalpy less than 170 cal/gm (306 BTU/lb) for fresh fuel and fuel irradiated less than 36,000 MWD/MTU, and

2. Average hot spot fuel enthalpy exponentially decreasing from 170 to 125 cal/gm (225 BTU/lb) for irradiated fuel from 36,000 MWD/MTU to 60,000 MWD/MTU.

Method of Analysis

The analysis of the RCCA-ejection accident is performed in two stages, consistent with the NRC-approved methodology in Reference 38. First, an average core nuclear power transient calculation is performed using point neutron kinetics methods to determine the average power generation with time, including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity and then a hot spot heat transfer calculation. Second, enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot-channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient.

The point kinetics model of the RETRAN computer code is used to perform the average core transient analysis. This code includes the simulation of prompt and delayed neutrons (using the six-group model), the thermal kinetics of the fuel and moderator and the balance of the NSSS primary and secondary coolant system. Thermal feedback effects are modeled via temperature dependent reactivity coefficients with a detailed multi-region, transient fuel-clad-coolant heat transfer model. Reactivity insertion from the ejection of the control rod and the subsequent reactor trip are accounted for. Since both the axial and radial dimensions are missing, it is necessary to use very conservative methods (described below) of calculating the ejected rod worth and hot-channel factor.

The average core energy addition, calculated as described above, is multiplied by the appropriate hot-channel factors, and the hot spot analysis is performed using the detailed fuel and clad transient heat transfer model of the RETRAN code termed the Hot Spot Model, (see Reference 38). This model calculates the transient temperature distribution in a cross section of a metal-clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power generation is used within the fuel rod.

Transient Analysis

The analysis considers both a full core of NAIF fuel and a full core of Advanced Mark-BW fuel. The transient is analyzed at four different core conditions consistent with the UFSAR analysis of record evaluation methodology. These conditions are:

- Beginning of cycle (BOC), hot zero power (HZP)
- Beginning of cycle, hot full power (HFP)
- End of cycle (EOC), hot full power
- End of cycle, hot zero power

The peak ejected F_Q and ejected rod reactivity parameters for the EOC HZP case were modified from the current values for both fuel types to accommodate the more limiting acceptance criteria described in Reference 37.

Results

Figures 6-9 through 6-12 show the transient response for the limiting cases, BOC-HFP and EOC-HZP. The analysis found that the results assuming a full core of Advanced Mark-BW fuel bound the results for a full core of NAIF fuel.

Beginning of Cycle, Full Power. The peak hot spot fuel center temperature exceeded the assumed BOC melt temperature of 4,900°F. However, melting was restricted to less than 10% of the pellet.

Beginning of Cycle, Zero Power. The peak hot spot clad temperature was less than the BOC melt temperature of 4,900°F.

End of Cycle, Full Power. The peak hot spot fuel temperature exceeded the assumed EOC melt temperature of 4800°F. However, melting was restricted to less than 10% of the pellet.

End of Cycle, Zero Power. The peak hot spot clad temperature was less than the EOC melt temperature of 4,800°F.

Conclusions

The analyses indicate that the described fuel and clad limits are not exceeded for current limits and for potential future limits described in Reference 37. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the primary loop. The analyses have demonstrated that the upper limit in fission product release, in terms of the number of fuel rods entering departure from nucleate boiling, amounts to 10%.

Figure 6-1: Locked Rotor, RCS Pressures.

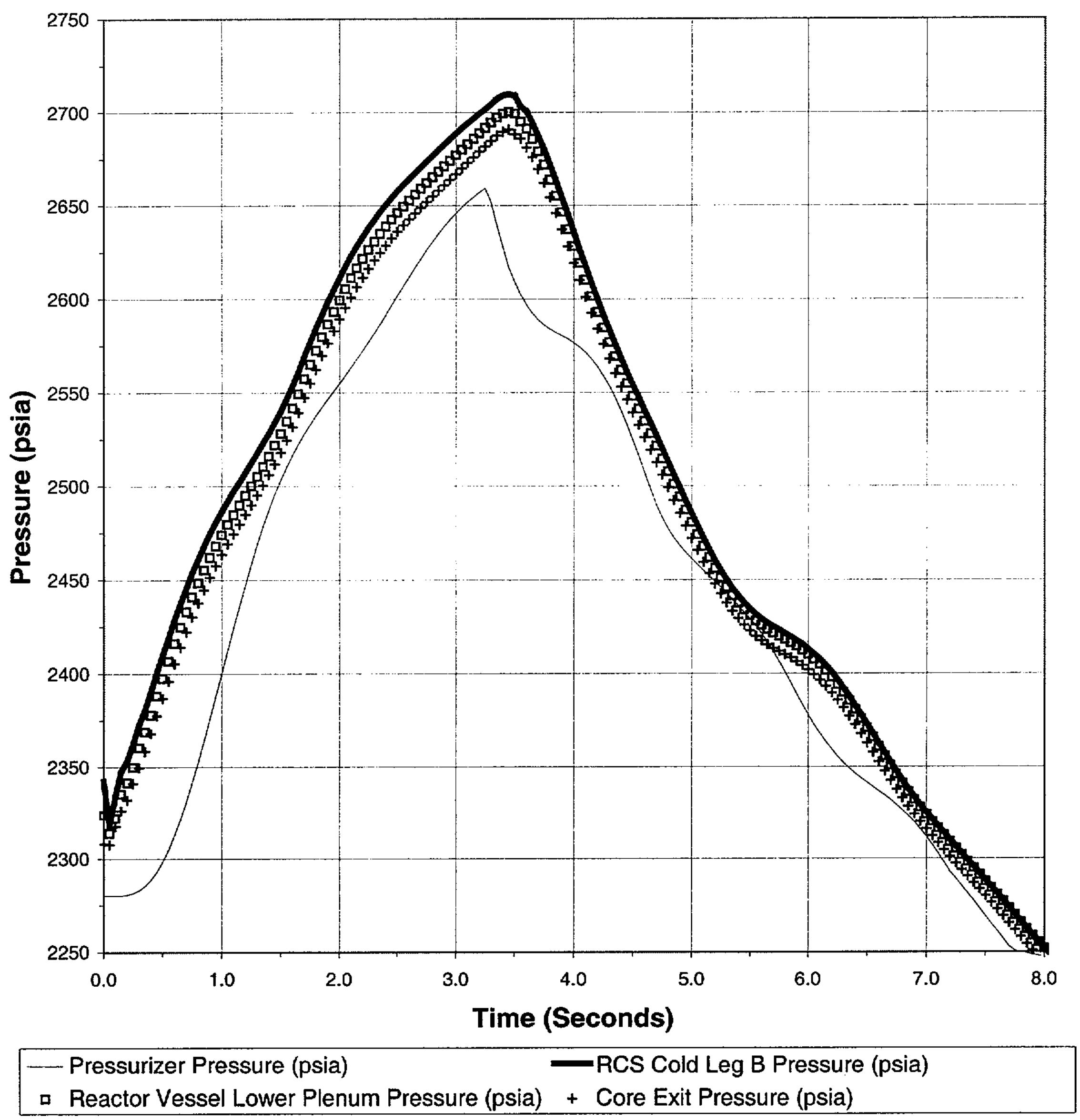


Figure 6-2: Locked Rotor, Core Inlet Temperature.

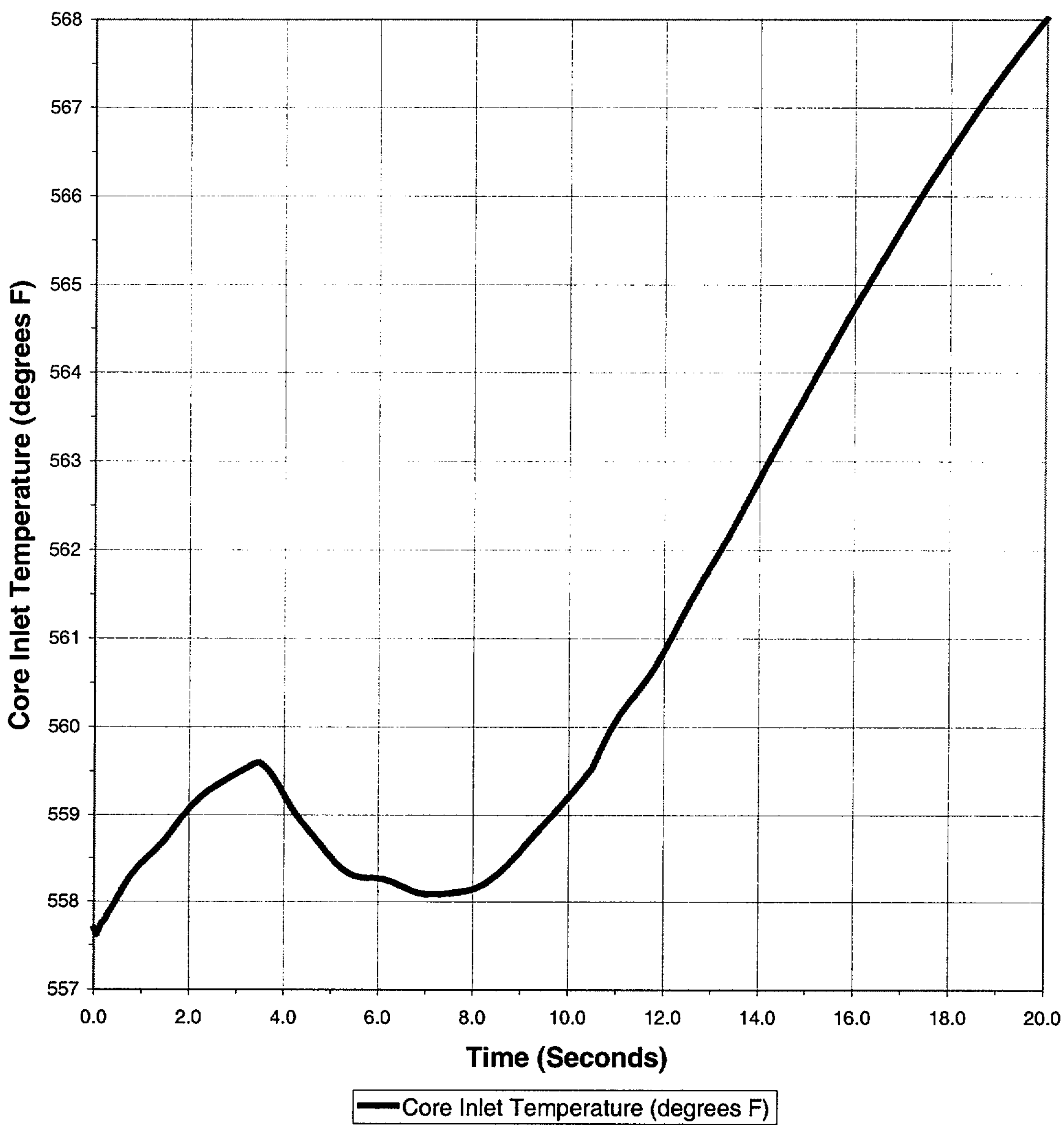


Figure 6-3: Locked Rotor, Core Inlet Flow Rate.

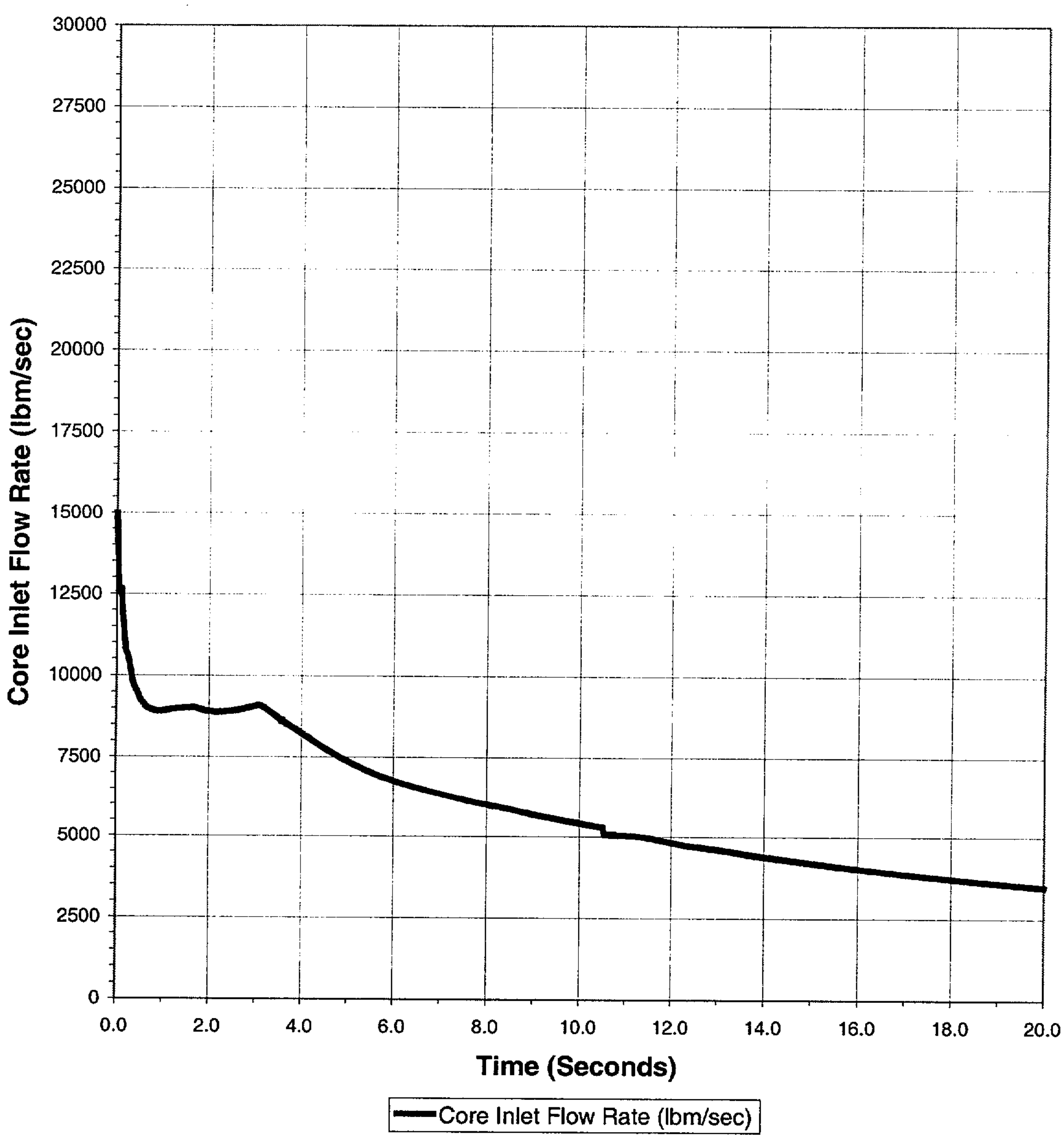


Figure 6-4: Locked Rotor, Core Average Heat Flux.

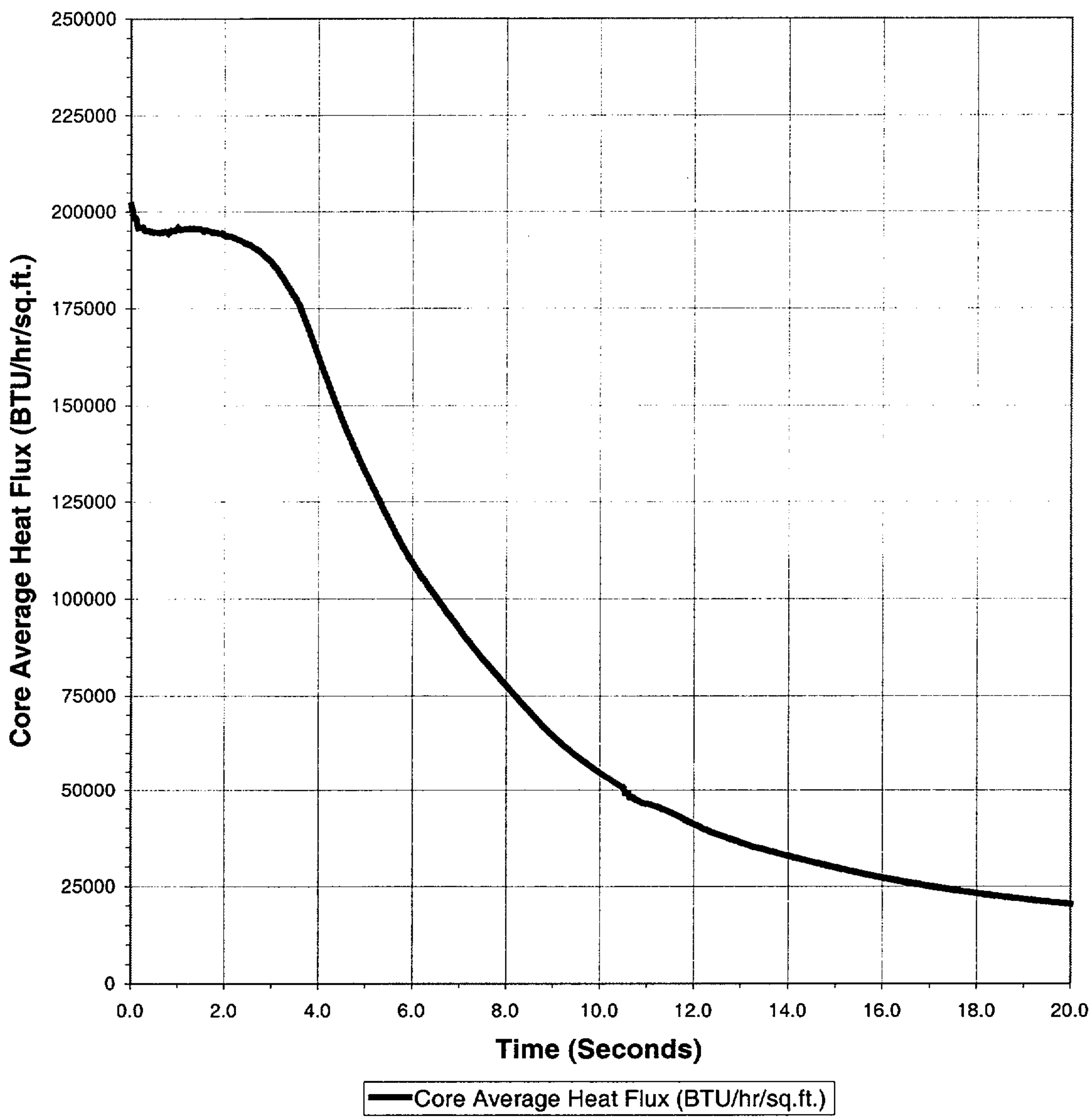


Figure 6-5: Locked Rotor, Nuclear Power.

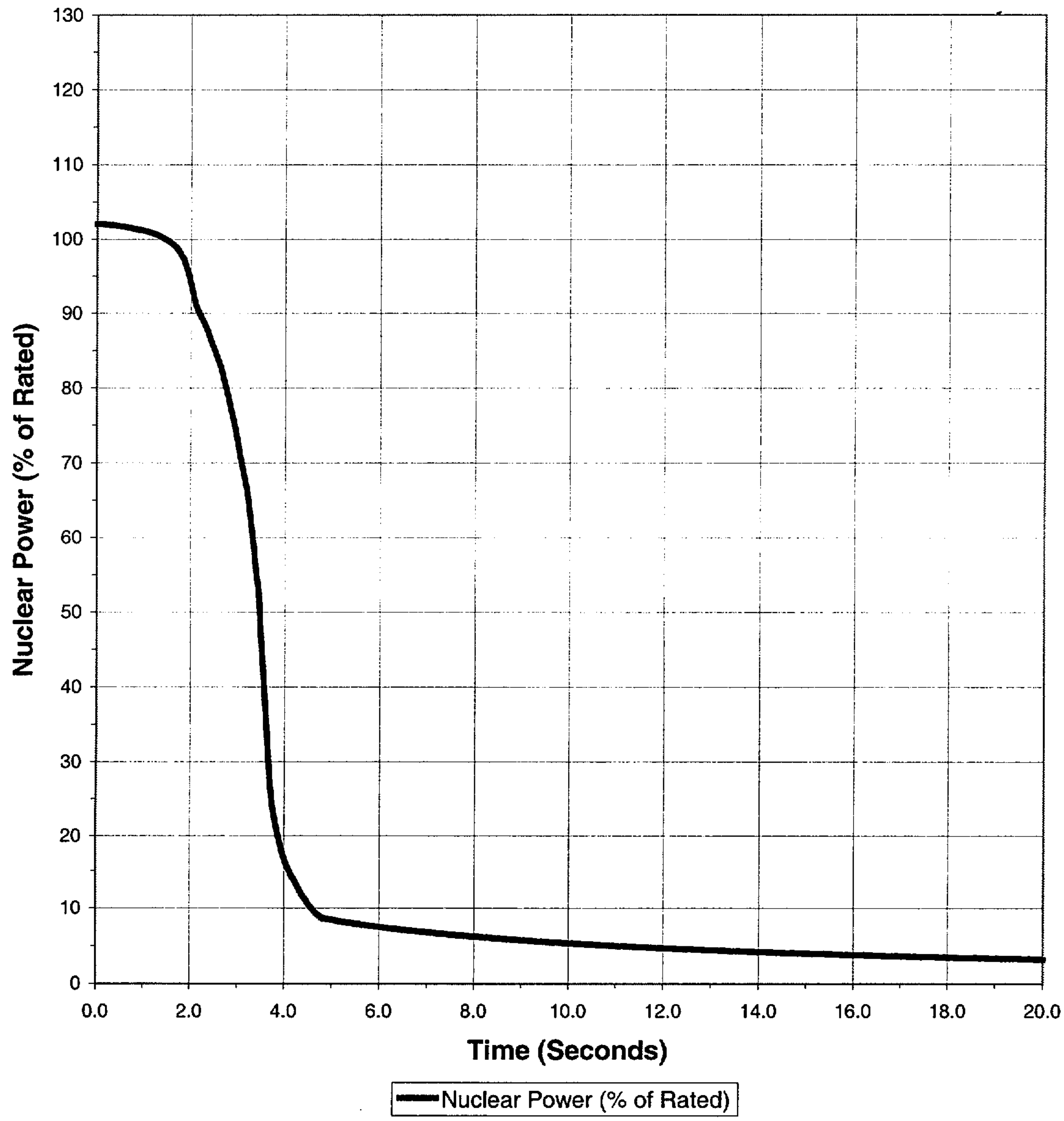


Figure 6-6: Locked Rotor, Main Steam Safety Valve #1 Flow Rate.

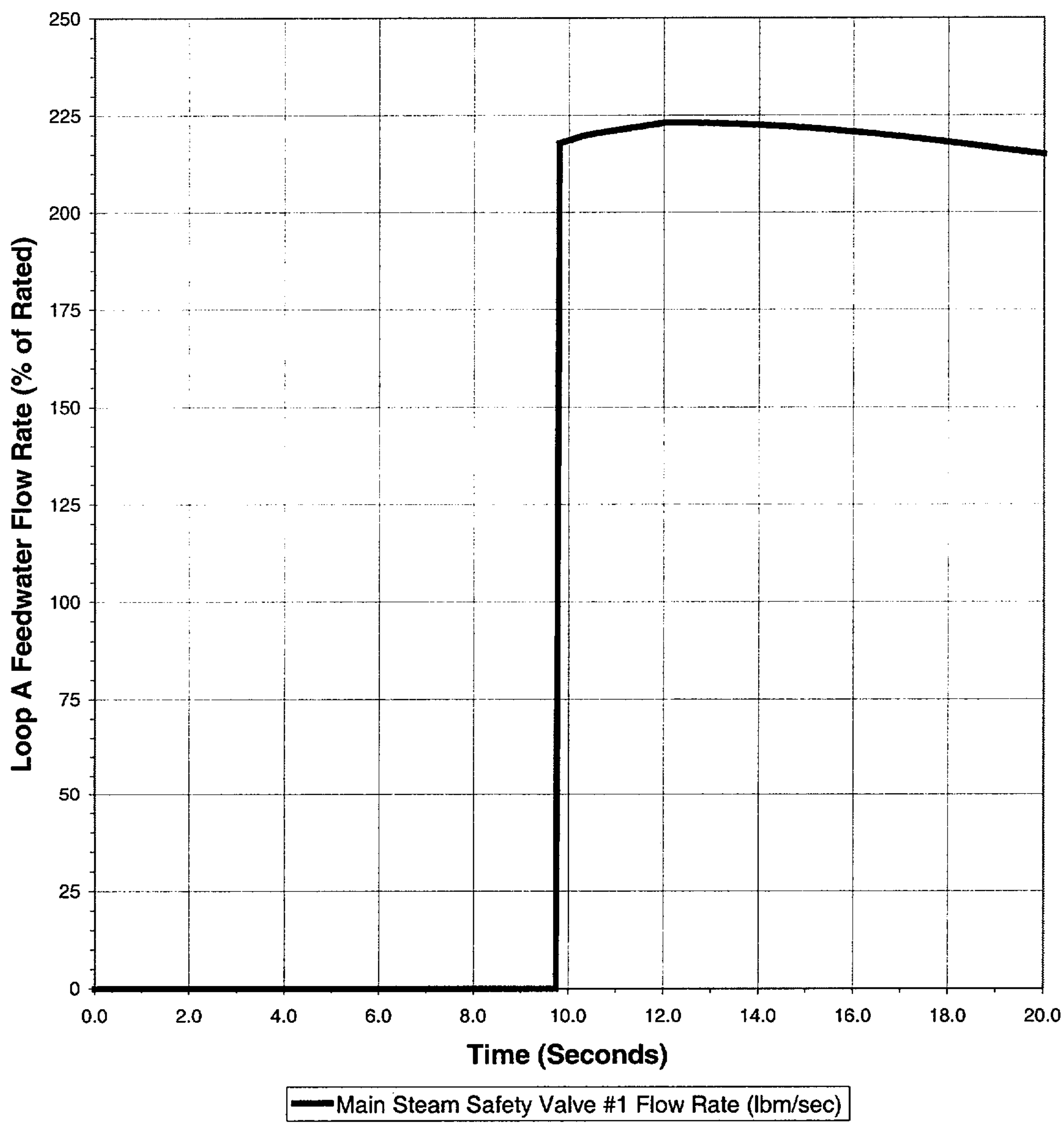


Figure 6-7: Locked Rotor, Pressurizer Safety Valve Flow Rate (Total of 3 PSVs).

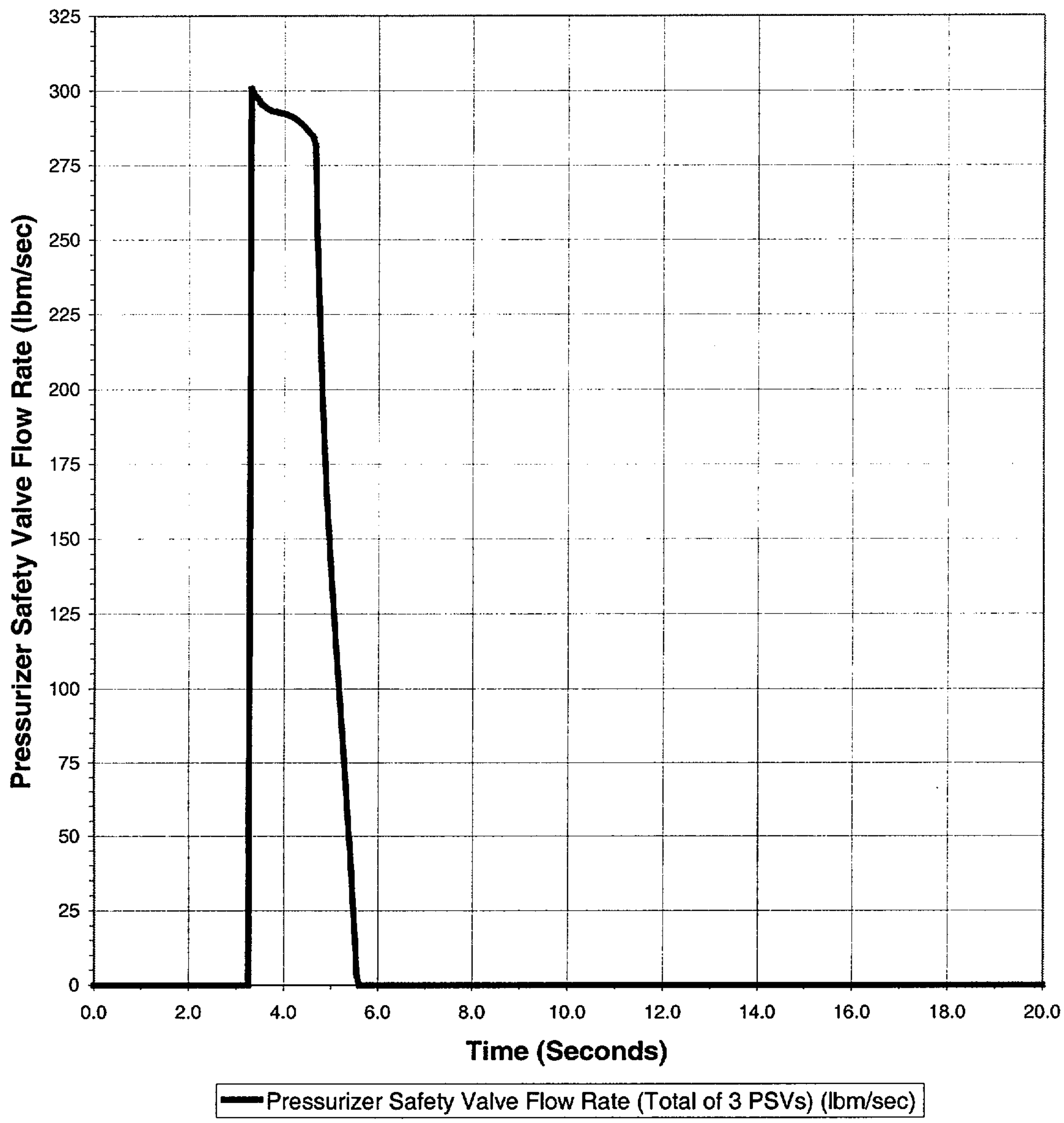


Figure 6-8: Locked Rotor, Steam Generator B Pressure.

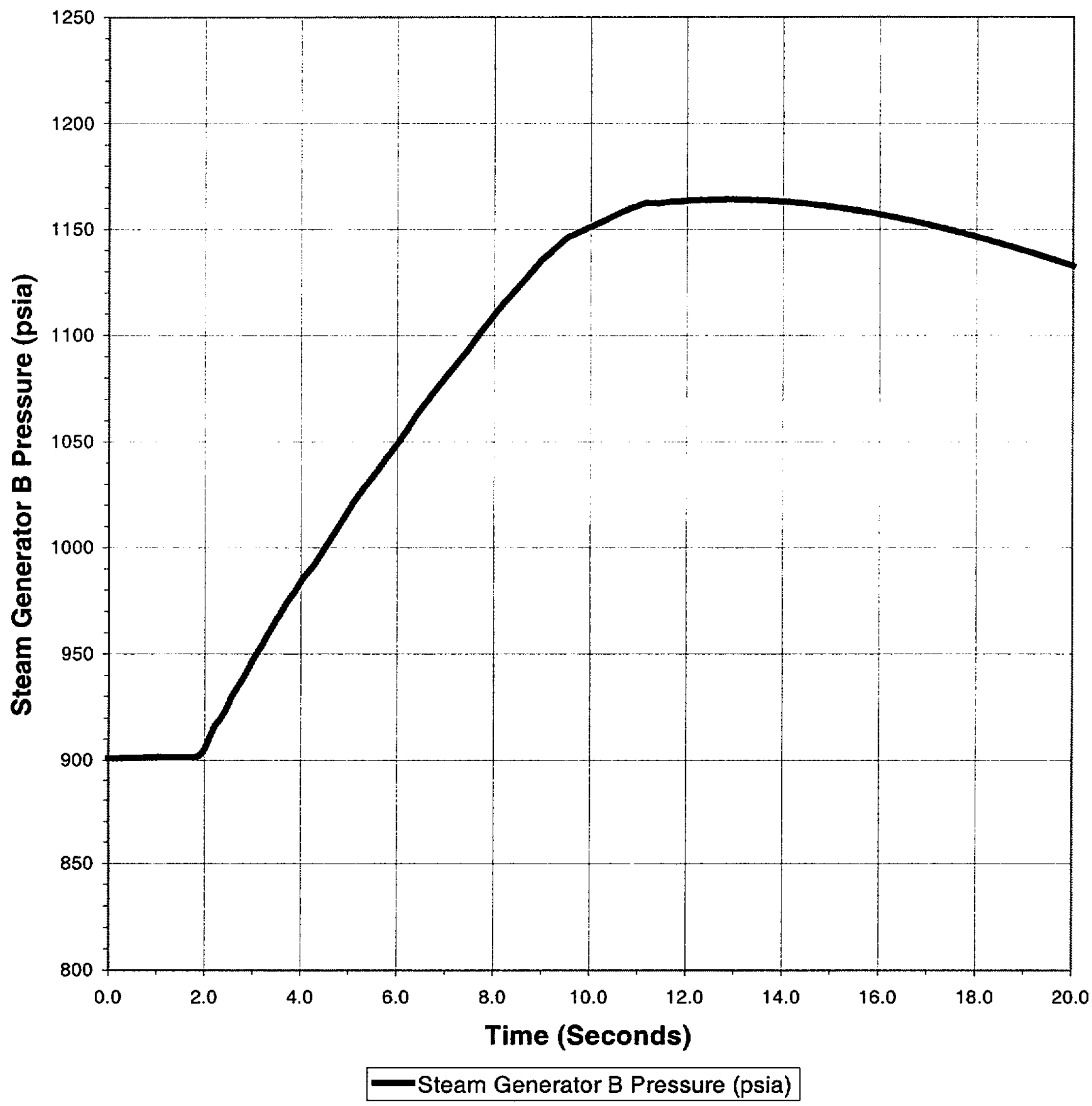


Figure 6-9. Nuclear Power Transient, BOL HFP Rod Ejection Accident. Framatome Fuel

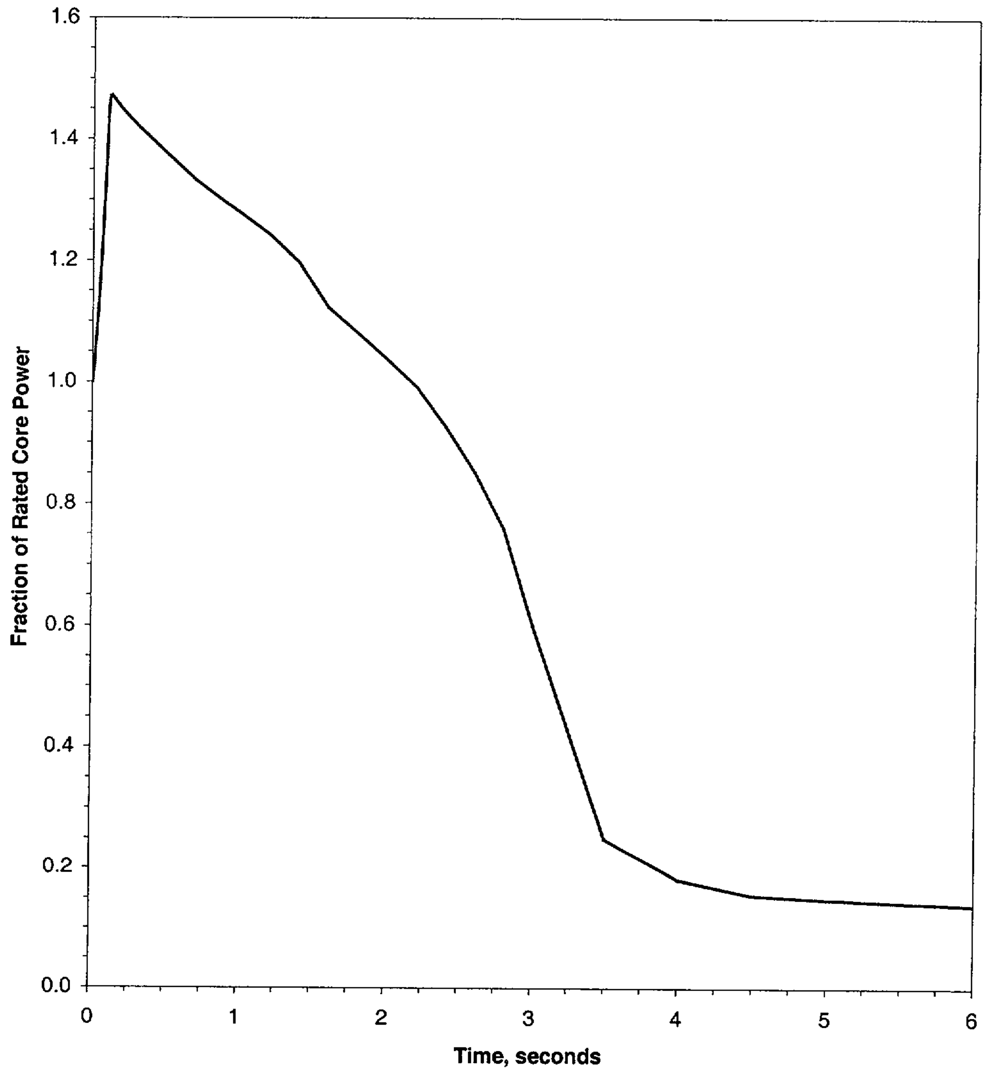


Figure 6-10. Hot Spot Fuel and Clad Temperature Versus Time, BOL HFP Rod Ejection Accident, Framatome Fuel

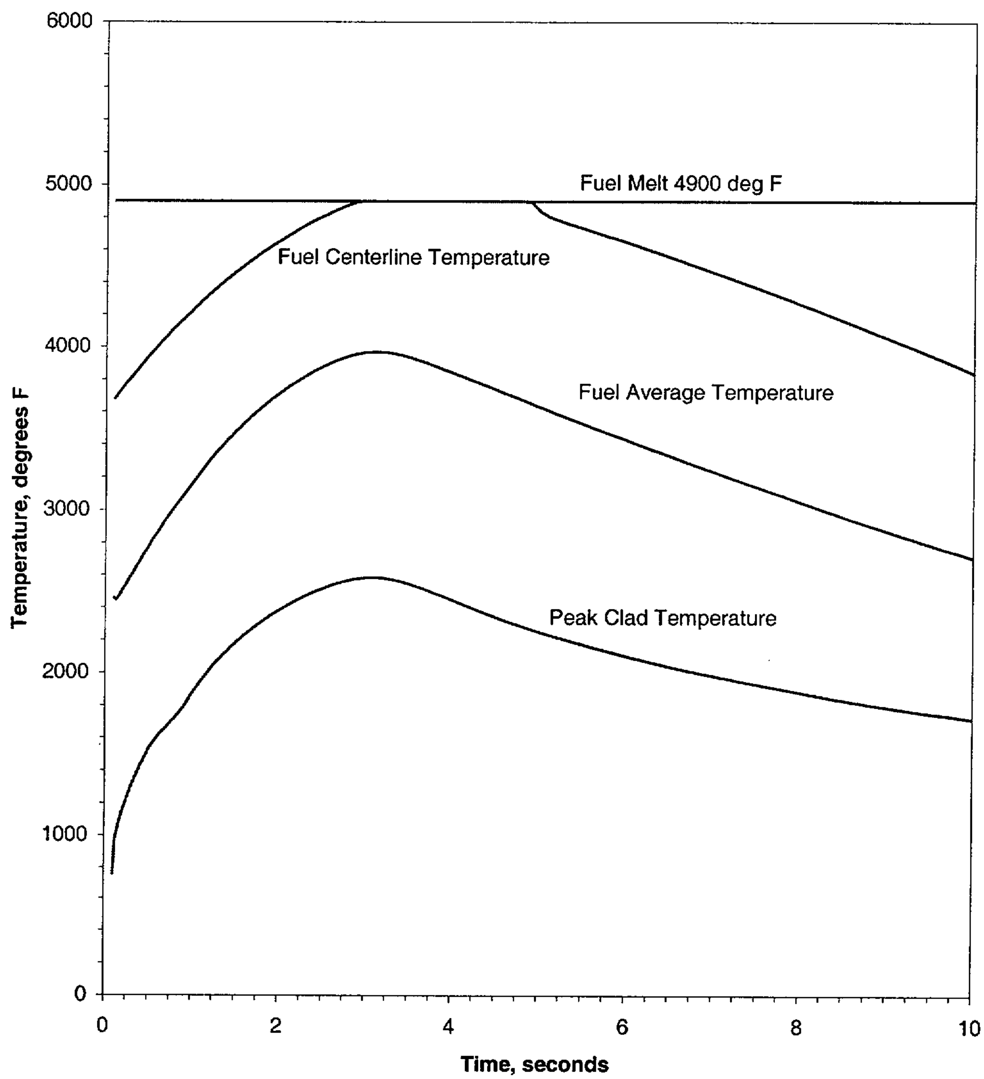


Figure 6-11. Nuclear Power Transient, EOL HZP Rod Ejection Accident, Framatome Fuel

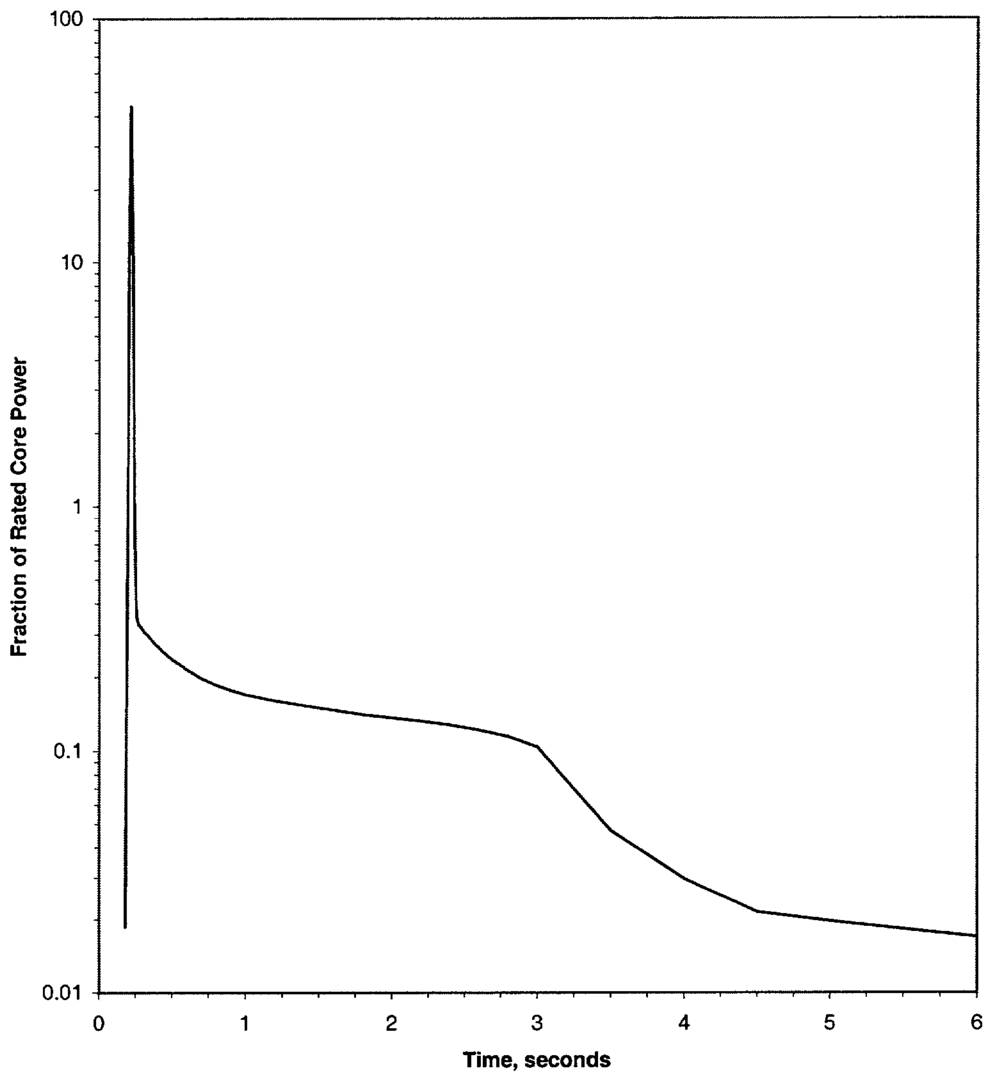


Figure 6-12. Hot Spot Fuel and Clad Temperature Versus Time, EOL HZP Rod Ejection Accident, Framatome Fuel

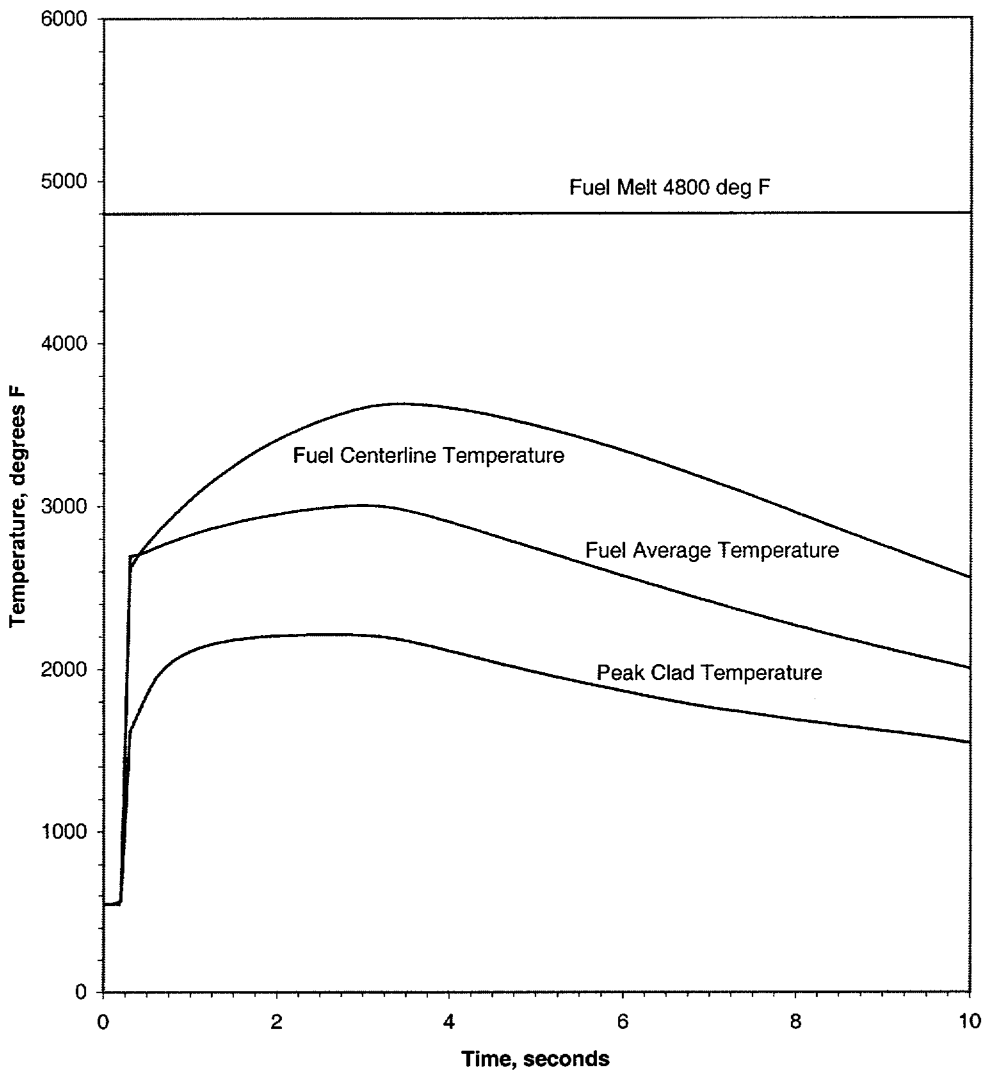


Figure 6-13: Loss of Flow Event, Undervoltage, Pressurizer Pressure versus Time, Framatome Uprate Core.

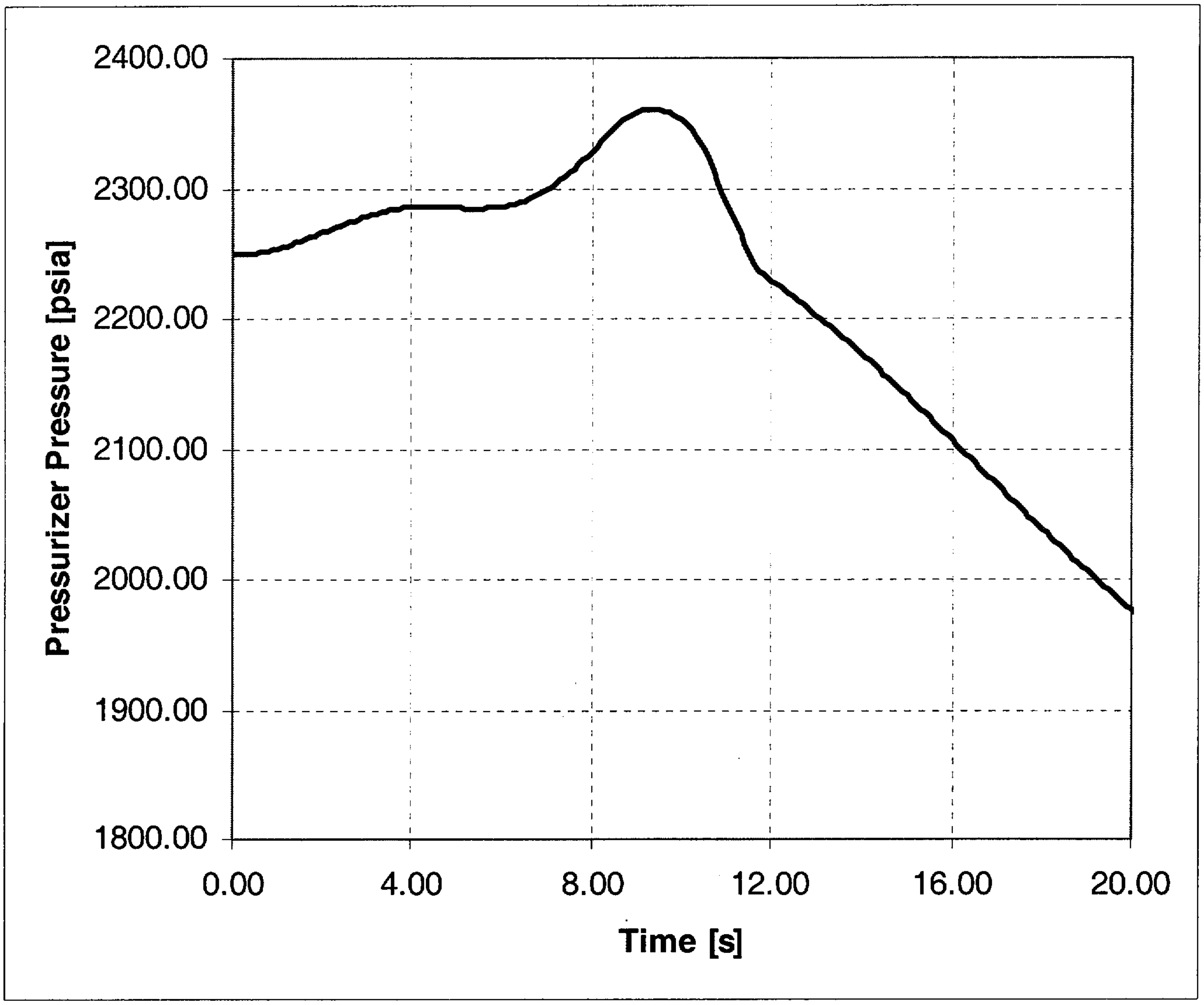


Figure 6-14: Loss of Flow Event, Undervoltage, Nuclear Power versus Time, Framatome Uprate Core.

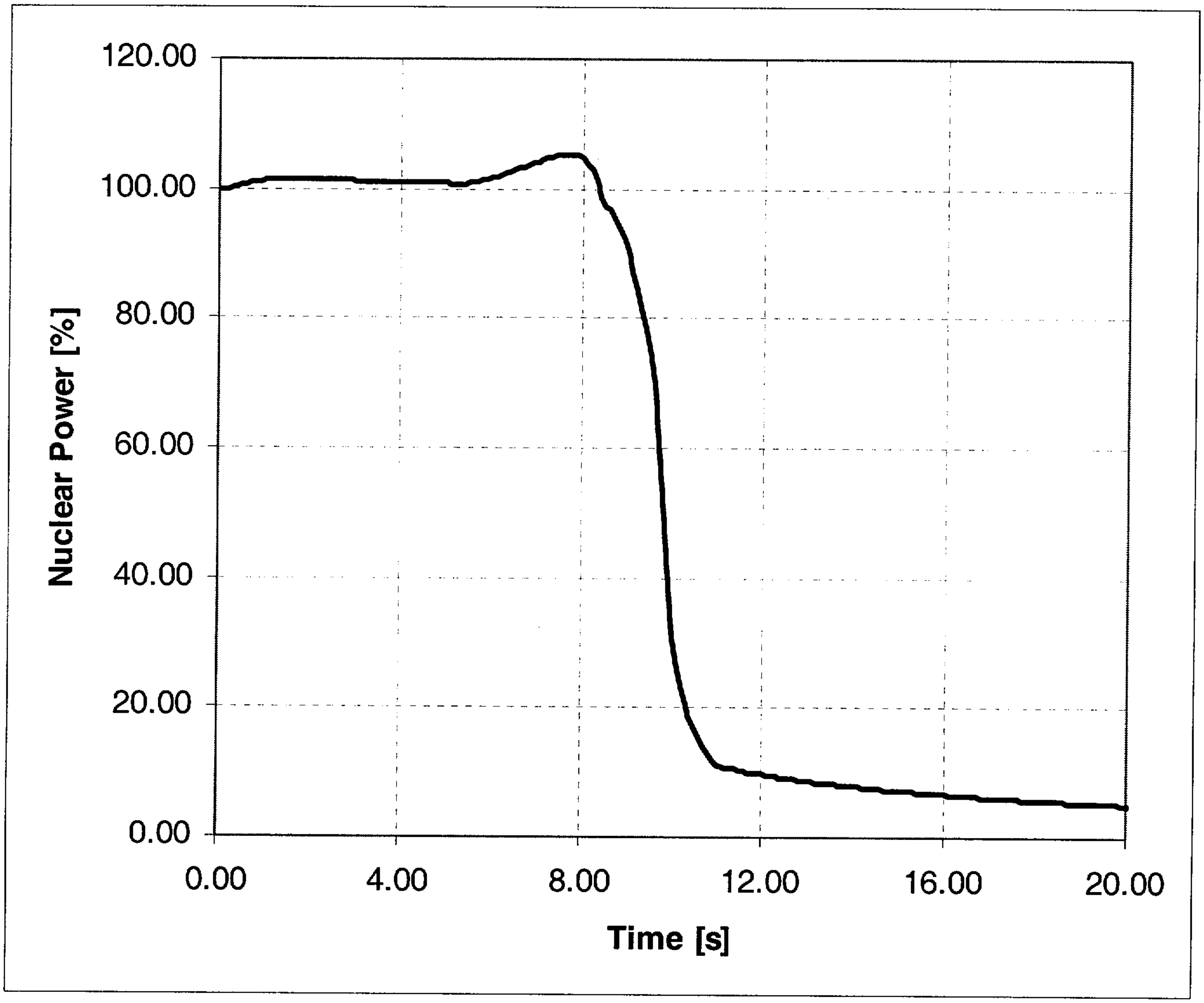


Figure 6-15: Loss of Flow Event, Undervoltage, Core Average Heat Flux versus Time, Framatome Uprate Core.

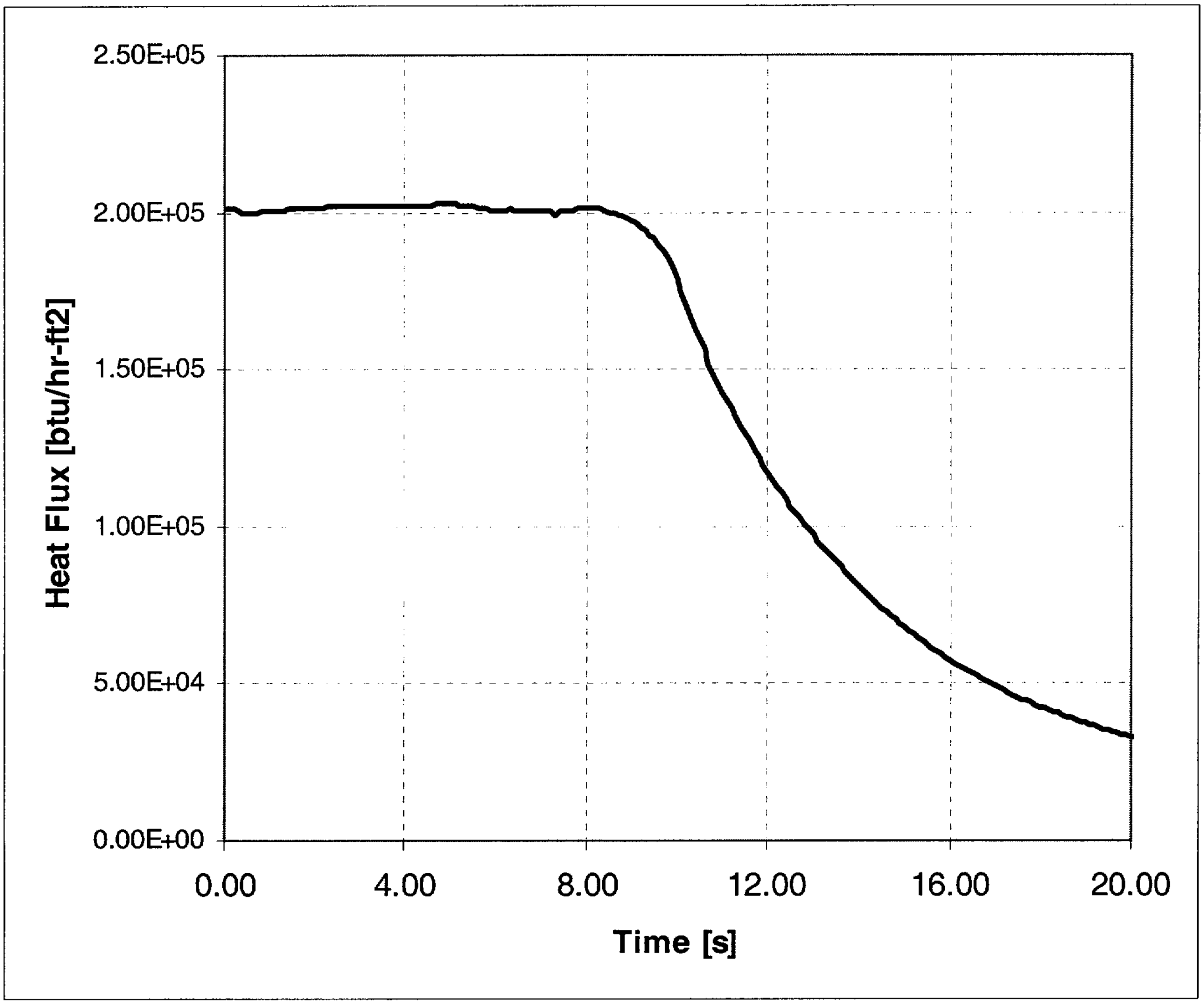


Figure 6-16: Loss of Flow Event, Undervoltage, RCS Mass Flow versus Time, Framatome Uprate Core.

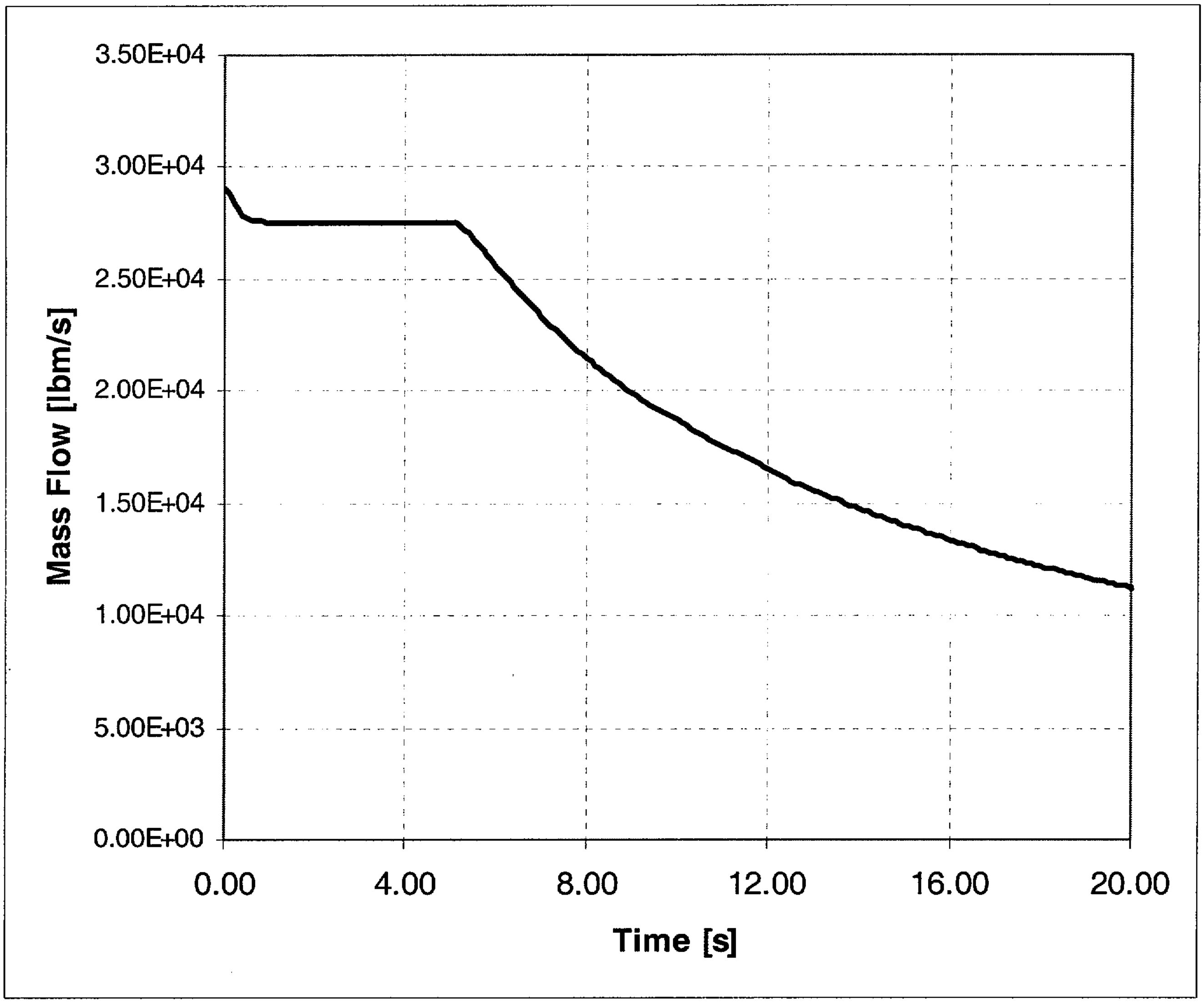


Figure 6-17: Loss of Flow Event, Undervoltage, Core Inlet Temperature versus Time, Framatome Uprate Core.

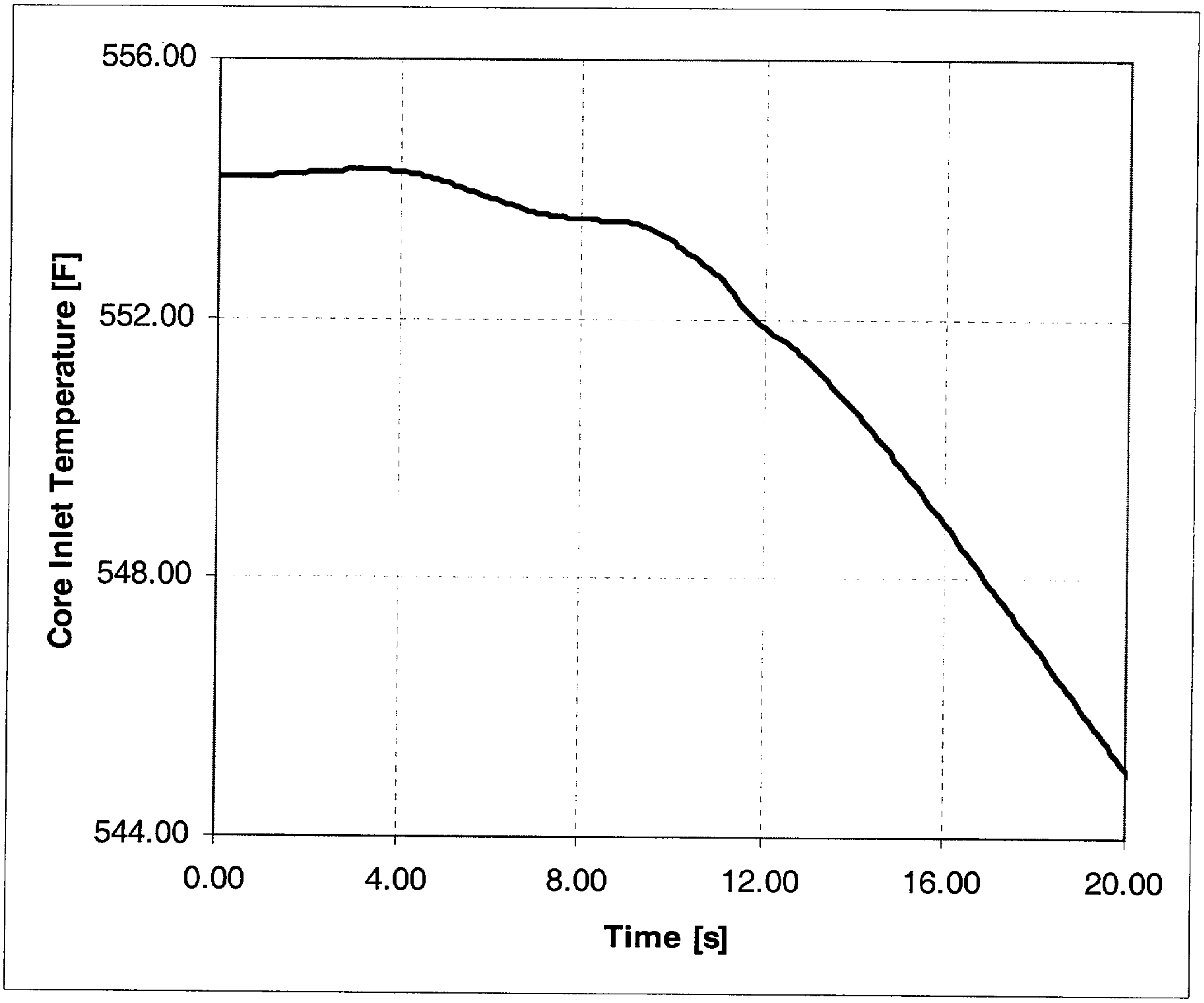


Figure 6-18: Loss of Flow Event, Underfrequency, Pressurizer Pressure versus Time, Framatome Uprate Core.

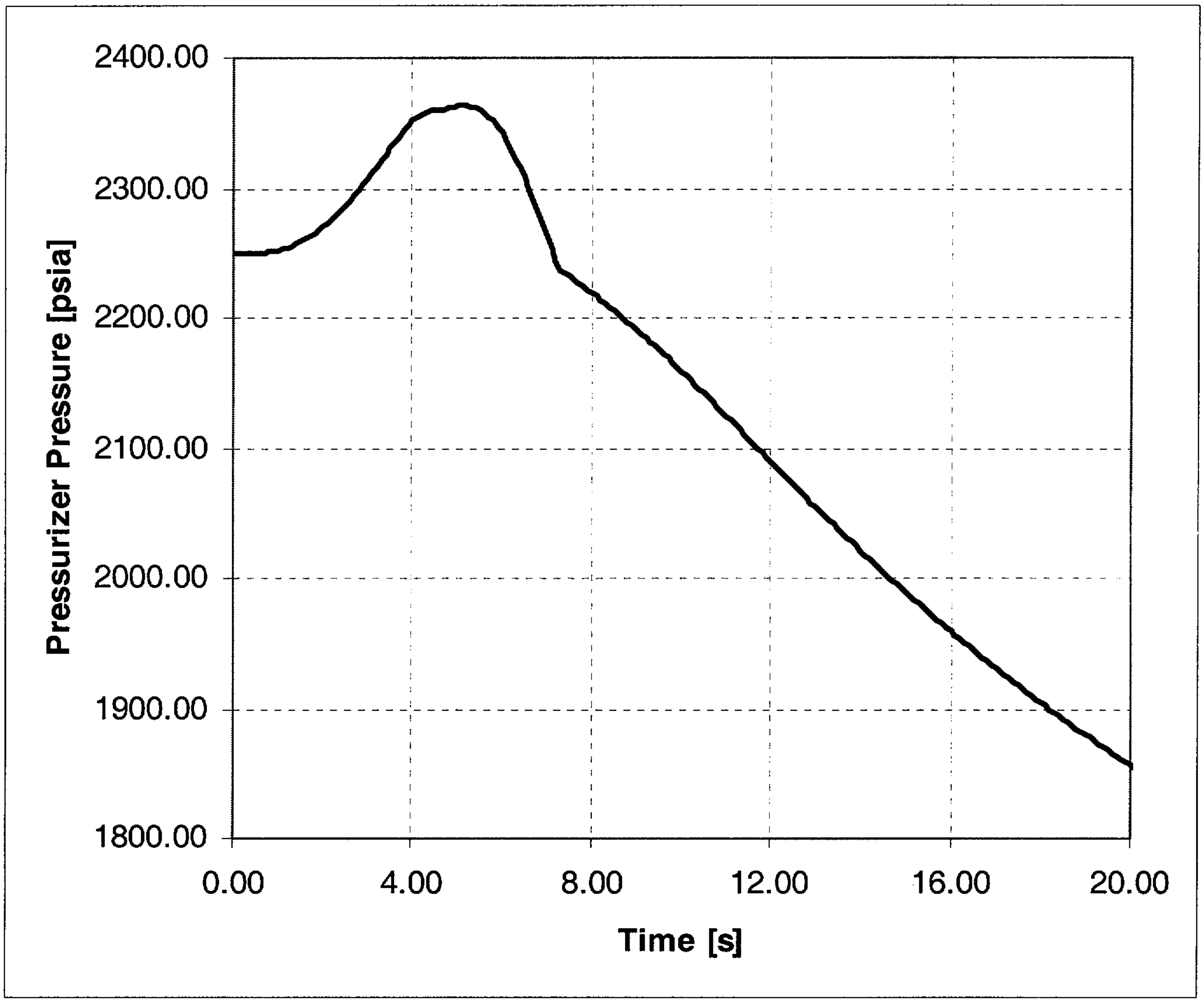


Figure 6-19: Loss of Flow Event, Underfrequency, Nuclear Power versus Time, Framatome Uprate Core.

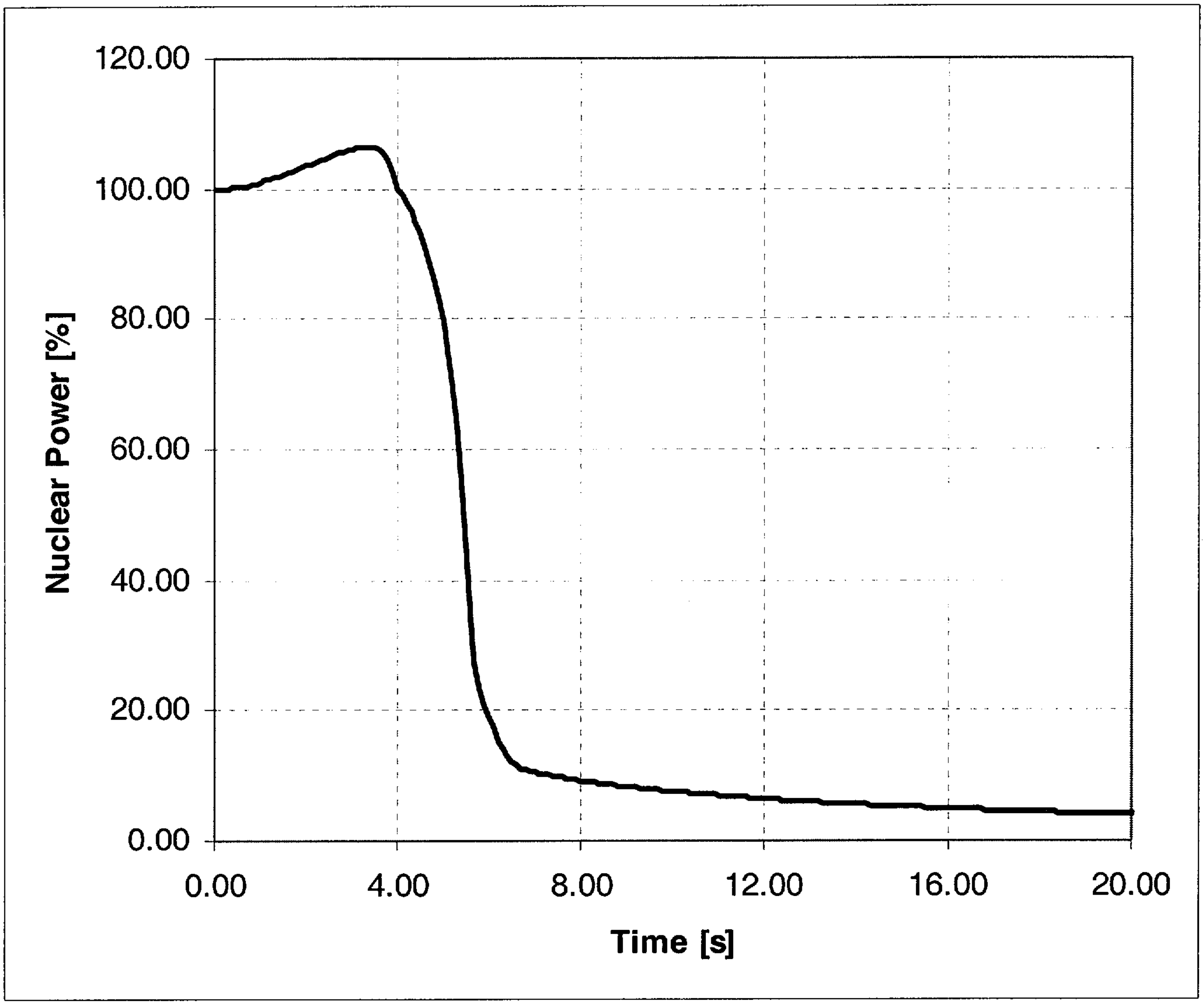


Figure 6-20: Loss of Flow Event, Underfrequency, Core Average Heat Flux versus Time, Framatome Uprate Core.

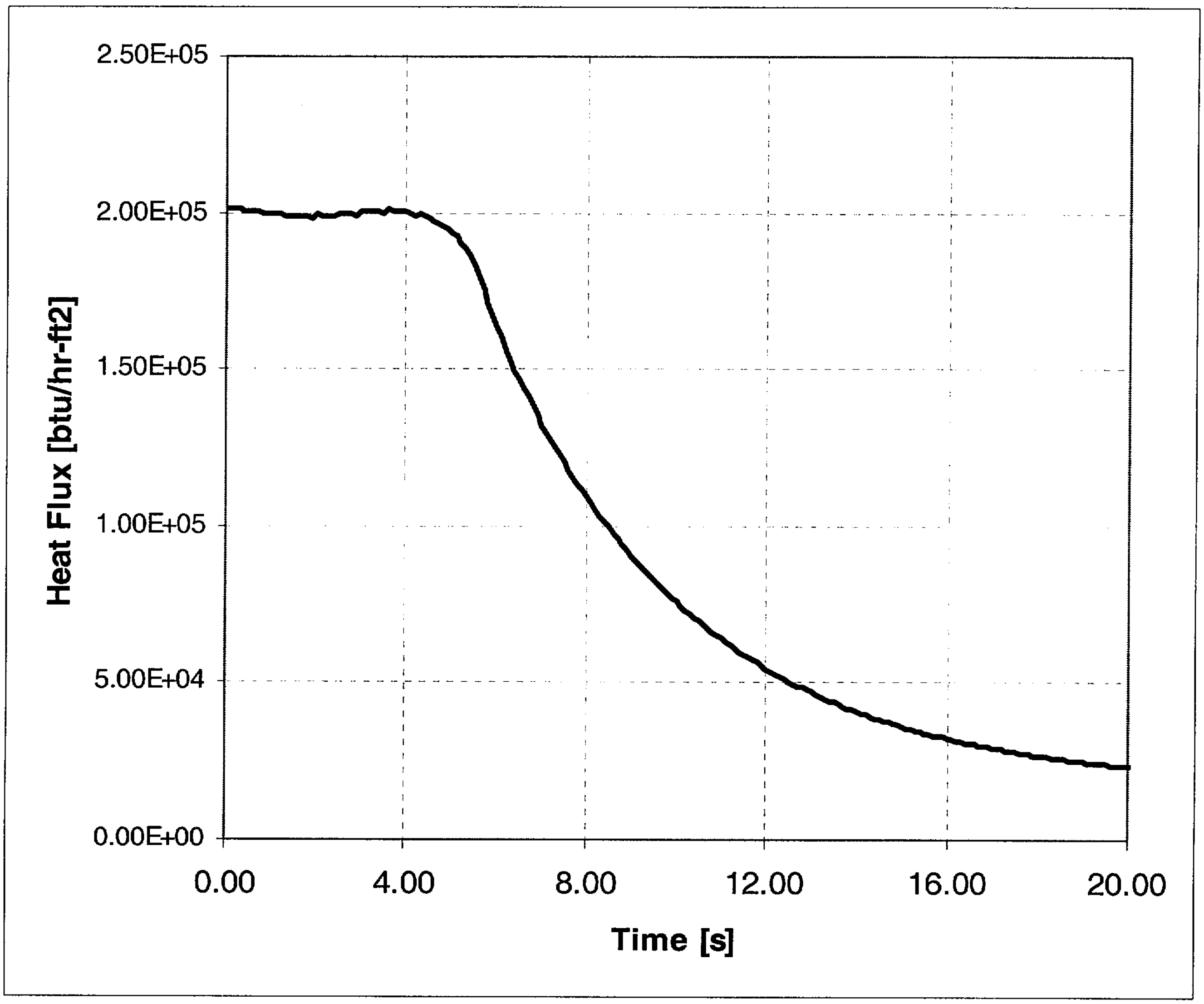


Figure 6-21: Loss of Flow Event, Underfrequency, RCS Mass Flow versus Time, Framatome Uprate Core.

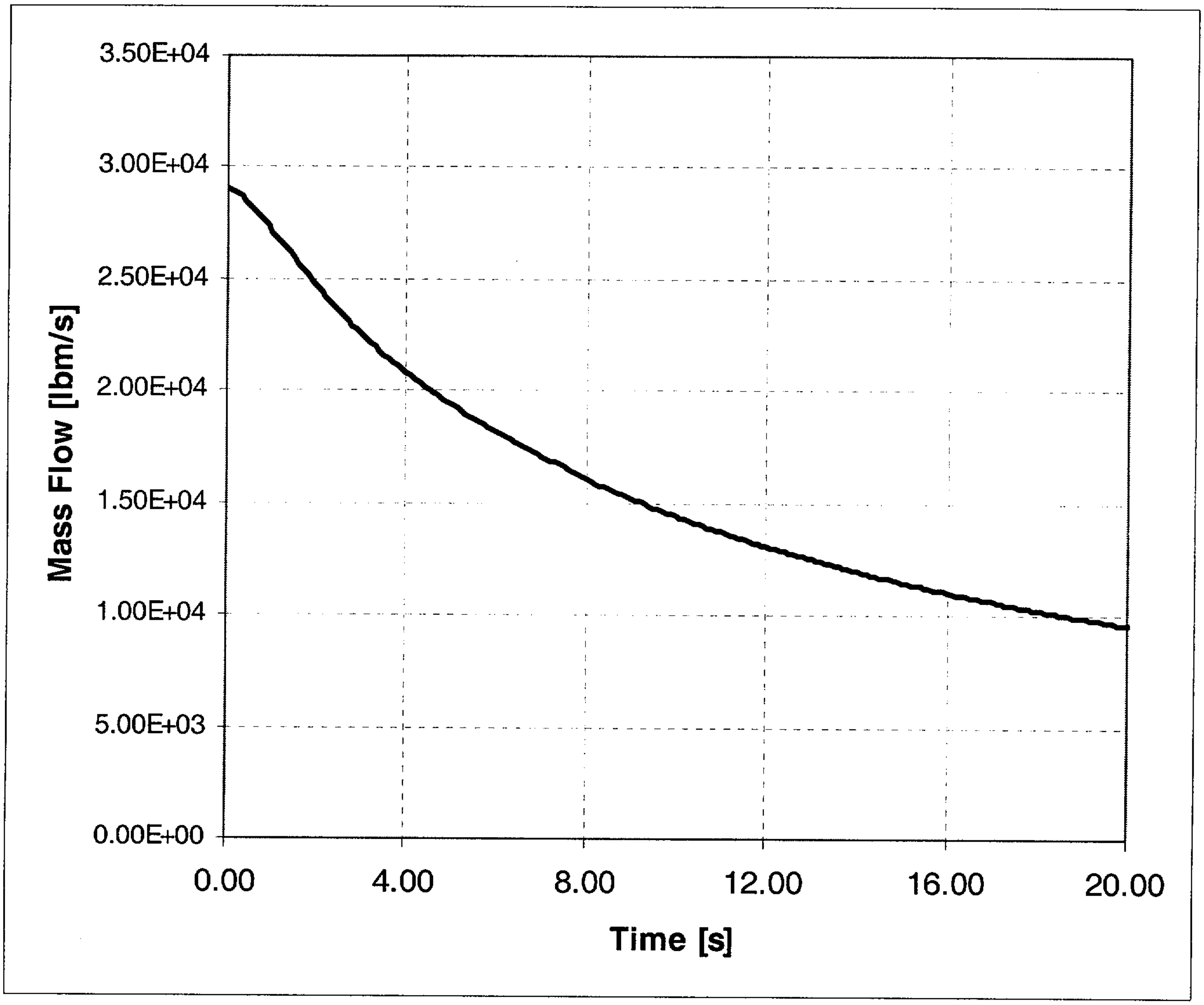
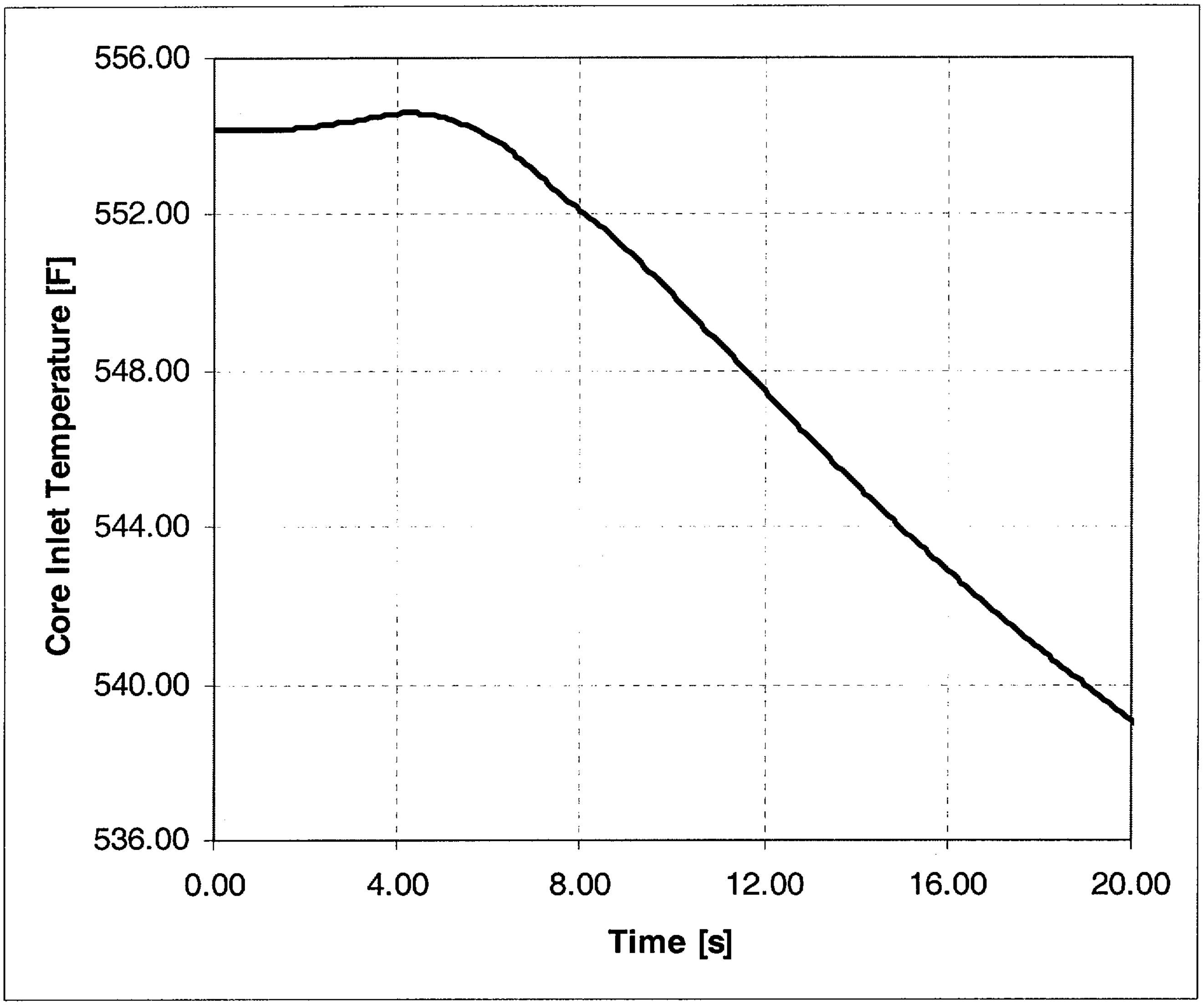


Figure 6-22: Loss of Flow Event, Underfrequency, Core Inlet Temperature versus Time, Framatome Uprate Core.



7.0 LOCA/ECCS Evaluation

In accordance with the requirements of 10 CFR50.46 and 10 CFR50, Appendix K, an evaluation of ECCS performance will be performed for the Advanced Mark-BW fuel. The analyses are presently underway and will be reported in a future revision to this report. The LOCA analyses employ methods awaiting approval by the Nuclear Regulatory Commission. This chapter will document compliance with 10 CFR50.46 when the North Anna units are fueled with Framatome Advanced Mark-BW fuel, including transition cores containing both the Framatome and Westinghouse fuel designs. LOCA predictions are divided into two categories based on break size. For breaks larger than 0.5 ft^2, compliance is demonstrated by analyses performed in accordance with Framatome's large break LOCA, recirculating steam generator (RSG), evaluation model (EM)—BAW-10168, Volume I (Reference 30). For small break LOCA (SBLOCA)—breaks less than 0.5 ft^2—Volume II of the RSG EM is the compliance document.

The future report revision will provide a brief description of the computer codes and calculation methods used in the North Anna LOCA analyses. The documentation will address key analysis parameters, generic sensitivity studies, plant-specific sensitivity studies and break spectrum analyses to determine the limiting break configuration. Also presented will be results of the LOCA limit calculations that demonstrate compliance to the first two criteria of 10 CFR50.46: peak clad temperature (PCT) and maximum cladding oxidation. Evaluation of the remaining acceptance criteria (maximum hydrogen generation, coolable geometry, and long-term cooling) is also addressed.

8.0 Applicability of Dominion Reload Design Methodology

Dominion performs reload safety evaluations using a bounding analysis method as described in Topical Report VEP-FRD-42, Rev. 1-A (Reference 27). This methodology defines a set of key analysis parameters that fully describe a valid conservative safety analysis ("reference analysis"). If all key analysis parameters for a reload core are conservatively bounded by the corresponding parameters in the reference analysis, the reference safety analysis is bounding, and further evaluation is not necessary. When a key analysis parameter is not bounded, further review is considered necessary to ensure that the required safety margin is maintained. This last determination is made through either a complete reanalysis of the accident, or through a simpler, though conservative, evaluation process using known parameter sensitivities. This same reload evaluation process will be employed for cores containing the Advanced Mark-BW fuel product.

Reference 27 has been revised to support the transition to Advanced Mark-BW fuel at North Anna. The NRC, in the SER for Reference 27 stated, "it is clear that the methodology presented is closely related to the Westinghouse methodology, and is applicable in its present form only to Westinghouse supplied reloads of Westinghouse nuclear plants." Revision 2 of this topical has been prepared and submitted to NRC (Reference 23) to address this restriction and to present revised discussion of the Dominion reload core design methodology. The changes address several types of items that are listed here:

- Applicability of methodology for analysis of incremental fuel design differences
- Generic methodology items impacted by transition to Framatome fuel
- Consolidation of prior Dominion submittals regarding code and model updates
- Responses to original NRC Staff review questions
- Miscellaneous editorial changes

The revised topical discusses the Dominion capability to assess changes in fuel design. The focus of the specific topical changes is primarily upon nuclear core design and NSSS safety analysis design inputs. The minor changes in the Advanced Mark-BW fuel features that could affect safety analysis design inputs are within the modeling capability of Dominion safety and core design analysis codes. These changes are within the scope of similar plant and fuel design changes that Dominion has successfully analyzed and implemented during prior operation of the North Anna and Surry plants.

A second topical report involved in the Dominion reload design methodology that is impacted by the transition from Westinghouse fuel to Framatome Advanced Mark-BW fuel is VEP-NE-1-A, "Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," or RPDC. VEP-NE-1-A documents the Dominion methodology for determining the maximum allowed axial power skewing permissible for core operation. This methodology is used for establishing operating limits for axial offset at North Anna. The NRC Staff reviewed VEP-NE-1-A for applicability to analyze non-Westinghouse fuels and reported its findings in Reference 24. The Staff concluded that this methodology is plant-specific, but is not fuel-specific and that its use for non-Westinghouse cores and mixed cores is within the scope of Generic Letter 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses." It is concluded from the Reference 24 review that the revision to VEP-NE-1-A for application to the Advanced Mark-BW fuel does not require NRC approval.

Dominion will complete the revision to VEP-NE-1-A prior to the commencement of the applicable RPDC analyses for the loading of Framatome fuel (North Anna Unit 1, Cycle 17). VEP-FRD-42, Revision 2 is expected to be approved in Spring 2002, which is well before the introduction of the Advanced Mark-BW fuel in Spring 2003. These topical revisions will ensure that documented, valid methodologies are used in the standard Dominion reload design process for North Anna reload cores containing the Advanced Mark-BW fuel product.

9.0 Spent Fuel Pool Criticality

Criticality analyses of the North Anna fresh fuel storage area (FFSA) and the Spent Fuel Pool (SFP) have been reviewed for applicability with Advanced Mark-BW fuel. The original analyses were performed using Westinghouse fuel design data. The NRC Safety Evaluation and approval of new Technical Specifications based on these calculations is contained in Reference 31. Differences in key analysis parameters between the Advanced Mark-BW fuel and the original analysis data were reviewed. The only differences of significance for the SFP and FFSA analyses are use of the M5™ alloy (clad, grid, and guide thimbles), the increased nominal fuel density, and the tolerances on fuel density and pellet volume. Grid differences are not relevant since grids were intentionally ignored in the original calculations for conservatism.

The calculation estimated the impact of the Advanced Mark-BW design differences using sensitivity, tolerance, uncertainty and margin data from the existing calculational documentation. Using the approved methodology upon which the Reference 31 amendment was based, the higher fuel density of the Advanced Mark-BW design was evaluated to quantify the increase in SFP and FFSA K-effective.

The calculations resulted, as expected, in slightly higher calculated K-effectives for the North Anna FFSA and the SFP. These calculations have also demonstrated that there is sufficient margin to the K-effective limits to accommodate this increase with no changes to the methodology or to the Technical Specifications (geometric and burnup limitations). Storage of Advanced Mark-BW fuel with initial enrichments up to 4.6 w/o U-235 would continue to assure sufficient margin under the current design FFSA and SFP design bases and Technical Specifications.

10.0 Impact on Radiological Consequences of Accidents

The North Anna design basis dose consequence analyses were reviewed to determine whether the accident analyses bound the expected consequences for use of the Advanced Mark-BW fuel. The analyses reviewed include the loss of coolant accident (LOCA), fuel handling accident (FHA), locked rotor accident (LRA), main steam line break (MSLB), and steam generator tube rupture (SGTR).

Key design inputs of the radiological consequences analysis were assessed to identify those inputs that may potentially be impacted by use of the Advanced Mark-BW fuel at North Anna. The items that were considered for potential impact are assessed as indicated below:

Design Input	Change for Advanced Mark-BW
Fuel Theoretical Density (TD)	Increasing from 95% to 96%
Peak Rod Average Burnup	Unchanged at 60,000 MWD/MTU (lead rod)
Peak Enthalpy Rise Hot Channel Factor, $F_{\Delta H}$ (FHA only)	Unchanged at 1.65
Core average power	Unchanged at 102% of 2893 MWt
Enrichment	Unchanged at 4.6 w/o U-235
Event-Specific % Cladding (Fuel) Failure	Unchanged
Rod internal pressure (RIP) 100 hours after shutdown (FHA only)	Unchanged at $\leq$ 1200 psig

Of the items considered above, only the potential change in fuel theoretical density (TD) will be considered further. Theoretical density is expected to increase from a nominal 95% TD to a nominal 96% TD, which will result in a 1% increase in fuel in the core. This will produce an increase in cycle length as measured in Effective Full Power Days (EFPD). However, since the core power level, burnup and $F_{\Delta H}$ remain bounded by existing analysis values, this potential fuel inventory increase will not cause an increase in the radiological source terms assumed for the LOCA, FHA, or LRA.

Of the events evaluated, only the radiological analysis for the FHA has a direct dependency upon detailed fuel-related design parameters. The other event analyses source terms are related to the total core inventory or allowable values of primary-to-secondary leakage and primary coolant activity. These quantities are not impacted by operation with the Advanced Mark-BW fuel product. The FHA event is discussed further below.

The existing North Anna FHA dose consequences analyses generally follow the guidance of Regulatory Guide 1.25 (Reference 25) and the Standard Review Plan (Reference 26). For the FHA event, this results in a conservative analysis that models release of the fission product gap inventory of all the rods in a single fuel assembly having power of 1.65 times the core average value. The analysis assumes a pool decontamination factor (DF) of 100, which is valid for a depth of water above the failed fuel > 23 ft. and fuel rod internal pressure $\leq$ 1200 psig.

The core fission product inventory is very sensitive to burnup and power level. The production of short-lived isotopes is directly proportional to power level and the production of long-lived isotopes in directly proportional to burnup. The dose consequence analyses are typically limited by thyroid dose, which is a direct result of the quantity of I-131 (a short-lived isotope) in the core inventory. The I-131 inventory is most closely correlated with the core power immediately before reactor shutdown. As indicated above, the core power level and burnup for operation with the Advanced Mark-BW fuel is bounded by the values assumed in existing radiological analyses.

This review has indicated that the key design inputs to which the existing radiological event analyses are sensitive are either unchanged or the changes are of no impact. It is concluded that the existing radiological consequences analyses remain applicable for operation of North Anna with the Advanced Mark-BW fuel.

11.0 Environmental Assessment

These Technical Specification changes and the associated exemptions from the Code of Federal Regulations to allow the use of Framatome ANP Advanced Mark-BW fuel meet the eligibility criteria for categorical exclusion from an environmental assessment set forth in 10 CFR 51.22(c)(9), as discussed below:

(i) The license condition and associated exemptions from the Code of Federal Regulations involve no Significant Hazards Consideration.

As discussed in the attached evaluation of the Significant Hazards Consideration, the use of Framatome ANP Advanced Mark-BW fuel at North Anna will not involve a significant increase in the probability or consequences of an accident previously evaluated. The possibility of a new or different kind of accident from any accident previously evaluated is also not created, and the proposed use of Advanced Mark-BW fuel does not involve a significant reduction in a margin of safety. Therefore, the proposed use of the Framatome fuel product meets the requirements of 10 CFR 50.92(c) and does not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The Framatome ANP Advanced Mark-BW fuel is very similar in design to the resident fuel in the core, and will be handled and operated in the same manner as the other fuel assemblies in the North Anna core. Adherence to the fuel design criteria, verified as part of the cycle specific reload evaluation, will ensure the integrity of the cladding as a fission product barrier for the planned operating conditions. There will be no measurable increase in the isotopic levels in the coolant associated with the normal operation of this fuel, and so no effect on normal operating plant releases. As discussed in this evaluation (Section 10.0), it was concluded that the existing radiological consequences analyses remain applicable for operation of North Anna with the Advanced Mark-BW fuel. Therefore, use of the Advanced Mark-BW fuel will not significantly change the types, or significantly increase the amounts, of effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The Advanced Mark-BW fuel is functionally identical to the resident fuel, and will be handled, operated, and stored in the same manner as the other fuel assemblies in the North Anna core. Operation with the Framatome fuel product will not significantly affect the plant operating conditions. Cycle specific reload evaluations will verify that fuel rod design criteria are satisfied, ensuring that cladding integrity is maintained. Operation of Advanced Mark-BW fuel will not significantly increase radiation levels compared to the current NAIF fuel, so individual and cumulative occupational exposures are unchanged.

Based on the above, the proposed use of the Advanced Mark-BW fuel does not have a significant effect on the environment, and meets the criteria of 10 CFR 51.22(c)(9). It is concluded that the proposed Technical Specification changes and associated exemptions from the Code of Federal Regulations qualify for a categorical exclusion from a specific environmental review by the Commission, as described in 10 CFR 51.22.

12.0 Conclusions

Dominion plans to load the Framatome Advanced Mark-BW fuel into the North Anna Unit 1 core in Spring 2003. The Advanced Mark-BW 17x17 fuel assembly includes these key features:

- Quick disconnect top nozzle
- M5™ cladding, guide tubes and spacer grids
- TRAPPER™ bottom nozzle, with coarse mesh debris filter
- Floating spacer grid restraint system
- Mid-span mixing grids for improved thermal-hydraulic and LOCA performance

The predecessor to the Advanced Mark-BW, the Mark-BW, was previously approved for use in Portland General Electric's Trojan, Duke Power Company's Catawba and McGuire units, and the Tennessee Valley Authority's Sequoyah Nuclear Units 1 and 2. In these applications, the Mark-BW assembly was shown to be compatible with the Westinghouse standard, Vantage 5H, and OFA fuel assembly designs. For application to North Anna, mechanical and hydraulic evaluations were conducted to establish that the Advanced Mark-BW fuel assembly is structurally and hydraulically compatible with the Westinghouse-supplied NAIF fuel assembly currently in use at North Anna. Based on these specific analyses and the North Anna LTA operational experience with essentially the same design, it is concluded that the Advanced Mark-BW fuel assembly is compatible with the current resident fuel at North Anna.

The structural design requirements for the Advanced Mark-BW fuel assembly were derived in large part from Framatome experience, both in design and in-core operation of similar designs. For application to North Anna, plant specific design requirements and parameters augmented the currently established Mark-BW design criteria and were applied to the Advanced Mark-BW fuel design. These requirements in total are consistent with the acceptance criteria of the Standard Review Plan (NUREG-0800), Section 4.2, and follow the guidelines established by Section III of the ASME code. Code Level A criteria are used for normal operation and Code level D criteria are used for LOCA/seismic. The design basis analyses performed to verify the adequacy of the Advanced Mark-BW fuel assembly in North Anna were as follows:

- Normal operations
- Growth allowances
- Fuel assembly holddown
- Guide thimble buckling
- Spacer grid loads
- Interface with adjacent assembly
- Lateral seismic and LOCA loading
- Fuel assembly vertical LOCA loading
- Fuel assembly component stress
- Shipping and handling loads

These fuel design analyses and evaluations use approved NRC methods 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 Westinghouse fuel assemblies.

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 exceptions are as follows: (1) the 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) and (2) an addendum to the BWU-Z CHF topical report justifying the enhanced CHF performance of the MSMGs which is under review by the NRC.

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.

The fuel design analyses and evaluations have confirmed that all fuel-related design criteria will be met with acceptable margins. These conclusions are valid without restriction upon the placement of Advanced Mark-BW fuel in the core. This includes mixed-core configuration with the resident Westinghouse fuel designs up to and including full cores of Advanced Mark-BW fuel.

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Appendix A

Framatome Mixed-Core Analysis Methodology Description

Appendix A Framatome Mixed-Core Analysis Methodology Description

For any new fuel design inserted on a reload basis, thermal-hydraulic compatibility must be established with the existing, or resident, fuel in the reactor core. This compatibility must extend to protection for DNBR, hydraulic loads, and crossflow velocities (for flow-induced fuel rod vibrations). For each of the compatibility requirements the respective design criteria based on the Standard Review Plan (SRP) are used:

(1) During Condition I and II events, there is 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 on a core-wide basis.

(2) The largest hydraulic loads will be such that there is sufficient fuel holddown margin.

A.1 Framatome Mark-BW Mixed-Core History

Framatome's first mixed-core analyses using the Mark-BW in a core where Westinghouse fuel was the resident fuel was in 1990 for Duke Power Company's (DPCo) Catawba 1 Cycle 6. Since then Mark-BW fuel has been introduced into: DPCo's Catawba 2 Cycle 6, McGuire 1 Cycle 8, and McGuire 2 Cycle 8; Pacific Gas & Electric's (PG&E) Trojan Cycle 14; and Tennessee Valley Authority's (TVA) Sequoyah 1 Cycle 9 and Sequoyah 2 Cycle 9. Since 1990 the Mark-BW fuel has been successfully operated for more than 35 cycles.

In the McGuire and Catawba units the resident fuel was Westinghouse's Optimized Fuel Assemblies (OFA). For Trojan the resident fuel was the Westinghouse Standard Fuel Assemblies (STD). For the Sequoyah units the resident fuel was Westinghouse Vantage 5H (V5H).

A.2 Analyses, Models, and Codes for Mixed-Core Analyses

The specific analyses used to assess the compatibility of different fuel types are DNB, hydraulic loads, and crossflow velocity. These analyses use different models to evaluate the different aspects of a mixed-core. These models are all developed for use with the LYNXT computer program (Reference A-1).

A.2.1 Models

LYNXT (Reference A-1) is used for the thermal-hydraulic calculations that demonstrate the compatibility of the resident and new fuel. The compatibility verification process uses a wide variety of LYNXT models, covering a wide range of model sizes, to adequately justify a core configuration that conservatively represents the mixed-core.

The DNBR analysis of record is typically based on a standard 12-channel one-eighth core LYNXT model as shown in Figure A-1. In this model the power-limiting bundle (PLB), the fuel assembly with the highest radial power, is conservatively modeled by placing the power-limiting pin (PLP), fuel rod with the highest radial power, in the most limiting pin location within the fuel assembly and the most-limiting fuel assembly location in the core. For the 17x17 Mark-BW or Advanced Mark-BW the most-limiting location in the fuel assembly is [

] In addition to the conservative placement of the PLB and PLP the radial power profiles have smaller radial power gradients than typically encountered during the cycle operation, which is also conservative. The 12-channel model spans the length of the fuel pin.

The 12-channel model can be extended for licensing analyses in which additional detail is required. An example is the main steamline break (MSLB) where the inlet boundary conditions are not uniform across the core. [

]

The mixed-core DNBR performance is determined by using a more detailed radially- and axially-noded LYNXT model. [

]

The LYNXT model used for the hydraulic loads and crossflow velocity analyses models each physical fuel assembly in an eighth-core as a separate channel. The model represents the hydraulics axially and radially over the length of the fuel assembly, incorporating the bottom and top nozzles. Figure A-4 shows the [ ] for a 157-fuel assembly plant.

In addition to modeling the fuel assemblies and core geometry it is necessary to model the hydraulic characteristics of the resident and new fuel assemblies using form loss coefficients (FLC). For each Framatome fuel type the FLCs associated with the hardware components are determined with flow tests. For the resident fuel the fuel assemblies may be tested on site using a facility such as Framatome's Transportable Flow Test Rig (TFTR) or the utility may supply axial pressure profiles of the resident fuel's hydraulic performance. For the McGuire and Catawba units the OFA was tested in the TFTR as was the STD fuel for Trojan and the V5H for Sequoyah. [

]

A.2.2 Codes

Framatome's core thermal-hydraulic computer code LYNXT (Reference A-1) is used for licensing-basis analyses for full- and mixed-core configurations. Reference A-1 contains the results of a number of code benchmarks to experimental data and to other NRC-approved computer codes. Two additional benchmarks of LYNXT are included in this appendix to further validate the code's ability to accurately analyze situations where significant amounts of flow are transferred laterally from flow channel to flow channel. Significant lateral flows from fuel assembly to fuel assembly might occur as the core transitions to a new fuel type, in the case of North Anna Units 1 and 2 from the North Anna Improved Fuel (NAIF) to the Advanced Mark-BW.

The InterBundle Diversion Crossflow (IBDCF) tests were unheated crossflow tests based on the Framatome 0.430-inch Mark-B fuel 15x15 design. The fuel assemblies modeled in the IBDCF tests were two 8x15 arrays of fuel pins placed side-by-side. A schematic of the IBDCF test configuration is contained in Figure A-5. These tests were conducted with inlet temperatures and system pressures ranging from [ ]. The axial velocity ratio (V_R) is used to compare the experimental and LYNXT results. V_R is defined as follows:

$$V_R = \frac{(\text{Assembly 1 axial velocity})}{(\text{Assembly 2 axial velocity})} \qquad \text{(A-1)}$$

where "Assembly 1" is the inlet flow-starved fuel assembly. The range of V_R values tested was [ ]. Chapter 6 in Reference A-1 contains comparisons of LYNXT and the experimental data for V_R values of 0.75, 0.90, and 0.95. Appendix I of Reference A-1 contains the results for V_R of 0.0 (no inlet flow into one of the half fuel assembly arrays). This V_R represents a larger flow mismatch (full flow in one bundle, no flow in the other) than is encountered in Condition I and II events in any mixed-core. Figure A-6 shows a comparison of the axial velocity ratios for LYNXT and the experimental data for the IBDCF test with a V_R of 0.0 (same as Figure 4-4 in Reference A-1's Appendix I). The maximum difference between LYNXT and the experimental results is [ ]. This is an excellent comparison of LYNXT and the IBDCF test data and demonstrates LYNXT's ability to properly calculate the axial flow fields for large axial flow mismatches. The axial flow differences in the IBDCF tests are much larger than expected in North Anna Units 1 and 2 between a NAIF and an Advanced Mark-BW.

Additional tests were performed at the Commissariat à l'Energie Atomique's (CEA) Centre d'Etudes Nucléaires de Cadarache. The tests, called the Marignan tests, measured the axial flow profiles in two dissimilar fuel assemblies, one with mid-span mixing grids (MSMGs) and the other without MSMGs. The tests were [

] Figure A-7 contains a schematic of the two fuel assemblies in the Marignan test facility. Figure A-8 shows the region over which detailed LDV measurements were taken. Figure A-9 compares the LYNXT and experimental axial velocity profiles [ ] as a function of the distance from the bundle interface. The distance from the bundle interface is the "X" variable in Figure A-8. Figure A-10 shows a comparison of the axial velocity profiles in the fuel assembly with and without the MSMG for both LYNXT and the experimental results. The axial velocity profiles across the two fuel assemblies upstream of the third MSMG are within approximately [ ]. The Figure A-9 axial velocity profiles for LYNXT and the Marignan tests are within [ ], on the average.

The IBDCF and Marignan tests as well as the comparisons in Reference A-1 demonstrate that LYNXT accurately predicts the hydraulic behavior of significantly different adjacent flow profiles (resulting from different inlet flows or from different fuel designs). Thus, LYNXT is capable of predicting the hydraulic behavior of mixed-core conditions. LYNXT can also analyze the less hydraulically challenging situation where similar fuel assembly types are inserted into a reload cycle, as has been demonstrated for Catawba, McGuire, Trojan, and Sequoyah.

A.3 Overview of Mixed-Core Methodology Process

Four different thermal-hydraulic parameters are evaluated during any transition from one fuel type to another. These are DNBR, pressure drop, hydraulic loads, and span-average crossflow velocity. All of these parameters are evaluated using LYNXT.

This section will provide an overview of the process used to calculate the various thermal-hydraulic parameters.

A.3.1 DNBR Mixed-Core Methodology Process

For the DNB compatibility evaluations, the design criterion is assessed by comparing the calculated DNB ratios (DNBR) to critical heat flux (CHF) correlation design limits, using either deterministic or statistical techniques. In deterministic analyses the random uncertainties on the data modeled in LYNXT are conservatively compounded. As an example, the maximum error on the inlet temperature is used in conjunction with the maximum uncertainty of the radial peaking, etc. This is a conservative analysis process. For statistical DNB analysis, some of these uncertainties are statistically combined in the determination of a statistical design limit (SDL). DNB margin is added to the SDL to define an analysis limit, called the Thermal Design Limit (TDL). The retained thermal margin (RTM) made available by using the TDL is defined as follows:

$$\text{Retained Thermal Margin, percent} = \left(\frac{\text{TDL} - \text{SDL}}{\text{TDL}}\right) * 100 \text{ percent} \qquad \text{(A-2)}$$

The RTM is used to provide flexibility in the fuel cycle design by accommodating cycle-specific DNB effects such as the mixed-core penalty associated with transitioning from the resident fuel type to the new fuel type or SCD boundary condition uncertainties greater than those used in the development of the SDL.

The transition core penalty (TCP) [ ] is determined from the LYNXT analyses and is defined as follows:

$$\mathrm{TCP} = \frac{[(\mathrm{Transition - core\ MDNBR}) - (\mathrm{Full - core\ MDNBR})]}{[\mathrm{Full - core\ MDNBR}]} \qquad \text{(A-3)}$$

where the "Transition-core MDNBR" is the minimum DNBR (MDNBR) obtained from the specific loading pattern or core configuration and boundary condition being evaluated and "Full-core MDNBR" is the full core of the resident or new fuel type. A reference to full-core indicates a model containing all the same fuel type and not necessarily a model of every fuel assembly location in the core.

The level of detail in the mixed-core modeling is dependent on the hydraulic differences between the resident fuel (or fuels) and the fuel being introduced. [

] Typically LYNXT models of the type shown in Figure A-3 are used. The Figure A-3 model is for a 157-fuel assembly plant. For a 177- (15x15 Framatome fuel assembly) and 193-fuel assembly (17x17 mixing vane (MVG) fuel assembly) plant the mixed-core LYNXT models have [ ] channels, respectively.

The process for performing a mixed-core DNBR analysis is as follows:

(1) Determine a range of boundary conditions (core power level, core flow, inlet temperature, system pressure, and radial and axial peaking) that covers the expected range of core thermal-hydraulic conditions. These conditions are selected to encompass a range of steady-state and transient operation.

(2) Determine the MDNBRs for each set of boundary conditions in step 1 using a [ ]. A [ ] consists of [ ]. In generating [ ] the lines of natural symmetry are used in the analyses, as long as the boundary conditions support the symmetry. Since the MDNBRs determined in this step form the basis of the Equation A-3 TCP calculations, [

]

(3) MDNBRs are calculated using the boundary conditions from step 1 for each of the various core configurations evaluated for the mixed-core. These MDNBRs are used to determine the TCPs using Equation A-3. [

] This approach is generally conservative [

]

For the North Anna application the [ ] was used.

Steps 2 and 3 are repeated for each different fuel type in the mixed-core serving as the PLB.

A.3.2 Pressure Drop, Hydraulic Loads, and Crossflow Velocity Transition-Core Methodology Process

A single LYNXT model is used for the pressure drop, hydraulic loads, and crossflow velocity transition-core analyses. A typical LYNXT model for the 157-fuel assembly mixed-core analyses is shown in Figure A-4. This model consists of [ ]. For the 177- and 193-fuel assembly cores (representative of 15x15 B&W and 17x17 Westinghouse 4-loop plants) the LYNXT models would consist of [ ], respectively.

The boundary conditions used to determine the pressure drops, hydraulic loads, and crossflow velocities cover the range of reactor operation, including low temperature isothermal operation with all the reactor coolant pumps operating. The pressure drop, hydraulic loads, and span-average crossflow velocities are provided to mechanical analysis for fuel assembly hold-down margin, gripping forces on fuel assembly components, and flow-induced vibration calculations.

The various core configurations examined are full-cores of the various fuel types and limiting configurations of the potential mixed-cores. Typically the limiting configuration consists of a [

]

A.4 Summary and Conclusions

The thermal-hydraulic compatibility of any new fuel design inserted on a reload basis must be established relative to the resident fuel. This compatibility must extend to protection for DNBR, hydraulic loads, and crossflow velocities. The Framatome mixed-core methodology as described herein provides a systematic and conservative approach to these evaluations.

A.5 References

A-1 Framatome Topical Report BAW-10156-A, "LYNXT: Core Transient Thermal-Hydraulic Program," Revision 1, August 1993.

Figure A-1 - North Anna 12-channel one-eighth core LYNXT model.

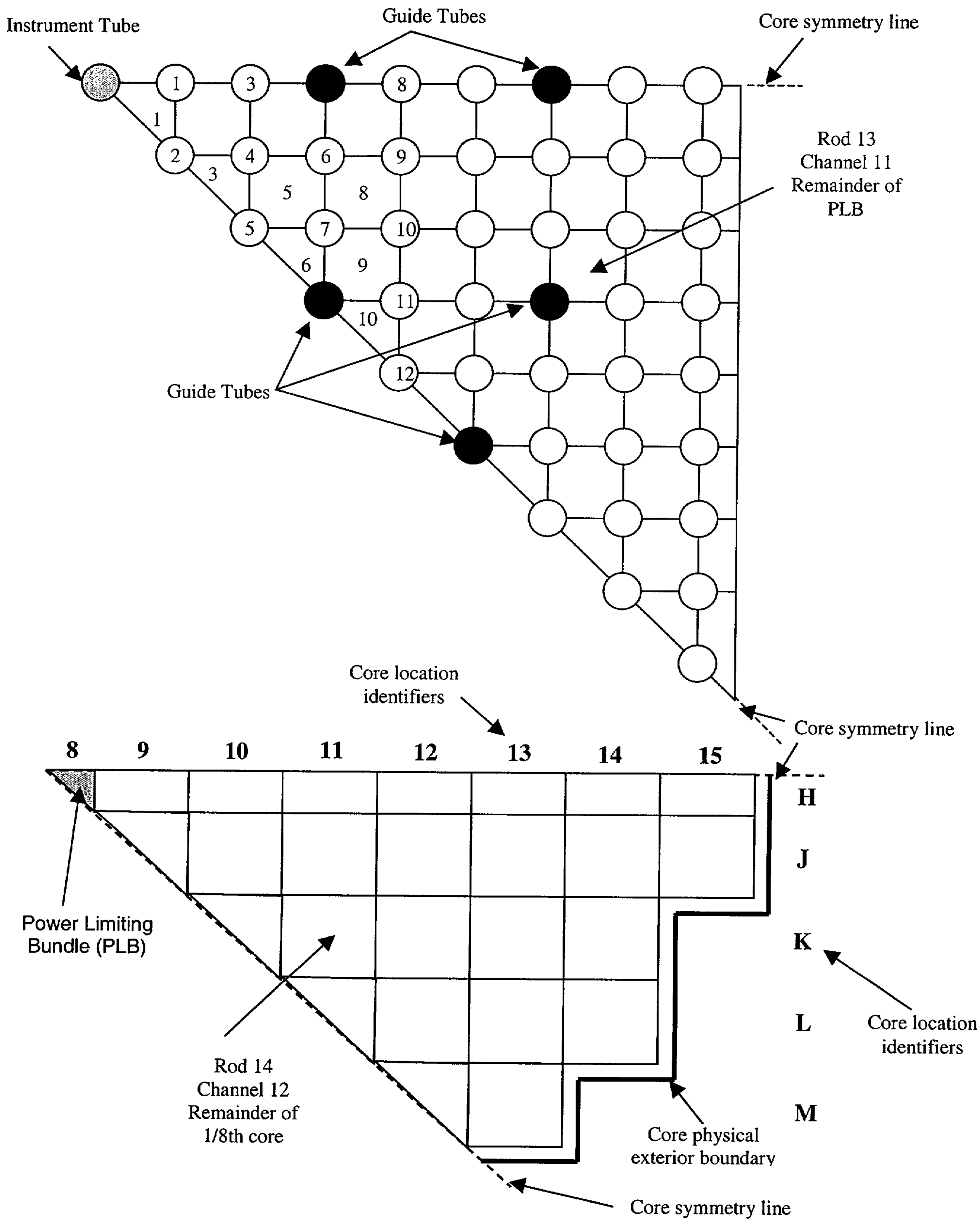


Figure A-2 - North Anna [ ] LYNXT model.

[ ]

Figure A-3 - North Anna [ ] LYNXT model.

Figure A-4 - North Anna [ ] LYNXT model.

Figure A-5 - InterBundle Diversion Crossflow Test Schematic

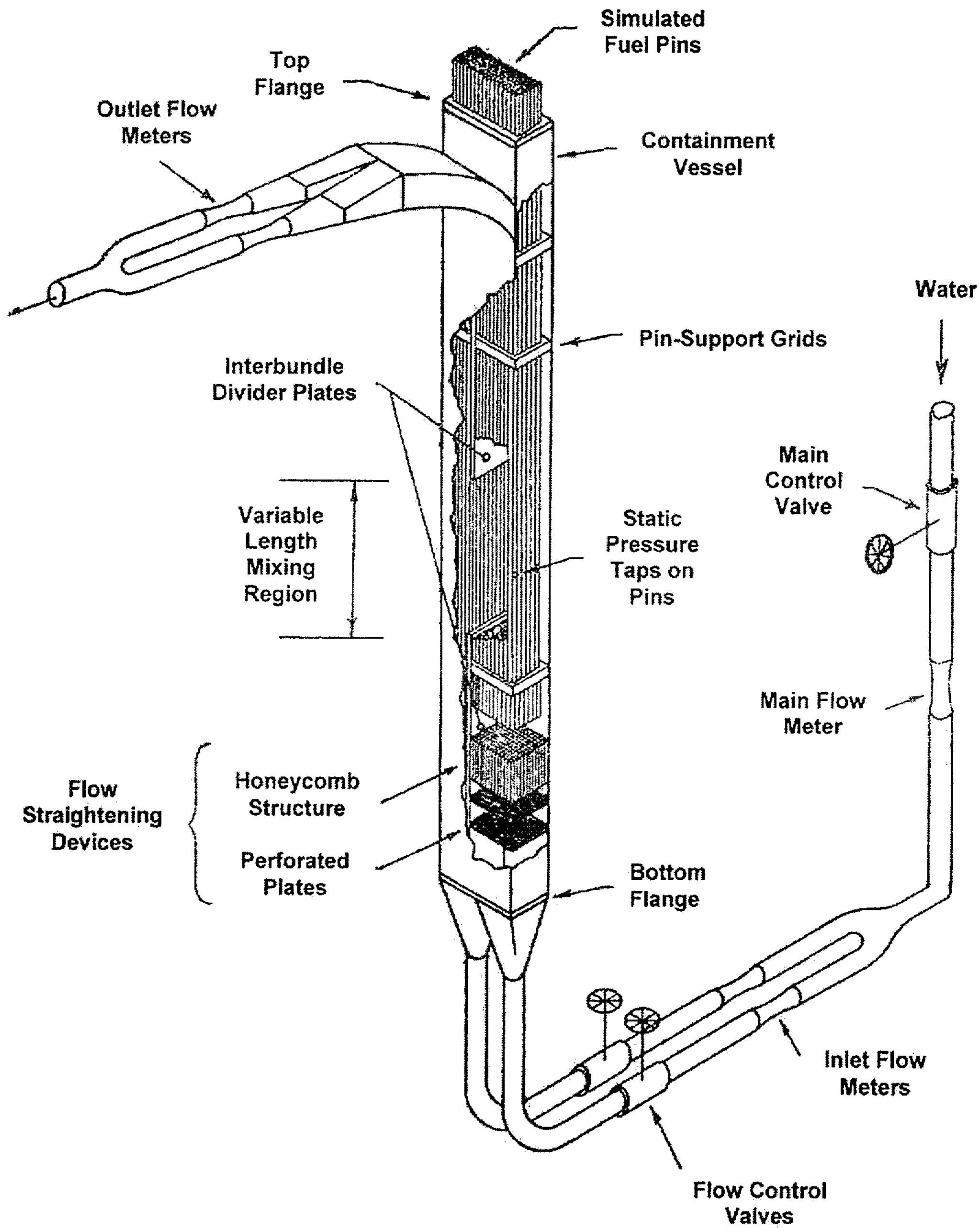
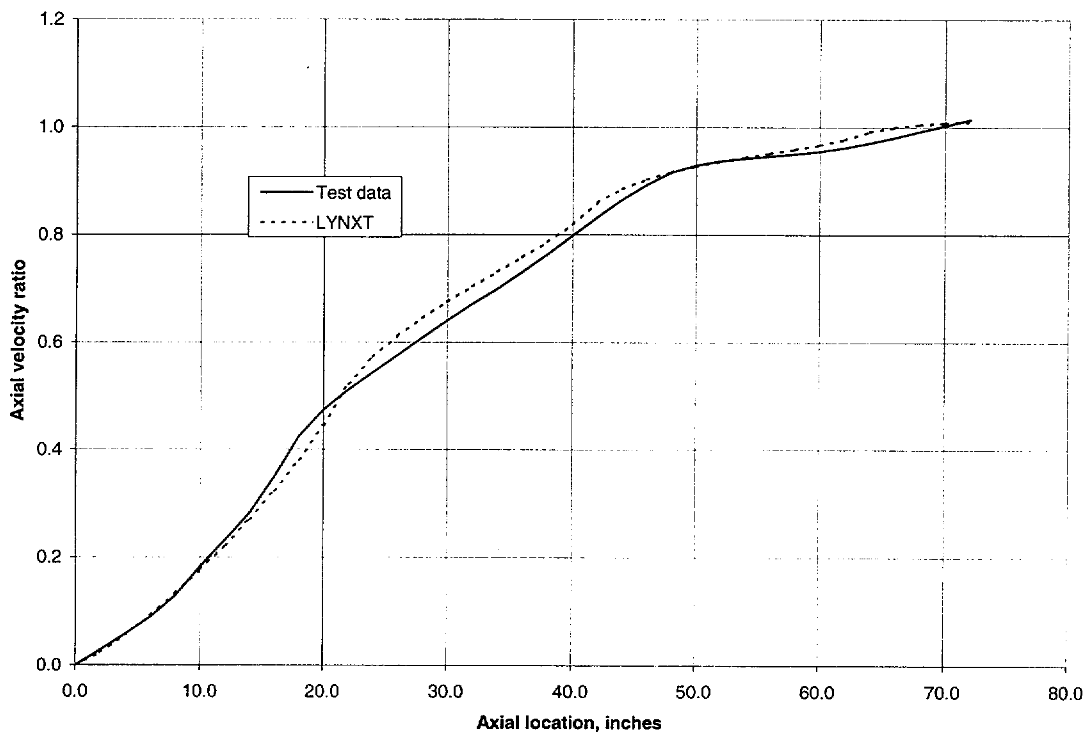


Figure A-6 - IBDCF Experimental and LYNXT Axial Velocity Ratios.



This is the IBDCF test with a velocity ratio, V_R ((Assembly 1 axial velocity)/(Assembly 2 axial velocity)), at the inlet of 0. The maximum difference [

]

Figure A-7 - Marginan Fuel Assembly Configuration.

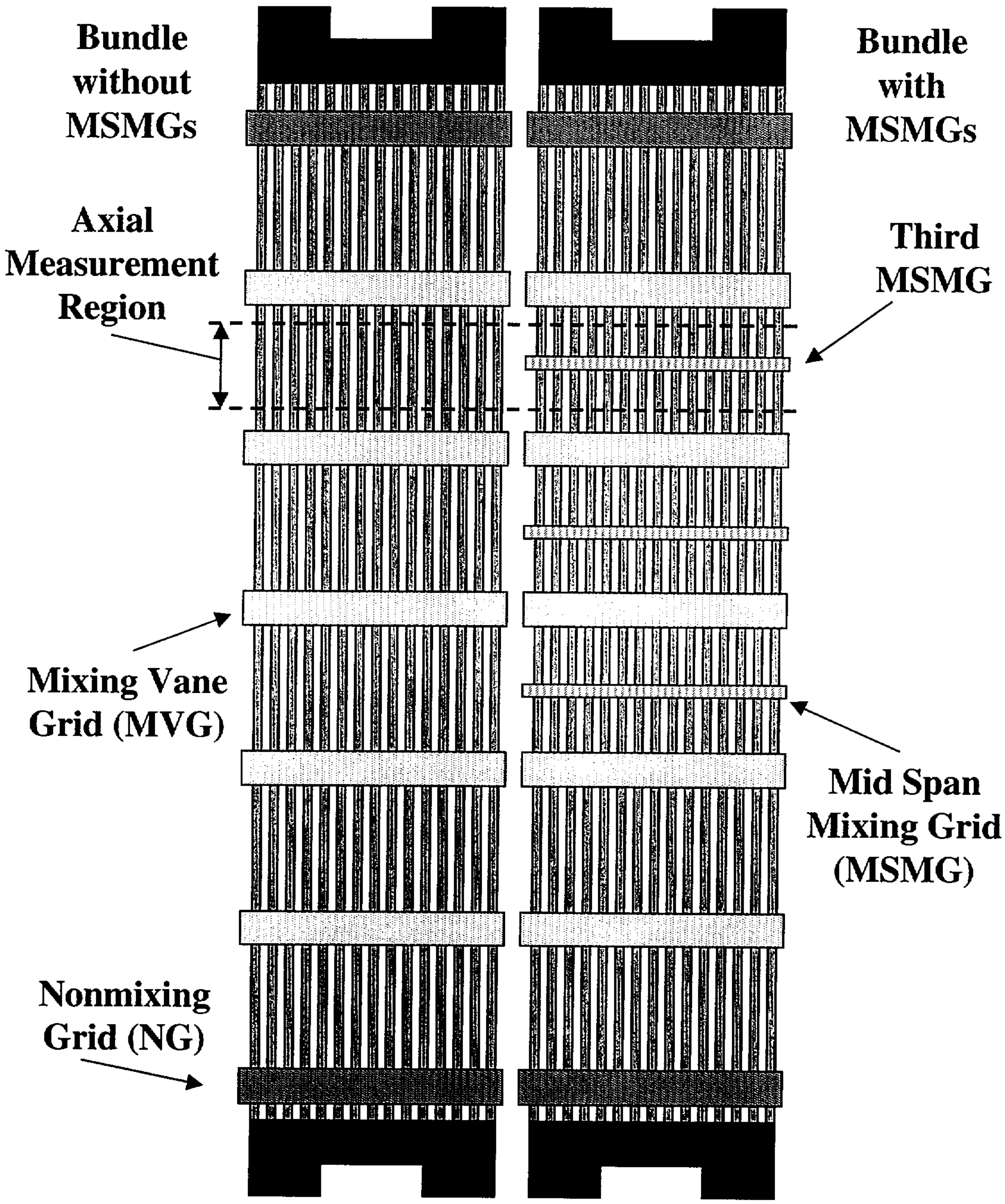


Figure A-8 - Marignan Radial Cross Section Schematic.

Figure A-9 - Marignan and LYNXT Axial Velocities Across The Test Section.

Figure A-10 - Marignan and LYNXT Axial Velocity Profiles