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CENPD-287-NP-A

Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors

ABB Combustion Engineering Nuclear Operations

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CENPD-287-NP-A

Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors

July 1996

ABB Combustion Engineering Nuclear Operations

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CENPD-287-NP-A REPORT

Part I

NRC Acceptance Letter, Safety Evaluation Report (SER), and Technical Evaluation Report (TER)

ABB Combustion Engineering Nuclear Operations

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 17, 1996

Mr. D. B. Ebeling-Koning, Manager
Licensing and Safety Analysis
ABB CENO Fuel Operations
P. O. Box 500
1000 Prospect Hill Road
Windsor, CT 06095-0500

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT CENPD-287-P, "FUEL ASSEMBLY MECHANICAL DESIGN METHODOLOGY FOR BOILING WATER REACTORS," (TAC NO. M90189)

Dear Mr. Ebeling-Koning:

We have reviewed the subject topical report of June 1994, and your responses of August 1995 and February 1996, to our requests for additional information. On the basis of our review, we conclude that the BWR fuel mechanical design methodology documented in CENPD-287-P are acceptable for licensing applications. Enclosed is our safety evaluation report (SER), which details the basis for and limitations of our approval.

The staff will not repeat its review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented applies to the specific plant involved. NRC acceptance applies only to the matters described in the report. In accordance with procedures established in NUREG-0390, ABB/CE should publish accepted versions of this topical report, proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted versions shall include an "-A" (designated accepted) after the report identification symbol.

Should our acceptance criteria or regulations change, so that our conclusions as to the acceptability of the report are no longer valid, applicants referencing this topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued applicability of the topical report without revision of their respective documentation.

Sincerely,

Robert Jones

Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosure:
CENPD-287-P Evaluation

ENCLOSURE

SAFETY EVALUATION OF ABB/CE
TOPICAL REPORT CENPD-287-P
"FUEL ASSEMBLY MECHANICAL DESIGN METHODOLOGY FOR BOILING WATER REACTORS"

1.0 INTRODUCTION

In a letter of June 17, 1994, from D. B. Ebeling-Koning, ABB Combustion Engineering (ABB/CE), to the U.S. Nuclear Regulatory Commission (NRC), ABB/CE submitted a Topical Report CENPD-287-P, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors," for NRC review.

CENPD-287-P describes a fuel mechanical design methodology that provides thermal, mechanical, and creep collapse analyses for BWR fuel. CENPD-287-P utilizes three computer codes, STAV6.2, VIK-2, and COLLAPS-II, described in CENPD-285-P for fuel mechanical designs and safety analyses. ABB/CE will apply this methodology for reload licensing applications.

The NRC staff was supported in this review by its consultant, Pacific Northwest Laboratory (PNL). The staff has adopted the findings recommended in our consultant's technical evaluation report (TER), which is attached, as described by this safety evaluation report.

2.0 EVALUATION

Based on our consultant PNL recommendations and the staff review of the TER, we agree with the PNL evaluation and conclude that the TER provides adequate technical basis to approve CENPD-287-P.

3.0 CONCLUSIONS

The staff has reviewed the ABB/CE BWR fuel mechanical design methodology described in CENPD-287-P and finds the described mechanical design methodology acceptable for reference in licensing applications for ABB/CE BWR fuel.

However, our approval of this document for reload applications is limited to the following conditions.

(1) Based on the validity of fission gas release (FGR) and corrosion models, and other models, the application of CENPD-287-P is approved to a rod average burnup of 50 GWd/MTU.

(2) The cladding creep model has too strong a dependence on cladding stress that results in a non-conservative estimate of the rod pressure. The staff concludes that the stress exponent of the ABB/CE creep equation model should be limited to 1.5. The uncertainty in the creep model estimated by ABB/CE is too small and should be increased by a factor of 2 for both the creep equation and the creep relationship in STAV6.2.

(3) The ABB/CE BWR FGR model is approved to 40 GWd/MTU rod average, and the PWR FGR model is approved to 50 GWd/MTU rod average in STAV6.2. Thus, the PWR FGR model is the only acceptable model for fission gas release calculation for burnups between 40 and 50 GWd/MTU.

(4) The STAV6.2 code is acceptable for application to urania-gadolinia fuel with gadolinia content up to 8 wt%. The ABB/CE urania-only fission gas diffusion constants should be used for both urania-only and urania-gadolinia fuel rod applications.

(5) The calculation of uniform cladding strain should be the elastic plus inelastic strains due to power increases in normal operations and AOOs.

TECHNICAL EVALUATION REPORT OF THE
TOPICAL REPORT CENPD-287-P, ENTITLED
"FUEL ASSEMBLY MECHANICAL DESIGN
METHODOLOGY FOR BOILING WATER REACTORS"

C. E. Beyer
D. D. Lanning

April 1996

Prepared for Reactor Systems Branch
U.S. Nuclear Regulatory Commission
Washington, DC
under Contract DE-AC06-76RLO 1830
NRC JCN I2009

Pacific Northwest National Laboratory
Richland, Washington 99352

ABBREVIATIONS

ABB/C-E	Asea Brown Boveri/Combustion Engineering, Inc.
AOO	Anticipated Operational Occurrence
BOL	Beginning-of-Life
BWR	Boiling-Water Reactor
CFR	Code of Federal Regulations
ΔCPR	Delta Critical Power Ratio
CRDA	Control Rod Drop Accident
DPH	Design Power History
EOL	End-of-Life
EPRI	Electric Power Research Institute
FGR	Fission Gas Release
GDC	General Design Criteria
JCN	Job Control Number
LOCA	Loss of Coolant Accident
LHGR	Linear Heat Generation Rate
LWR	Light Water Reactor
MCPR	Minimum Critical Power Ratio
NRC	Nuclear Regulatory Commission
PCI	Pellet Cladding Interaction
PCT	Peak Clad Temperature
PIE	Post-Irradiation Examination
PWR	Pressurized-Water Reactor
SAFDL	Specified Acceptable Fuel Design Limit
SRP	Standard Review Plan

TER	Technical Evaluation Report
The Laboratory	Pacific Northwest National Laboratory
TMOL	Thermal Mechanical Operating Limit

1.0 INTRODUCTION

This technical evaluation report (TER) was prepared by Pacific Northwest National Laboratory (referred to as the Laboratory)[1] under U.S. Nuclear Regulatory Commission (NRC) financial identification number JCN I2009. This TER is a review of the methodology by which Asea Brown-Boveri/Combustion Engineering Nuclear Operations (ABB/C-E) will apply its steady-state thermal and mechanical fuel performance codes in the design and licensing of fuel assemblies for boiling-water reactors (BWRs). The methodologies are applied to demonstrate that the assemblies and rods meet their design criteria. These criteria are drawn from the NRC's Standard Review Plan (SRP) Section 4.2 (Reference 1) and hence are related to the general design criteria of 10CFR50 (Reference 2). ABB/C-E's computer codes used for this purpose are:

STAV6.2 calculates fuel rod behavior to verify that thermal and mechanical limits are met, and provides input to fuel reload safety analyses

VIK-2: performs stress analyses of fuel rod cladding

COLLAPS-II: calculates cladding ovalization and performs creep collapse analysis

These codes are described in Reference 3, and the design methodologies are described in Reference 4. In response to NRC requests for additional information (Reference 5) ABB/C-E has provided two supplements to Reference 4 as References 6 and 7.

ABB/C-E has stated in Reference 4 that the application of these codes will be limited to design and licensing analyses for BWR fuel rods, operated up to the specific limits for assembly-average and peak pellet burnup. These limiting values are hereafter referred to as the requested burnup limits.

Summarized in this TER are the Laboratory's examination and evaluation of the proposed general fuel system design criteria (Section 2), the specific fuel assembly and fuel rod design criteria (Section 3), and the analysis methodologies that are used to demonstrate that the fuel assembly and fuel rod design criteria are met (Section 4). Conclusions are stated in Section 5.

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[1]Operated for the U.S. Department of Energy by Battelle Memorial Institute under Contract DE-AC06-76RLO 1830.

2.0 DESCRIPTION AND EVALUATION OF GENERAL DESIGN CRITERIA

The methodologies used and the limits applied to meet general fuel system design criteria are stated in Sections 3.0 and 3.1 of Reference 4. The stress intensities and stress limits are drawn from the ASME code (Reference 8) and are acceptable. The materials properties used in the methodology are listed. These properties must be reviewed for the requested burnup limits in Reference 4 to verify their applicability at these burnup limits.

The ABB/C-E general design criteria are consistent with the specified acceptable fuel design limits (SAFDLs) of General Design Criteria (GDC) 10, the control rod insertability requirements of GDC 27, and the core coolability requirements of GDC 35 (as specified in Reference 2). In order to meet these general design criteria ABB/C-E follows the guidance in Section 4.2 of the SRP for 1) normal operation and anticipated operational occurrences (AOOs), and 2) accidents in regards to fuel coolability control rod insertability, and not underestimate fuel rod failure. The Laboratory concludes that the ABB/C-E General Design Criteria are consistent with the NRC GDC requirements and the guidance in the SRP and are acceptable.

3.0 DESCRIPTION AND EVALUATION OF SPECIFIC FUEL ASSEMBLY AND FUEL ROD DESIGN CRITERIA

This section addresses the SAFDLs for ABB/C-E fuel designs as presented in Sections 3.1, 3.2 and 3.3 of Reference 4 in relation to those SAFDLs referred to by the NRC in the GDCs for nuclear power plants (Reference 2) and in Section 4.2 of the SRP. The purpose of this evaluation is to verify that the ABB/C-E SAFDLs are consistent with GDC 10 for normal operation including AOOs, GDC 21 for control rod insertability, GDC 35 for core coolability, and the guidance in Section 4.2 of the SRP.

ABB/C-E specific criteria for accidents such as for control rod insertability and core coolability are primarily addressed in Section 3.1 of Reference 4 while specific criteria for normal operation (including AOOs) are primarily addressed in Section 3.2 for fuel assembly components and Section 3.3 for fuel rods.

3.1 Specific Criteria for Accidents

The ABB/C-E criteria for accidents including the criteria for cladding ballooning and rupture, cladding mechanical fracturing, violent expulsion (or dispersal) of fuel, excessive fuel enthalpy and fuel assembly structural damage are nearly identical to those provided in the 10CFR50, Paragraph 50.46, Appendix K, and Section 4.2 of the SRP. The ABB/C-E criterion for cladding embrittlement consists of a 2200°F peak cladding temperature (PCT) criterion also used by 10 CFR 50.46, Appendix K. However, the ABB/C-E criteria does not include the 17% oxidation limit required by Appendix K. This is an oversight by ABB/C-E in CENPD-287 and the 17% oxidation criterion has been included in CENPD-300.

Those ABB/C-E accident criteria that are identical to the Code of Federal Regulations and the guidance in Section 4.2 of the SRP will be individually discussed briefly.

The ABB/C-E cladding ballooning and rupture criterion is that unacceptable rupture shall not occur. In addition, Appendix K states that cladding ballooning and rupture not be underestimated. This issue is addressed in the LOCA review of ABB/C-E cladding ballooning and rupture models and is not part of this review.

The ABB/C-E criterion for mechanical fracturing is that external loads due to thermal hydraulics or earthquake must not result in fuel rod fracturing or unacceptable distortions in the assembly. This is identical to the requirements of 10 CFR 50 and Section 4.2 of the SRP and, therefore, is acceptable.

The ABB/C-E criterion for violent fuel expulsion is to limit the radially average peak fuel enthalpy to less than 280 calorie/g during a Control Rod Drop Accident (CRDA). This is identical to the guidance in Section 4.2 of the SRP and Regulatory Guide 1.77 (Reference 9). It is noted that the NRC staff is currently reviewing the 280 calorie/g limit and it may be decreased to a lower limit at high burnup levels. However, the current enthalpy limit remains valid at this time.

The ABB/C-E criterion for excessive fuel enthalpy is to assume failure when radially averaged fuel enthalpy exceeds 170 calorie/g at any axial location or when fuel cladding dryout occurs. It is noted that the NRC staff is currently reviewing the 170 calorie/g limit and this may be decreased to a lower enthalpy limit at high burnup levels. However, the current enthalpy limit remains valid at this time. The use of cladding dryout for fuel failure is acceptable as long as ABB/C-E assumes that cladding dryout occurs when the fuel rod exceeds their thermal margin limit [minimum critical power ratio (MCPR)].

The ABB/C-E criterion for fuel assembly structural damage is covered under their criterion for mechanical fracturing (no unacceptable fuel assembly distortions) and the ABB/C-E criterion that large distortion of failure of the spacer grids does not occur for a seismic-LOCA event. These general criteria are acceptable.

3.2 Specific Criteria for Fuel Assembly Components

ABB/C-E has identified nine specific criteria for fuel assembly components during normal operation and AOOs. Each of these nine criteria are briefly described below.

3.2.1 Mechanical Compatibility

Mechanical compatibility with core components and internals, e.g., control rods and other fuel designs, is required such that components are not damaged and can perform their intended functional requirements.

3.2.2 Geometry Changes (Axial Growth)

Geometry changes during the life of the fuel assembly components must not interfere or impair the performance of other components. This includes sufficient clearances between the fuel rods and upper tie-plate to prevent an interference fit or disengagement from the upper-tie-plate.

3.2.3 Handling and Shipping Loads

Assembly components shall be able to handle handling and shipping loads without damage.

3.2.4 Hydraulic Lift

The maximum hydraulic lift loads shall not exceed the hold down capability of the fuel assembly.

3.2.5 Stress

Assembly components must not exceed the design stress limits in accordance with the ASME Boiler and Pressure Vessel Code (Reference 8). The specific stress limits are provided in Section 4.0, pages 31 to 33 of Reference 4.

3.2.6 Fatigue

Assembly component fatigue failure shall not occur.

3.2.7 Fretting Wear

Assembly fretting wear must be accounted for in evaluating stress and fatigue limits. Fuel rod failure due to fretting shall not occur in an environment free of foreign material (debris).

3.2.8 Corrosion

Corrosion and crud of assembly components must be accounted for in evaluating functionality, stress, dimensional changes, and thermal hydraulics.

3.2.9 Hydriding

Hydriding of Zircaloy components shall not result in unacceptable strength losses. ABB/C-E's limit on hydriding is provided in Section 4.2.9 of Reference 4 and is the same as that stated for cladding hydriding (see Section 3.3.4 of this TER) which is based on an average concentration. The Laboratory concludes that ABB/C-E's hydriding limit is unacceptably high. However, hydriding is satisfactorily addressed with criterion and limits for corrosion. The level of hydriding is directly related to the level of corrosion and examination of the level of corrosion in the Zircaloy components and appears to be acceptable (see Section 4.1.8). Therefore, the issue of hydriding is acceptably addressed by the criterion for corrosion.

3.2.10 Conclusions

These criteria for assembly components are found to be consistent with the guidance in Section 4.2 of the SRP and, therefore, are acceptable.

3.3 Specific Criteria for Fuel Rods

ABB/C-E has identified ten criteria to prevent fuel rod damage or failure during normal operation and AOOs. Each of these ten criteria are briefly described below.

3.3.1 Internal Rod Pressure

Rod internal gas pressure shall not exceed a value which would cause the outward cladding creep rate to exceed the fuel swelling rate. This criterion is different from the guidance in Section 4.2 of the SRP where rod pressures are requested to remain below the coolant system pressure. However, NRC has recognized that a rod pressure limit of less than system pressure is conservative and has allowed fuel vendors to exceed coolant system pressure with appropriate justification and limits. The ABB/C-E criterion to limit rod pressures below a value that would cause the fuel cladding outward creep rate to exceed the fuel swelling rate is conservative and is also consistent with previous rod pressure criteria approved by the NRC. Therefore, the Laboratory concludes that this criterion is acceptable for application to the ABB/C-E SVEA-96 and 100 fuel design.

3.3.2 Stress

The ABB/C-E criteria for fuel rod cladding stresses are established limits in accordance with the ASME Boiler and Pressure Vessel Code (Reference 8). This is consistent with Section 4.2 of the SRP and, therefore, is acceptable for application to the SVEA-96 and 100 fuel designs.

3.3.3 Strain

The ABB/C-E criterion for fuel cladding strain is that uniform strain shall not exceed 1% excluding the effects of steady-state creepdown and irradiation growth. In addition, a second ABB/C-E criterion is that maximum permanent end-of-life strain shall be less than a proprietary value including cladding creepdown and irradiation growth. The 1% uniform strain criterion is consistent with Section 4.2 of the SRP. It should be noted that recent high burnup pressurized-water reactor (PWR) cladding with uniform corrosion on the order of 100 μm of thickness have shown that Zircaloy cladding fails at values less than 1% uniform strain (References 10 through 15). Therefore, the NRC is reassessing the 1% uniform strain and corrosion criteria for high burnup fuel rods within Section 4.2 of the SRP. However, the 1% uniform strain and corrosion criteria remain valid and are currently considered acceptable.

3.3.4 Cladding Hydriding

The ABB/C-E criterion for cladding hydriding is to prevent premature failure due to either internal hydriding or waterside corrosion. Internal hydriding is prevented by controlling the moisture and total hydrogen in the fuel pellets. The ABB/C-E controls on fuel pellet moisture in fuel rods are more conservative than the requirements in Section 4.2 of the SRP and, therefore, are acceptable.

The criterion and limit on uniform hydrogen content of the cladding from waterside corrosion is provided in Section 4.3.4, page 91 of Reference 4. This limit on uniform hydrogen content may be too high for high burnup cladding based on recent mechanical test data (References 10 through 15) that shows less than 1% cladding strain capability at this uniform hydrogen content. However, the ABB/C-E criterion and limit on waterside corrosion appears to be more conservative with respect to limiting Zircaloy hydrogen content (see discussion on waterside corrosion criterion in section below). The Laboratory concludes that the ABB/C-E criterion on maximum waterside corrosion will adequately control the level of cladding hydrogen due to waterside corrosion and the issue has been satisfactorily addressed.

3.3.5 Cladding Corrosion

The ABB/C-E criterion and limit on maximum cladding waterside corrosion is provided in Section 4.3.5, page 95 of Reference 4. This corrosion limit provides a maximum oxide thickness limit at any axial location on the fuel rod. The Laboratory concludes that this corrosion limit is acceptable. This limit on maximum corrosion is judged to also satisfactorily limit the level of hydrogen.

3.3.6 Cladding Collapse

The ABB/C-E criterion for cladding collapse is that cladding collapse will not occur during the life of the fuel rod. This is consistent with Section 4.2 of the SRP and, therefore, is acceptable.

3.3.7 Fatigue

The ABB/C-E criterion for fatigue damage is that fatigue damage shall not occur taking into account the effects of cladding corrosion. This is consistent with Section 4.2 of the SRP and, therefore, is acceptable.

3.3.8 Cladding Overheating

The ABB/C-E criterion to prevent cladding overheating is to maintain an adequate margin to boiling transition in terms of MCPR. This is consistent with Section 4.2 of the SRP and, therefore, is acceptable.

3.3.9 Fuel Overheating

The ABB/C-E criterion for maximum fuel centerline temperature is to keep temperatures below the melting temperature of the fuel. The ABB/C-E fuel melting limit is burnup and gadolinia dependent as discussed in Section 4.3.9, page 109 of Reference 4. In addition, in response to an NRC question on the lack of a burnup dependence on fuel thermal conductivity in STAV6.2, ABB/C-E has elected to provide an additional burnup dependent penalty on the fuel melting temperature (Reference 7). This will be discussed further in Section 4.3.9 of this TER that discusses the methodology utilized by ABB/C-E to demonstrate compliance to the criterion of no fuel melting. The Laboratory concludes that the ABB/C-E criterion and limit on fuel melting are acceptable.

3.3.10 Fuel Rod Bowing

The ABB/C-E criterion for fuel rod bowing is that excessive bowing shall be precluded for the fuel assembly life and any significant impact shall be accounted for in the thermal and mechanical evaluation. The ABB/C-E use of the words "significant impact shall be accounted for" is not very specific. In addition, in Section 4.3.10, pages 113-115 of Reference 4, ABB/C-E appears to claim that rod bowing is not significant as long as rod bowing does not lead to contact with other rods or the fuel channel. This condition of allowing rod bow without any penalty on thermal margins (MCPR) has not previously been approved by the NRC for other vendors in the U.S. In addition, the NRC has either required a penalty on MCPR or required reporting of rod bow when rod-to-rod spacings were decreased by 50% based on a 95/95% confidence bound on rod bowing data. The issue of rod bowing will be discussed further in Section 4.3.10 of this TER.

3.3.11 Pellet Cladding Interaction

ABB/C-E has no criterion for pellet cladding interaction (PCI). The NRC identifies PCI as a failure mechanism in Section 4.2 of the SRP but does not offer any specific criteria to prevent PCI other than the 1% uniform strain and no fuel melting criteria. These criteria have already been addressed satisfactorily by ABB/C-E and, therefore, the Laboratory concludes that ABB/C-E has satisfactorily addressed the issue of PCI failures. The ABB/C-E methodology for preventing PCI failure is discussed in Section 4.3.11 of this TER.

4.0 DESCRIPTION AND REVIEW OF DESIGN METHODOLOGY FOR ASSEMBLIES AND FUEL RODS

Section 4.0 of Reference 4 provides the ABB/C-E methodology for evaluation of fuel assembly and fuel rod integrity for normal operation and AOOs relative to the criteria given in Section 3 of Reference 4 and this TER. The methodology for initializing transients and accidents as a result of steady-state operation is also discussed in this section of the TER. The methodology for evaluation of accidents is discussed in other ABB/C-E topical reports and will not be reviewed in this TER. Therefore, the purpose of Section 4.0 in this TER is to review the ABB/C-E methodology for evaluating fuel assembly and fuel rod integrity for normal operation and AOOs and steady-state initialization of transients and accidents to the requirements in Section 4.2 of the SRP. Those methodologies used to demonstrate compliance to the fuel assembly criteria and integrity are reviewed in Section 4.1 of this TER while those methodologies used to evaluate compliance to the fuel rod criteria and integrity are reviewed in Section 4.2 of this TER. The methodologies used as input to initialize the transients and accidents as a result of steady-state operation are reviewed in Section 4.3 of this TER.

4.1 Description and Review of Design Methodology for Fuel Assemblies

The ABB/C-E analysis methodologies used to demonstrate compliance to the ABB/C-E fuel assembly criteria are discussed in this section of this TER.

4.1.1 Stress and Strain

The ABB/C-E methodology for calculating fuel assembly stress intensities and the resultant strains are based on the ASME Boiler and Pressure Vessel Code (Reference 8). The maximum stress intensities are calculated using standard engineering relationships provided in Section 4.0, page 33 of Reference 4 and a standard finite element model. Fuel assembly stress analyses are performed for transportation and handling loads to demonstrate that the assembly components will not be damaged during these operations. In addition, the stresses in the spacer springs, external compression springs and the channel walls are evaluated for normal operation and AOOs. Therefore, the Laboratory concludes that these analysis methods are acceptable.

4.1.2 Strain Fatigue

The only component in the assembly, other than the fuel rods, that experiences stress cycling that can result in strain fatigue is the fuel channel. ABB/C-E uses standard stress analysis methodology evaluated in terms of Reference 8 and the Zircaloy fatigue design curve of O'Donnel and Langer (Reference 16) to calculate fatigue usage factors. The design fatigue curve includes the more conservative of a factor of two on stress amplitude or 20 on the number of cycles. The calculated cumulative usage factor must be less than 1.0 for the design life of the fuel channel. For these calculations ABB/C-E assumes a conservative number of power cycles and maximum pressure differential cycles across the channel wall that bound those possible during normal operation. The Laboratory concludes that this analysis methodology is conservative and, therefore, acceptable.

4.1.3 Channel Bulge

The channel walls experience creep and elastic deflections inward and outward depending on the channel location. Of primary concern are the outward deflections because the outward movement can interfere with control rod movement. ABB/C-E has presented calculations that demonstrate satisfactory outward wall deflections out to an assembly-average burnup of 55 GWd/MTU. ABB/C-E has also provided channel creep deformation data with assembly-average burnups up to 48 GWd/MTU from the SVEA-64 fuel assembly that has a similar channel design to the SVEA-96/100 design. The Laboratory concludes that this analysis methodology is acceptable up to a rod-average burnup of 50 GWd/MTU, which is consistent with the conclusions from the NRC review of CENPD-285-P.

4.1.4 Channel Bow

The primary concerns with channel bow are interference with control rod insertion and a reduction in neighboring fuel rod thermal margin. The impact of channel bow on MCPR is addressed in the review of thermal hydraulics and will not be discussed further in this TER.

ABB/C-E has provided channel bow data from SVEA fuel in a Nordic reactor with a symmetric lattice in Figure 4.2.2 of Reference 4 with assembly average burnups up to 43 GWd/MTU and in an asymmetric lattice in Figure A4-1 of Reference 7 with assembly average burnups up to 43 GWd/MTU. ABB/C-E has not explicitly provided an analysis methodology other than stating that data uncertainties are included. Therefore, the Laboratory recommends that the upper 95/95 confidence level of the channel bow data be used in the analysis of control rod clearances. The Laboratory performed a preliminary analysis of control rod clearances for a symmetric lattice assuming: 1) worst case as-fabricated tolerances that minimize the control rod clearance between the outer channels, 2) assuming maximum channel bulging due to creep and elastic deflection, and 3) the upper 95/95 confidence level of the channel bow data at a burnup level of 43 GWd/MTU for both channels on either side of the control blade. The analysis gave results with a very small clearance (near zero). ABB/C-E has stated in Reference 4 that an interference fit of 4.5 mm is required to achieve two-thirds of the force available to withdraw a control rod. The maximum interference fit at a maximum rod-average burnup of 50 GWd/MTU will be less than 0.2 mm and due to the flexibility of ABB/C-E channels this is acceptable.

The Laboratory concludes that the ABB/C-E analysis methodology for channel bowing is acceptable up to a rod-average burnup of 50 GWd/MTU, which is consistent with the conclusions from the NRC review of CENPD-285-P.

4.1.5 Axial Growth

The three components that experience axial growth are the fuel channel, the tie rods (sub-assembly), and the fuel rods. Each of these components experience axial growth at different rates that can cause the following problems: 1) interference between the fuel rod end cap extension (depends on fuel rod growth) and the frame of the assembly handle (depends on channel growth), 2) interference between the top end plug shoulder, the compression springs and the top tie plate (disengagement of the fuel rod end cap from the top tie plate can also occur if fuel rod growth is significantly less than the tie rod growth), and 3) if the channel springs and control rod gap spacer buttons of the SVEA-96/100 design do not overlap with those of non-SVEA assemblies, due to differential growth of the two different channels, this can allow interference with the control blades.

ABB/C-E has provided axial growth data for their SVEA fuel channels and fuel rods as a function of assembly average burnups up to ~48 GWd/MTU (Figures 4.2.3 and 4.2.8 in Reference 4). As a result of responses to NRC questions (Reference 5) they have also provided the top end plug shoulder to upper tie plate gap data as a function of assembly-average burnup up to 48 GWd/MTU in Figure A6-3 and further channel growth data for asymmetric lattices in Figure A5-1 both from Reference 7. The channel growth data in Figure A5-1, includes data from 8x8 channels (without the water cross) up to an assembly-average burnup of 67 GWd/MTU. The 8x8 channel growth data appear to be consistent with the SVEA water cross designs and, therefore, the 8x8 data is applicable to the SVEA channels.

ABB/C-E has stated that they use worst case tolerances for determining channel and subbundle growth; however, worst case tolerances are not used in the end plug shoulder to upper tie plate gap evaluation because they are already implicitly accounted for in the measured gap data. The Laboratory concludes that their use of tolerances is acceptable.

The ABB/C-E methodology for calculating maximum or minimum growth of these components for conservative licensing analyses appears to be based on selecting a value of growth that bounds the upper or lower bound of the scatter in the data. The Laboratory has determined that the bounding growth values used by ABB/C-E are equal to or conservative in relation to the 95/95 confidence bounds of the ABB/C-E data up to the maximum burnup of the data (an assembly-average burnup of 48 GWd/MTU) and, therefore, are acceptable. However, due to the lack of data the Laboratory does not recommend approval of the ABB/C-E methodology for growth at the burnup level requested by ABB/C-E but concludes that the methodology is acceptable up to a rod-average burnup level of 50 GWd/MTU, which is consistent with the conclusions from the NRC review of CENPD-285-P.

4.1.6 Hydraulic Assembly Liftoff

Fuel assembly liftoff is not permitted during normal operation and AOOs because this can adversely impact core physics and thermal hydraulics. The ABB/C-E analysis methodology for assembly liftoff is to provide conservatively bounding values for those parameters such as core flow, power, inlet enthalpy, axial and radial peaking factors, and cladding crud that impact fuel assembly liftoff. These values are input into an ABB/C-E thermal hydraulics code to calculate maximum hydraulic lift loads. The ABB/C-E thermal hydraulics code is reviewed in a separate submittal.

4.1.7 Fretting Wear

There are no analytical models for predicting fretting wear for new assembly designs. The ABB/C-E methodology for avoiding fretting wear consists of three features:

1) Design features are retained from previous designs that have demonstrated acceptable in-reactor fretting wear performance.

2) Full scale hydraulic testing of new fuel assembly designs that have design modifications that could affect hydraulic characteristics and fretting wear.

3) New fuel designs are examined for fretting wear during post-irradiation examinations (PIE) after in-reactor operation.

The Laboratory has reviewed this methodology and concludes that it is consistent with Section 4.2 of the SRP and, therefore, acceptable.

4.1.8 Corrosion and Hydriding

The ABB/C-E methodology for minimizing both corrosion and hydriding is discussed as one subject because, as noted earlier in Sections 3.2.9 and 3.3.4, the level of Zircaloy hydriding is a direct result of the level of waterside corrosion and is a function of time in reactor and, to some extent fuel burnup.

The ABB/C-E methodology for minimizing corrosion in assembly components has three components:

1) Use materials and fabrication processes for which corrosion performance has been satisfactorily demonstrated based on in-reactor experience.

2) Perform out-of-reactor corrosion tests on new or improved materials.

3) Examine corrosion in new assembly designs. Corrosion levels should also be measured when they become significant to confirm the operability of the component and the corrosion correlations.

ABB/C-E has provided maximum measured corrosion data for their older channel materials and their newer channel materials in Figure 2.15a of Reference 4. However, these data only extend to a maximum assembly-average burnup of approximately 50 GWd/MTU for the older Zircaloy material and to 40 GWd/MTU (assembly-average) for the newer Zircaloy material used in the SVEA 96/100 design. The older material shows a significant amount of corrosion at the burnup of 50 GWd/MTU (assembly-average). However, the corrosion data from the newer Zircaloy channel material do indicate that corrosion will be satisfactory up to an assembly-average burnup of 48 GWd/MTU.

Based on the ABB/C-E corrosion data and the conclusions from the NRC review of CENPD-285-P, the Laboratory concludes that corrosion is acceptable up to rod-average burnups of 50 GWd/MTU.

4.2 Description and Review of Design Methodology for Fuel Rods

This section reviews the ABB/C-E analysis methodologies for demonstrating that ABB/C-E SVEA designs demonstrate compliance with the Criteria for Fuel Rods discussed in Section 3.3 of this report. ABB/C-E utilizes the STAV6.2, VIK-2, and COLLAPS-II computer codes to demonstrate compliance with some of these criteria.

The STAV6.2 code is used for analyses of rod internal pressure, cladding strain, strain fatigue and fuel temperatures. The VIK-2 code is used to calculate fuel rod stresses and the COLLAPS-II code is used to evaluate cladding collapse.

Many of these analyses are dependent on the fuel rod linear heat generation rate (LHGR) and, therefore, dependent on the power history of the fuel rod during its irradiation life. ABB/C-E has established two methodologies for determining the input power history for these analyses.

The first methodology is based on a Design Power History (DPH) established for each plant application that represents the maximum nodal power within the core for a reload. The DPH is a conservative (bounding) power history for each reload and is calculated based on the computer codes discussed in this section and the criteria discussed in Section 3.3 of this report. A conservative power uncertainty is also added to the DPH when it is used in licensing calculations. In ABB/C-E's responses (Reference 7), they have modified the application of DPH in STAV6.2 code calculations. They have stated that continuous operation of a single fuel rod at the DPH for its design life can not be sustained due to fissile depletion of the rod. Therefore, ABB/C-E has proposed a series of six power histories with each of the six power histories at the DPH for one-sixth of the fuel rod's life at different burnup levels. Each of the six power histories will operate at the DPH for a burnup interval such that they will bound the power history of all possible fuel rods.

The base irradiation power history (not at the DPH) that makes up five-sixths of each of the six power histories is based on the limiting power history determined from the approach discussed below. It is these six power histories along with a conservative uncertainty applied to these power histories that will be used in STA6.2 calculations to demonstrate adherence to their criteria.

ABB/C-E was questioned on whether the DPH included transients due to AOOs because the power history must include the effects of AOOs. Specifically, the DPH must include those AOOs that can affect the calculational results for the specific criterion being evaluated. ABB/C-E responded in Reference 7 that the DPH is provided to the plant operator in terms of an LHGR operating limit, referred to as the Thermal-Mechanical Operating Limit (TMOL), that should not be exceeded during normal steady-state operation and, therefore, does not include transients due to AOOs. ABB/C-E further stated that they will impose simulated power excursions initiated from a limiting power history to simulate transients which can occur during AOOs. The limiting power history and the simulated power excursions represents the proposed ABB/C-E second power history methodology discussed below.

The second power history methodology is designated by ABB/C-E as a limiting power history and the power histories are selected from the physics calculations for a specific plant application. ABB/C-E originally proposed (Reference 4) to select the limiting power histories based on three selection criteria and nine rod power histories from three maximum burnup/LHGR assemblies. ABB/C-E in response (Reference 7) to questions has elected to modify and expand the number of fuel assemblies and fuel rods selected to include the following: a) three assemblies with the maximum LHGR in each of the three cycles for a given fuel type, and b) three assemblies with maximum discharge burnup after N-2, N-1, and N cycles, where N is the maximum number of cycles the assemblies are to operate. Therefore, six fuel assemblies are selected from which individual power histories will be calculated and selected. The two UO_2 rods and the one UO_2-Gd_2O_3 rod which achieve maximum nodal LHGR from each of these six bundles will be selected for the limiting power histories. In addition, the two UO_2 rods and the one UO_2-Gd_2O_3 rod which achieve the maximum burnup from each of these six bundles will be selected for the limiting power histories.

The 36 limiting power histories that are chosen from this selection process are increased by a power uncertainty factor and then input into the STAV6.2 code with nominal dimensions and best estimate models to determine which rods and their power histories are the most limiting for the specific criterion. The most limiting power histories in terms of centerline temperature, rod pressure and cladding strain are selected for a UO_2 and a UO_2-Gd_2O_3 fuel rod. These limiting power histories are superimposed with the simulated transients from AOOs along with other code input conservatisms to perform the licensing calculations that will be discussed below for each criterion. The simulated transients are interspersed throughout each cycle of operation. The Laboratory concludes that the DPH and limiting power histories are in principle acceptable; however, their application to specific fuel rod criterion will be evaluated in the appropriate subsections that follow.

4.2.1 Rod Internal Pressure

There are two major calculations in regards to rod internal gas pressures, the first being the rod internal pressure limit for each ABB/C-E fuel design and the second the calculation of rod internal pressures to demonstrate that the ABB/C-E fuel design does not exceed the rod pressure limit. The following discussion is divided into two parts, the Rod Pressure Limit and Calculation of Rod Pressure.

4.2.1.1 Rod Pressure Limit

As discussed earlier in Section 3.3.1 the ABB/C-E criterion is to prevent the fuel cladding outward creep rate from exceeding the fuel swelling rate. This is sometimes referred to as the "cladding liftoff" criterion because if the cladding creep rate exceeds the fuel swelling rate the cladding will eventually liftoff from the fuel surface after some period of time under this condition. The calculation of the rod pressure limit is only dependent on the creep rate of the cladding and the fuel swelling rate and their respective uncertainties such that ABB/CE utilizes the upper bound creep rate and the lower bound swelling rate for this calculation. The cladding creep rate has been shown to be dependent on the flux rate, cladding temperature and stress, and has been shown to have a factor of 2 uncertainty even for cladding of similar types/fabrication history (Reference 17). Fuel swelling for light-water reactor (LWR) fuel at steady-state operating conditions is only dependent on fuel burnup and has less uncertainty than the cladding creep model.

The NRC has evaluated the ABB/C-E cladding creep and fuel swelling models as part of the review of CENPD-285(P) and found the secondary creep model to have too strong of a dependence on cladding stress. In addition, the

ABB/C-E uncertainty on their creep model was too small that resulted in a non-conservative estimate of the rod pressure limit. As a result of the review of CENPD-285(P) the NRC concluded that, for the purpose of providing a conservative estimate of the rod pressure limit, a simple creep equation should be used along with double the uncertainty quoted in Reference 3 for determining the upper bound creep rate. The simple creep equation should consist of the creep rate equal to a constant multiplied by equivalent stress raised to the exponent of 1.5. The constant for this equation is determined from the best estimate equivalent stress and creep rate, which is determined using the best estimate STAV6.2 model, nominal fuel rod dimensions, and the thin wall stress model, at cladding liftoff. The lower bound of the design pressure should then be determined using the simple creep rate equation described above with the lower bound pellet swelling rate, the upper bound cladding creep rate, and conservative fuel rod dimensions. This simple creep rate equation described above is only to be applied in determining the rod pressure design limit.

The Laboratory's evaluation of the ABB/C-E fuel swelling model has found the best estimate model and its associated uncertainty to be conservative relative to the calculation of the rod pressure limit.

The Laboratory concludes that the methodology for determining the rod pressure limit with the required modifications to the cladding creep model is acceptable.

4.2.1.2 Calculation of Rod Internal Pressures

The calculation of rod internal pressures is dependent on fission gas release (FGR) and rod void volume predictions that are in turn dependent on the fuel rod power histories. The calculation of rod internal pressures is important because it usually limits the DPH and TMOL for maximum nodal LHGR at extended burnup levels for a given fuel design application.

The STAV6.2 fuel performance code and the "BWR" and "PWR" FGR models are used to calculate rod internal pressures. The NRC review of this code concluded that the "BWR" FGR model was valid to a rod-average burnup of 40 GWd/MTU and the "PWR" FGR model was valid to a rod-average burnup of 50 GWd/MTU. ABB/C-E has proposed (Reference 7) to using the following conservative input to the STAV6.2 code: include a conservative power uncertainty factor on the power histories; fabrication uncertainties such as dimensions, pellet density, densification, and fill gas pressure; and uncertainties in some of the STAV6.2 models such as the athermal FGR, pellet swelling, and cladding creep rate models. The NRC has concluded in the review of CENPD-285(P) (Reference 3) the uncertainty for cladding creep rate should be double the value quoted in Reference 3.

ABB/C-E was questioned on why the thermal "BWR" and "PWR" FGR models did not have a calculational uncertainty particularly since this is the major quantity that is estimated in the rod pressure calculation. ABB/C-E responded (Reference 7) that their very conservative estimate of uncertainty on rod powers used for DPH and limiting power histories bounded their FGR predictions. They further stated that the large uncertainty on the rod powers, plus the fabrication and other model uncertainties will provide a conservative prediction of rod internal pressures. In addition, ABB/C-E has provided STAV6.2 predictions against FGR data with their power uncertainty factor applied that demonstrates their predictions bound the FGR data within the burnup restriction of 50 GWd/MTU (rod average) placed on the code. It is also noted that the "PWR" FGR model appears to conservatively predict FGR up to rod average burnups of 45 GWd/MTU. ABB/C-E also provided an example of the impact these uncertainties have on the rod pressure calculation in Reference 7.

The Laboratory has reviewed the STAV6.2 model uncertainties and analysis methodology and concludes that the ABB/C-E internal pressure calculations are adequately conservative with the burnup limitations on the FGR models and the increase in the uncertainty in the cladding creep model noted above and, therefore, acceptable.

4.2.2 Cladding Stresses

The cladding stresses are calculated using the VIK-2 code described in CENPD-285(P) (Reference 3). ABB/C-E has proposed that the cladding stress intensities relative to ASME (Reference 8) derived design stress intensities will be more limiting at beginning-of-life (BOL) than at end-of-life (EOL) because of the increase in yield strength and, therefore, only BOL conditions need to be considered.

ABB/C-E has used conservatively bounding values for input to this calculation including worst case fabrication tolerances, maximum LHGR, maximum cladding temperatures, and maximum coolant pressure (due to the design over pressurization event) to demonstrate that their current SVEA-96/100 designs are acceptable.

The Laboratory concludes that the ABB/C-E methodology for calculating cladding stresses is conservative and, therefore, acceptable.

4.2.3 Cladding Strains

The cladding strains are calculated using the STAV6.2 fuel performance code along with some input conservatisms such as worst case fabrication tolerances, minimum pellet densification, maximum pellet swelling, and cladding creep rate. The modifications on the stress exponent and uncertainty to the ABB/C-E cladding creep model discussed in Section 4.2.1.1 of this report and further discussed in the review of CENPD-285(P) should be used for this analysis.

ABB/C-E has provided examples of their cladding strain calculations in Reference 4; however, these examples used only steady-state power histories without any transients due to AOOs. In addition, ABB/C-E has suggested in Reference 4 that only their calculated elastic strains be included because all of the plastic strain is due to creep. Section 4.2 of the SRP that addresses the 1% strain limit states that "uniform strain (elastic and inelastic) is defined as transient-induced deformation with gage lengths due to cladding dimensions." Therefore, inelastic or plastic strains of the cladding due to gradual fuel swelling at constant power are not included but elastic and plastic strains induced by sudden power changes should be considered in evaluating the 1% strain criterion. The earlier examples of ABB/C-E power histories in Reference 4 did not include the effects of transients due to AOOs and, therefore, are not considered to be acceptable for calculating cladding strain. However, the two new power history methodologies provided in Reference 7 and discussed in Section 4.2 of this report are considered acceptable. The six power histories with power increases to the DPH at various intervals should include the elastic plus plastic strains that result from the power increase to the DPH (plus the power uncertainty factor) for calculating cladding strains for each of the six histories. The second power history methodology that includes the limiting power histories with transients due to AOOs interspersed throughout each cycle should include those elastic plus inelastic strains that result from these transients. The maximum elastic plus inelastic strains from these two power history methodologies should be used for comparison to the 1% strain limit.

The Laboratory concludes that the modified power history methodologies in Reference 7 and the cladding strains (elastic plus inelastic) due to sudden power increases as defined above provide an acceptable methodology for ABB/C-E to calculate strains.

4.2.4 Cladding Hydriding

The ABB/C-E criterion and analysis methodology for cladding hydriding due to waterside corrosion is not considered to be adequate because it is based on an average hydrogen content in the cladding rather than a maximum or localized hydrogen amount. However, the criterion and analysis methodology for waterside corrosion appears to be acceptable. Because the level of waterside corrosion and hydride levels are directly related, the Laboratory concludes that the level of cladding hydriding is adequately controlled through the criterion on maximum cladding corrosion (see following Section).

4.2.5 Cladding Corrosion

The cladding corrosion model in STAV6.2 is claimed to predict maximum oxide thickness for ABB/C-E designs; however, it has only been verified against data up to a maximum rod-average burnup of 21 GWd/MTU. In response to questions ABB/C-E has provided additional cladding corrosion data for their current cladding type in Figures A6-1 (average oxide thicknesses) and A6-2 (maximum oxide thicknesses) of Reference 7 up to an assembly-average burnup of 49 GWd/MTU. Comparing the STAV6.2 corrosion model predictions to these higher burnup data demonstrates that the model under predicts cladding corrosion by a factor of 3 to 4 at extended burnups. However, the data in Figure A6-2 of Reference 7 do demonstrate that corrosion for ABB/C-E's current cladding type is within the ABB/C-E limit for maximum corrosion up to a rod-average burnup of 50 GWd/MTU that is consistent with the conclusions from the NRC review of CENPD-285-P. The non-conservatism in the ABB/C-E corrosion does not significantly affect ABB/C-E performance predictions because as demonstrated in Figure A-2 of Reference 7 maximum corrosion levels observed in ABB/C-E designs are acceptable at rod average burnups up to 50 GWd/MTU

The Laboratory concludes that the current SVEA 96/100 fuel cladding corrosion is acceptable up to the burnup restrictions identified in this report based on the data in Reference 7.

4.2.6 Cladding Collapse

The ABB/C-E collapse calculations are performed with the COLLAPS-II code that was evaluated in the NRC review of CENPD-285(P). The review of CENPD-285(P) imposed an increase in the uncertainty of the ABB/C-E creep model prediction that is also briefly discussed in Section 4.2.1.1 of this TER. ABB/C-E has provided an example of the conservative input used in the calculation of cladding collapse in Section 3.3.6 of Reference 4. The Laboratory questioned ABB/C-E on the use of the mean calculated cladding creep rate rather than an upper bound creep rate for cladding collapse. ABB/C-E responded (Reference 7) that the use of maximum cladding temperatures and in-reactor residence times, and the DPH power history (fast flux) will bound the uncertainty in the cladding creep model. The Laboratory has examined this assertion and found that the increased cladding temperature has virtually no effect (<2%) on the cladding creep; however, the fast neutron flux used by ABB/C-E and the maximum in-reactor residence times for this analysis are more than a factor of 2 conservative when compared to the best estimate core average fast flux. The cladding creep model is nearly linearly dependent on fast flux so this results in greater than a factor of 2 conservative increase in the creep rate. This, along with the conservative assumptions on system pressure and reduction in wall thickness due to corrosion, is judged to more than compensate for not using the NRC recommended doubling of the creep uncertainty factor from that quoted in Reference 3 (Section 4.2.1.1) for the cladding collapse analysis. In addition to these conservatisms, ABB/C-E assumes worst case fabrication tolerances including cladding ovality, no fission gas release, and no cladding support from the fuel pellets (assumes an infinite empty cladding tube).

The Laboratory concludes that the ABB/C-E methodology for calculating cladding collapse is conservative and, therefore, acceptable.

4.2.7 Fatigue

The ABB/C-E fatigue calculation for the cladding is based on the calculation of maximum alternating stress intensities in accordance with the ASME Boiler and Pressure Vessel Code (Reference 8) and the Zircaloy fatigue design curve of O'Donnel and Langer (Reference 16) to calculate fatigue usage factors. The design fatigue curve includes the more conservative of a factor of two on stress amplitude or 20 on the number of cycles. The calculated cumulative usage factor must be less than 1.0 for the design life of the fuel. For these calculations ABB/C-E conservatively assumes worst case fabrication tolerances, minimum densification, maximum fuel swelling, maximum cladding creep rate and cladding thinning due to maximum allowed corrosion for the entire life of the rod.

ABB/C-E has calculated the maximum alternating stresses using the STAV6.2 code (Reference 3) and a conservative power cycling scheme from hot 100% to 0% to 100% power superimposed on both the DPH and the plant specific limiting power history in Section 4.3.7 of Reference 4. For the example provided, the plant specific limiting power history with the conservative power cycling scheme superimposed provided the more limiting usage factor.

In addition, ABB/C-E has also calculated the maximum stress intensity for startup from cold conditions to an overpower condition a conservative number of times over the life of the fuel using the VIK-2 code. Combining the limiting fatigue usage factor from the hot power cycling scheme and the usage factor from power cycling from the cold condition were significantly below the 1.0 usage factor for the SVEA 96 design application example provided.

ABB/C-E has also calculated fatigue due to hydraulic forces and the resulting bending forces at 100% flow. These calculations also demonstrate substantial fatigue margin.

In addition, ABB/C-E has proposed in Reference 4 that they also be able to modify the fatigue analysis to reflect actual plant power cycling schemes rather than the conservative power cycling scheme discussed above. The use of actual plant power cycling schemes reduces the conservatism in the fatigue analysis.

The Laboratory concludes that the conservative methodology including the bounding power cycling scheme provided by ABB/C-E for fuel rod fatigue analysis is acceptable for application to current fuel designs.

4.2.8 Cladding Overheating

The ABB/C-E methodology for maintaining adequate margin to boiling transition in terms of the MCPR is addressed in Reference 18 and has been

addressed in the review of ABB/C-E thermal hydraulics methods and will not be discussed further in this TER.

4.2.9 Fuel Overheating

The ABB/C-E methodology for calculating maximum fuel centerline temperatures (and, therefore, preventing fuel melting) is based on STAV6.2 calculations as discussed in Section 4.3.9 of Reference 4. The original plant specific limited power history provided by ABB/C-E was found to not include the power transients from AOOs. As a result ABB/C-E has altered the plant specific limited power history methodology to include transients from AOOs (Reference 7). This should include both fast and slow transients to find those that are the most limiting with regards to the fuel melting analysis. This methodology also includes the use of worst case fabrication tolerances, uncertainties in power, and maximum densification as input to the STAV6.2 code. The individual temperature uncertainties from each of these worst case inputs to STAV6.2 is squared and the square root of their sum represents the total temperature uncertainty. As noted in the review of STAV6.2, the code conservatively calculates fuel temperatures at low to moderate burnup levels even with best estimate input parameters. Therefore, the Laboratory concludes the ABB/C-E methodology is conservative and, therefore, acceptable at low to moderate burnup levels.

However, there is growing evidence of a decrease in fuel thermal conductivity at high burnup levels (References 19 and 20) and the STAV6.2 fuel thermal conductivity model does not account for this high burnup effect. The STAV6.2 code thermal calculations also have not been validated at high burnups. ABB/C-E was questioned (Reference 5) on the possible under prediction of fuel centerline temperatures at high burnup due to their lack of a burnup degradation on fuel thermal conductivity. ABB/C-E responded (Reference 7) that they would reduce their fuel melting temperature by an amount equal to the reduction in power-to-melt estimated by Lancing et al. (Reference 21) using the Lucuta burnup degradation effect (Reference 20). The Laboratory concludes that this revised methodology for preventing fuel melting is acceptable.

4.2.10 Fuel Rod Bowing

ABB/C-E has claimed in Section 4.3.10 of Reference 4 that fuel rod bowing has no impact on the mechanical or thermal performance margins of their fuel designs. The review of the impact of fuel rod bowing on thermal performance margins is not within the scope of this review and will not be discussed.

ABB/C-E was requested (Reference 5) to supply their fuel rod bowing data for their applicable SVEA-96/100 fuel design. ABB/C-E responded with gap measurements on 2,482 fuel rods with up to assembly-average burnups of 49 GWd/MTU that showed some rod-to-rod gap closure due to rod bowing in BWR plants in Europe. However, ABB/C-E indicated that all those assemblies with rod-to-rod gaps with greater than 40% closure were due to either excess fuel rod growth or corrosion at the fuel rod end caps (creating interference problems with the tie plate). The former excessive growth has been corrected and the end cap corrosion is not suspected to be a problem in U.S. reactors with different primary coolant chemistry requirements. Eliminating all of the assemblies with these two problems leaves 1,770 rods with gap measurements of which 1,763 had no detectable rod bowing and 7 fuel rods had rod-to-rod gap closures of less than 40%. These data demonstrate that rod-to-rod gap closures greater than 40% are not likely up to an assembly-average burnup of 49 GWd/MTU as long as rod growth and corrosion (see Sections 4.1.5 and 4.2.5, respectively) are adequately controlled.

The Laboratory concludes that fuel rod bowing in the SVEA-96/100 fuel designs will not decrease mechanical performance margins up to a rod-average burnup of 50 GWd/MTU, which is consistent with the conclusions from the NRC review of CENPD-285-P.

4.2.11 Pellet Cladding Interaction (PCI)

No specific criteria is identified in Section 4.2 of the SRP (Reference 1) for PCI other than the 1% strain limit and the prevention of fuel melting already discussed in Sections 4.3.3 and 4.3.9, respectively, in this TER. However, ABB/C-E does have a PCI threshold curve for those SVEA-96/100 fuel designs without a cladding liner that is shown in Figure 4.3.11-1 of Reference 4. The fuel rods with a cladding liner contain a thin liner of soft Zirconium alloy on the inside surface of the cladding. The soft liner helps to prevent the very high localized stresses resulting from PCI from developing in the Zircaloy-2 cladding and thus prevent cladding failure. Therefore, the ABB/C-E fuel designs without a liner have restrictions on fuel power ascension. Those designs with the cladding liner have no restrictions on fuel power ascension.

The Laboratory concludes that ABB/C-E methodology adequately addresses PCI for their fuel designs.

4.3 Steady-State Initialization for Transient and Accident Analyses

ABB/C-E's original submittal (Reference 4) did not include the application methodology for their steady-state codes that are used to initialize transient and accident analyses; ABB/C-E was requested to provide this methodology (Reference 5). ABB/C-E responded (Reference 7) that the STAV6.2 code is the principal code used to calculate steady-state initial conditions. The STAV6.2 code provides initial steady-state gap conductance, gap sizes, gas composition and plenum volumes for the loss of coolant accident (LOCA) analysis. The LOCA analysis code then uses this input along with the same gap conductance and thermal models as in STAV6.2 to calculate the same fuel rod steady-state thermal conditions as those calculated by STAV6.2.

The STAV6.2 code is also used to initialize gap conductance for fast transients (for calculating the delta critical power ratio, ΔCPR), control rod drop accident, stability, and dose analyses. The steady-state methodology for initialization of each of the above transient and accident analyses are discussed in the following subsections.

4.3.1 LOCA Initialization

For the ABB/C-E methodology the initialization of LOCA requires input of an average core gap conductance for the core vessel thermal-hydraulic response and a hot channel gap conductance. ABB/C-E has provided an example STAV6.2 calculation of LOCA initialization that uses worst case fabrication tolerances, bounding model uncertainties, and bounding power histories as described in References 4 and 7. The bounding power history for the LOCA hot channel analysis is provided in Figure A2-1 of Reference 7. In addition, ABB/C-E has provided an example of the degree of conservatism in the STAV6.2 calculation of fuel temperature for LOCA initialization in Figure A2-2 of Reference 7 by comparing LOCA initialization temperatures to the best estimate calculated temperatures based on the plant specific limiting power history. This is not considered to be a valid comparison because the bounding conservative powers used in the LOCA analyses are generally not considered an uncertainty in the calculation by the NRC and, therefore, not included in calculated temperature uncertainties. The NRC is concerned with the calculational uncertainties in the code itself and that LOCA calculated stored energy is calculated at a bounding 95/95% confidence level. However, the STAV6.2 calculation of fuel temperatures at the bounding power histories and the LOCA LHGR limits are considered to be conservative as noted in the review of STAV6.2. The additional conservatism in the STAV6.2 inputs for LOCA (worst case fabrication tolerances and the bounding model uncertainties) are considered to bound the STAV6.2 calculational uncertainties by at least a 95/95% confidence level.

The hot channel gap conductance and STAV6.2 thermal models are used in the hot plane fuel rod heat-up analysis code. This code calculates the fuel stored energy by ramping the rod to the LOCA LHGR limit and the resulting fuel stored energy is used to calculate PCTs.

Based on the above analysis methodology the Laboratory concludes that the STAV6.2 initialization for LOCA is acceptable.

4.3.2 Fast Transients

The calculation of the ΔCPR from fast transients due to AOOs is dependent on a core-average gap conductance and hot channel gap conductance. The change in gap conductance with power increases is important for both core-average and hot channel conditions. A slow increase (with low values) in core-average gap conductance will conservatively maximize core power changes while a fast increase (with high values) in the hot channel gap conductance will conservatively maximize the ΔCPR.

ABB/C-E has two approaches for providing gap conductance values for the ΔCPR analysis. The first is to use conservatively low values of changes in gap conductance with changes in fuel-average temperatures for the core-average channel, and a conservatively high values of changes in gap conductance for the hot channel analysis. The STAV6.2 code is used to provide the conservative (high and low) values of gap conductance (core-average and hot channel) as a function of average fuel temperatures.

The second ABB/C-E approach is to calculate nominal values of gap conductance as a function of average fuel temperatures and the estimated uncertainties in the nominal gap conductance using the STAV6.2 code. The core-average gap conductance is determined by looking at a wide range of assemblies in the core and representative rods are selected from these assemblies and STAV6.2 calculations are performed for these representative rods at various burnup levels. These calculations provide various values of gap conductance versus average fuel temperature that are fit to polynomial expressions at different burnup levels for the different assemblies. The core-average gap conductance is then calculated for different burnup levels using the polynomial expressions for the different assemblies weighted by the number of the assemblies in the core. The hot channel expressions for gap conductance versus average temperature are similarly determined except only those assemblies suspected to contain the hot channel are examined and these values are not averaged. The gap conductance uncertainties for this approach are determined using STAV6.2 and perturbing the input uncertainties for the code. These gap conductance uncertainties are used to determine their impact on ΔCPR and then their change in ΔCPR are statistically combined with the other ΔCPR uncertainties.

The Laboratory concludes that the STAV6.2 initialization of gap conductance methodology for this transient analysis is acceptable.

4.3.3 Control Rod Drop Accident

The CRDA uses nominal values of gap conductance calculated with the STAV6.2 code for those bundles next to the dropped rod. The uncertainties in gap conductance are included in the CRDA analysis, as described in Reference 22.

The Laboratory has concluded that the STAV6.2 code initialization of gap conductance methodology for this accident is acceptable.

4.3.4 Stability Analysis

The ABB/C-E stability analysis calculates a nominal gap conductance for each bundle type. There are competing gap conductance effects in the stability analysis that makes it difficult to assess whether a low or high gap conductance value is conservative for this analysis. Therefore, nominal values are used for this calculation although the uncertainty in gap conductance is one of the uncertainties used in the evaluation of uncertainties in the stability analysis.

The Laboratory concludes that the STAV6.2 code initialization of gap conductance methodology for the stability analysis is acceptable.

4.3.5 Dose Calculations

In the original response to questions (Reference 6), ABB/C-E indicated that the STAV6.2 code would be used to calculate the fission product inventory for dose calculations for LOCA and fuel handling accidents. ABB/C-E has revised their response in Reference 7 to state that STAV6.2 will not be used for input to dose calculations for LOCA and they will instead use the conservative assumptions in Regulatory Guide 1.3 (Reference 23) and for the fuel handling accident the assumptions in Regulatory Guide 1.25 (Reference 24). This revision is consistent with the SRP and, therefore, the Laboratory concludes that it is acceptable.

5.0 CONCLUSIONS

The Laboratory has completed its review of the ABB/C-E mechanical design criteria and analysis methodology for demonstrating acceptable fuel assembly and fuel rod performance as defined by Section 4.2 of the SRP and has come to the following conclusions.

The ABB/C-E fuel assembly and fuel rod analysis methodology is acceptable with the following limitations.

- Based on the validity of the FGR and corrosion models, and other models, the application of CENPD-287 is approved up to a rod average burnup level of 50 GWd/MTU.

- The cladding creep model in STAV6.2 has too strong a dependence on cladding stress that results in a non-conservative estimate of the rod pressure limit. For determination of the rod pressure limit, use of a simple creep equation that is a function of stress to the exponent of 1.5 has been proposed. The uncertainty in creep as originally proposed in Reference 3 is too small and should be increased by a factor of 2 for both the simple creep equation and the creep relationship in STAV6.2

- The Laboratory recommends that the PWR FGR model be used in STAV6.2 at rod-average burnups above 40 GWd/MTU.

- The STAV6.2 code is acceptable for application to urania-gadolinia fuel with gadolinia content up to 8 wt%. The ABB/CE urania-only fission gas diffusion constants should be used for both urania-only and urania-gadolinia fuel rod applications.

- The calculation of uniform cladding strain should be the elastic plus inelastic strains due to power increases due to normal operation and AOOs (see Section 4.2.3).

6.0 REFERENCES

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CENPD-287-NP-A REPORT

Part II

Body of Report

ABB Combustion Engineering Nuclear Operations

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1 SUMMARY AND CONCLUSIONS

Summary

This report contains the ABB methodology for the fuel assembly and fuel rod mechanical evaluation identified in Section 4.2 of the Standard Review Plan, NUREG-0800 (Reference 1.3). It also contains an application of that methodology to the ABB SVEA-96 fuel assembly which demonstrates that the SVEA-96 assembly satisfies the ABB design criteria. Satisfaction of the ABB design criteria assures compliance with the objectives of Section 4.2 of the SRP, and, therefore, assures compliance with General Design Criteria 10, 27, and 35 of 10CFR50, Appendix A (Reference 1.4). Similar information supporting the thermal-hydraulic, nuclear, and safety analyses evaluations are provided in Reference 1.1.

Specifically, this report contains the following:

1. Description of the ABB SVEA-96 BWR watercross fuel assembly design,

2. The ABB fuel assembly and fuel rod mechanical design criteria,

3. The ABB design evaluation methodology for evaluation of performance relative to those criteria for normal operations and Anticipated Operational Occurrences (AOOs),

4. Sample application of the ABB design evaluation methodology demonstrating compliance of the SVEA-96 assembly with the design criteria for normal operations and AOOs,

5. Summary of the computer codes used in ABB methodology,

6. Description of the manufacturing inspection measures which assure that the assembly is constructed as required by the design specifications,

7. Summary of the operating experience with the SVEA-96 design and similar ABB designs,

8. Summary of the ex-core prototype test programs,

9. Discussion of ongoing testing, inspection, and surveillance plans.

As explained in Section 3, methodologies for the evaluation of accident conditions and sample applications of those methodologies to SVEA-96 are contained in other topical reports.

General design criteria as well as the design criteria for the fuel rods and other assembly components are clearly stated. The methods used to evaluate assembly and component performance against these design criteria are then systematically addressed. An illustrative evaluation of the SVEA-96 design relative to the design criteria using the methodology described is also provided. This evaluation is described in conjunction with the methodology description to assist the reader in understanding compliance with the requirements of Section 4.2 of the Standard Review Plan.

Conclusions

The information contained in this report supports the following conclusions regarding the fuel assembly and fuel rod mechanical characteristics of the SVEA-96 BWR fuel assembly:

1. The design bases identified are sufficient to assure that the requirements and guidelines identified in Section 4.2 of NUREG-0800 10 CFR 50, Appendix A and Section III of the ASME Code (Reference 1.2) will be satisfied.

2. The methodology for evaluating fuel assembly and fuel rod mechanical behavior relative to the design bases is acceptable for licensing and design purposes, and

3. The SVEA-96 BWR fuel assembly evaluations provide an illustration of the methodology to be utilized for each application of SVEA-96 BWR fuel assemblies. These evaluations also demonstrate that the SVEA-96 assembly meets the fuel performance, mechanical, and materials design bases under normal operation and anticipated operational occurrences to a peak pellet burnup of 65 MWd/kgU and a peak assembly burnup of 55 MWd/kgU.

2 GENERAL DESCRIPTION

2.1 Assembly Description

The primary objective of the SVEA design is integrity and reliability of the fuel rod and assembly. To this end, numerous features have been adopted with the goal of achieving zero fuel rod failures during reactor operation. While these features will be discussed more fully in the following detailed mechanical design description, it is instructive to summarize some of the major SVEA-96 mechanical characteristics which have contributed to the demonstrated reliable operation of the SVEA-96 and SVEA-100 assemblies over the past several years.

- The relatively large number of fuel rods (96 or 100) in an assembly allows relatively high bundle powers while maintaining very modest rod powers. The low linear heat generation rate (LHGR) and increased heat transfer area allow the fuel to operate at substantially lower temperatures than traditional designs with fewer fuel rods per bundle. Lower fuel temperatures reduce fission gas release, which provides greater margins to fuel thermal-mechanical design criteria for a given bundle burnup, or allows higher bundle discharge burnup for the same margin to fuel thermal-mechanical limits. The reduced cladding heat flux associated with the larger number of fuel rods also improves Critical Power performance and reduces the rate of Zircaloy corrosion. The reduction in LHGR is also sufficient to allow the operation of the SVEA-96 bundle below the Pellet Clad Interaction (PCI) threshold for most applications, thus reducing the probability that operating guidelines (e.g., PCIOMRs) or a zirconium liner will be required.

- The integral construction provided by welding the watercross to the midspan of the outer channel results in substantially enhanced channel dimensional stability. ABB BWR reactor experience has shown that the SVEA channels are less susceptible to channel bulge and bow than open lattice designs. Reduction of the unsupported outer channel transverse span by a factor of two substantially reduces channel bulge. Furthermore, the axial restraint that the watercross exerts on the outer channel restricts differential outer channel growth and reduces channel bow.

- The SVEA channel design allows unrestricted growth of the fuel rods inside the channel. This feature allows the channels to be rigidly attached to the bottom nozzle avoiding an exposure-dependent leakage flow path between the channel and bottom nozzle. It also tends to reduce channel bulge. Furthermore, the tensile load associated with the assembly weight during fuel handling is carried by the channel rather than the fuel rods.

The subbundles inside the SVEA channel can grow independently of the channel, and the overall assembly length increase with burnup is relatively low since the channel grows less than the fuel rods.

- The fully recrystallized Zircaloy-2 cladding is beta-quenched at an intermediate reduction stage. This cladding is referred to as "LK-II" and has been demonstrated to exhibit excellent resistance to nodular corrosion as well as Crud Induced Localized Corrosion (CILC). There also is convincing evidence that the onset of nodular corrosion in general, and the rate of CILC in particular, increase with increasing surface heat flux. Consequently, the relatively low surface heat flux associated with the SVEA-96 design is a major contributing factor to its observed high level of corrosion resistance.

The SVEA-96 fuel assembly was designed specifically for U.S. domestic BWRs. The SVEA-96 fuel assembly consists of three basic components:

- The fuel bundle,
- The fuel channel, and
- The handle.

Section 5 provides typical numerical data concerning the SVEA-96 design for a 3810-mm active fuel C-lattice plant and a 3689-mm D-lattice plant, and Figure 2.1a, 2.1b, and 2.2 show the SVEA-96 assembly. Figure 2.1a is based on a 3810-mm active fuel length and is referred to as "Style 1". Figure 2.1b is based on a 3689-mm active fuel length and is referred to as "Style 2".

The fuel bundle consists of 96 fuel rods arranged in four 5x5 minus 1 (5x5-1) subbundles. The channel has a cruciform internal structure with a square center channel that forms gaps for non-boiling water during normal operation. The subbundles are inserted into the channel from the top and [Proprietary Information Deleted] This design principle has been used in various ABB BWR fuel assembly designs for many years, and eliminates the leakage flow path at the bottom end of the channel. This design feature also avoids stresses in the tie rods during normal fuel handling operations. The fuel assembly is lifted with a handle connected to the top end of the channel.

The subbundles are freestanding inside the channel. There is sufficient space for subbundle growth at the top of the assembly to avoid restriction due to differential growth between the fuel bundles and the channel.

The bottom of the transition piece, or "nose piece," seats in the fuel support piece. The top ends of fuel assemblies are supported laterally against the adjacent assemblies through the interaction of leaf springs on two sides, and the upper core grid on the other two sides. Compatibility evaluations and operating experience, have confirmed the mechanical compatibility of the SVEA-96 assembly with U.S. reactors and several existing fuel types.

The control rod gap, and the gap that does not contain a control rod, depends on the plant lattice geometry. Typical values for SVEA-96 fuel assemblies in C-lattice and D-lattice plants are shown in Figures 2.3a and 2.3b. [Proprietary Information Deleted] These gap widths provide adequate clearances to the control blades and rollers. The SVEA-96 assemblies also provide adequate clearances to instrument guide tubes. The improved resistance of the SVEA channel to bulge and bow assures that these conclusions based on beginning-of-life dimensions continue to apply throughout the lifetime of the bundle.

Reference is frequently made in this report to the "SVEA-100" design as opposed to the SVEA-96 design, and it is instructive to explain the difference. The "SVEA-100" design is very similar to the "SVEA-96" design discussed in this report with four additional fuel rods in the center of the bundle. "SVEA-100" is the designation of the ABB 10x10 SVEA design which has been optimized for use in BWRs built by ABB. The SVEA-96 design has been optimized for reactors designed by General Electric and Siemens. The ABB BWRs have an assembly pitch slightly greater than 152.4 mm. These reactors are operating in Sweden and Finland. [Proprietary Information Deleted] Therefore, due to the similarity of the two designs, reactor experience and mechanical test results obtained for the SVEA-100 assembly are generally applicable to the SVEA-96 assembly.

Reference is also made in this document to the SVEA-64 design. The SVEA-64 design utilizes a [Proprietary Information Deleted] 4x4 fuel rod array. [Proprietary Information Deleted]

2.1.1 Handle with Spring

Figure 2.7 shows the SVEA-96 handle and leaf spring design. The handle and leaf spring configuration are fitted to the top end of the channel. [Proprietary Information Deleted]

The handle is equipped with a double leaf spring which maintains contact with the corresponding springs on adjacent assemblies and firmly presses the fuel assembly into the corner of the upper core grid.

An individual identification number for each fuel assembly is engraved in the handle.

[Proprietary Information Deleted]

2.1.2 Fuel Transport

Transport of the fuel to the reactor site is performed in approved shipping containers. [Proprietary Information Deleted] Shipping tests are utilized to fully qualify the transport method.

2.1.3 Lattice and Fuel Rod Types

Each subbundle is a 5x5-1 lattice. The fuel assembly has [Proprietary Information Deleted]

2.2 Fuel Subbundle Description

The fuel subbundle designs are shown in Figures 2.5a and 2.5b. Each subbundle is a separate unit with top and bottom tie plates. [Proprietary Information Deleted]

The tie rods are connected to the top and bottom tie plates with threaded end plugs extending through the plates and secured by nuts. [Proprietary Information Deleted]

2.2.1 Top and Bottom Tie Plates

The top tie plates are [Proprietary Information Deleted]

ABB has accumulated extensive in-reactor experience with these basic tie plate designs. [Proprietary Information Deleted]

2.2.2 Standard Fuel Rods and Tie Rods

Typical standard fuel rods are shown in Figure 2.8a and 2.8b. The tie rods are shown in Figures 2.9a and 2.9b.

The fuel consists of UO_2 or, in case of Burnable Absorber (BA) rods, UO_2-Gd_2O_3 ceramic pellets. The pellets are contained in Zircaloy-2 cladding tubes which are plugged and welded at the ends to encapsulate the uranium fuel. [Proprietary Information Deleted] They are fabricated from enriched uranium dioxide powder that has been compacted by cold pressing and then sintered to the required density.

The top and bottom end plugs are manufactured [Proprietary Information Deleted]

The top of the fuel rod has a plenum to accommodate fission gases as they are released from the pellets during irradiation. An [Proprietary Information Deleted] spring is located in the plenum. This spring is shown in Figure 2.11. Its function is to avoid fuel pellet stack motion and pellet damage during shipping and handling prior to irradiation.

The fuel rods are internally pressurized [Proprietary Information Deleted] Internal pressurization improves heat transfer inside the fuel rods and minimizes compressive cladding stresses and creep-down due to the coolant operating pressure.

The two tie rods are identical to the standard rods with the exception of the top and bottom end plugs. These rods are structural members of the fuel assembly, and establish the overall subbundle length. [Proprietary Information Deleted]

Cladding

The cladding tube is an [Proprietary Information Deleted] Zircaloy-2 tube. The final surface treatment of the inner diameter of the tubes is [Proprietary Information Deleted]

The cladding tubes are manufactured according to specifications and procedures which produce optimum corrosion resistance. The LK-II process utilizes [Proprietary Information Deleted] The excellent corrosion performance of tubing manufactured with this process has been verified by several years of operation in a variety of BWRs.

2.2.3 Spacer Capture Rod

The spacer capture fuel rods are shown in Figure 2.10. A spacer capture rod is [Proprietary Information Deleted]

The tab welding process is performed such that the inside surface of the clad is not affected. This type of welded tab has been used since 1983. Annual post irradiation examinations have confirmed satisfactory performance of the tabs and the resistance welds.

2.2.4 Pellets

The pellet for SVEA-96 is especially designed to [Proprietary Information Deleted] A sketch of the enriched fuel pellet is shown in Figure 2.12.

The pellet sintering process is designed to minimize in-pile fuel pellet densification. [Proprietary Information Deleted]

Fuel pellets with burnable absorber (BA) consist of mixed Gd_2O_3 and uranium oxide powders. [Proprietary Information Deleted]

2.2.5 Spacers

The spacer is shown in Figure 2.13. The spacer design is based on earlier ABB 8x8 and SVEA-64 grid cell designs and utilizes [Proprietary Information Deleted]

The spacer grid is a [Proprietary Information Deleted]

The spacer grid is designed for [Proprietary Information Deleted] and to withstand all dynamic loads encountered during reactor operation. The spacers provide lateral support for the fuel rods, and minimize rod vibrations and axial loads that could lead to rod bowing. The spacers must also maintain sufficient space between fuel rods and between the rods and the channel to assure that thermal-hydraulic conditions are not compromised during reactor operations.

The spacers are fabricated from strip material [Proprietary Information Deleted]

The spacer design is well proven. The basic design was used originally for the ABB 8x8 assemblies and used subsequently in the SVEA-64 design. It is currently used for the SVEA-100 and SVEA-96 designs. Extensive in-reactor experience has not revealed any evidence of stress corrosion cracking, and has demonstrated that the spacers satisfactorily provide their intended function to high burnups. Mechanical testing has confirmed that the spacer functions as designed under loading associated with accident conditions.

2.3 SVEA-96 Fuel Channel

The Zircaloy-4 channel consists of a square outer channel with a double-walled internal cross structure which forms channels for non-boiling water. Cross sections are shown in Figures 2.2, 2.6a and 2.6b. The inner, cross channel (or "watercross") has a square central water channel and smaller water channels in each of the four wings. [Proprietary Information Deleted] The outer channel and the watercross structure form four subchannels for the subbundles.

[Proprietary Information Deleted]

In addition to providing channels for non-boiling water, the integral watercross design results in improved dimensional stability leading to reduced bow and bulge of the channels.

The outer channel wall thickness is [Proprietary Information Deleted] This provides greater strength [Proprietary Information Deleted] See Figures 2.6a and 2.6b. Screws in each of the four sides of the assembly secure the outer channel and the transition piece to the bottom support plate. The transition piece fits into the fuel support piece. [Proprietary Information Deleted] The channel and inlet transition piece are designed for compatibility with the reactor internals as well as other fuel types in the core. The outer envelope of the SVEA-96 channel and transition piece provide ample clearance for control rods and in-core instrumentation. The dimensional stability of the SVEA channel assures that ample clearances are maintained with burnup. The length of the assembly is compatible with the relative positions of the fuel support piece and upper core grid.

2.4 Offset of the SVEA-96 Assembly

[Proprietary Information Deleted]

All SVEA fuel assemblies installed in Swedish and Finnish reactors have been displaced toward the control rod gap, including the SVEA-64, SVEA-100, and SVEA-96 designs. The Nordic experience has been very good. No impact on the control rod motion has been observed.

[Proprietary Information Deleted]

2.5 Advanced Features

The operating experience described in Section 7 has involved the SVEA-96 assembly described in Sections 2.1 through 2.4. We anticipate that several advanced features for the SVEA-96 design, which are either qualified currently or in the qualification process, will be made available for reload application in the U.S. market in the next few years. Therefore, five of these features are described herein. Each of the features are intended to further improve fuel reliability.

2.5.1 Debris Filter

Experience in BWRs over the past several years has demonstrated the need for increased protection from debris failures. Therefore, ABB has developed a debris filter for the SVEA-96/100 design.

The objective of the debris filter is to prevent the most troublesome debris from entering the fuel bundles, and thus, to minimize the risk of fretting damage. The SVEA-96 debris filter is shown in Figure 2.14. [Proprietary Information Deleted]

Full-scale hydraulic testing of the debris filter has been carried out using characterized debris of various sizes and types. [Proprietary Information Deleted]

Lead Fuel Assemblies with debris filters were inserted in European plants in 1992 and additional LFAs are scheduled for 1994. [Proprietary Information Deleted]

2.5.2 ABB Sn-Alloy Zirconium Liner

The probability of PCI failure with unlined 10x10 SVEA-96 fuel in most BWR applications is very low. Therefore, lined fuel has not been considered necessary in the SVEA-96 design to date for most applications.

The usefulness of lined fuel for 8x8 assemblies, including our SVEA-64 design, has been recognized for some time. The potential for enhanced corrosion of zirconium liner was recognized during the early development of a lined fuel rod for our 8x8 designs. Therefore, during the 1980's, ABB developed a tin-alloy liner with corrosion resistance similar to that of Zircaloy-2. This decision has been justified by recent industry experience which indicates that the presence of an unalloyed zirconium liner appears to substantially increase the probability of severe secondary failures.

The ABB Sn-alloy zirconium liner has been utilized extensively in our SVEA-64 design. Table 2.1 is a summary of the application of this liner to date. [Proprietary Information Deleted]

[Proprietary Information Deleted] testing of the current Sn-alloy liner cladding has been quite extensive, and it is considered to be fully qualified. [Proprietary Information Deleted]

Current plans call for operation of [Proprietary Information Deleted]

2.5.3 Zircaloy-2 Fuel Channels

Zircaloy-4 has traditionally been utilized for BWR channels. Over the past several years ABB has been improving the heat treatment of our [Proprietary Information Deleted]

2.5.4 SVEA-96+

[Proprietary Information Deleted]

2.5.5 Improved Cladding

It is our impression that a primary design goal in meeting the utility needs over the next ten years will be to provide designs allowing higher discharge burnups without compromising fuel reliability. High back-end costs in Europe are already generating substantial pressure to increase discharge burnups, and we anticipate similar pressures in the U.S. market. Therefore, we are continuing and expanding the high burnup development program which we have been engaged in over the past several years.

As part of this program, [Proprietary Information Deleted]

3 DESIGN CRITERIA

The principal objective of the SVEA mechanical design is to meet the acceptable fuel design limits of General Design Criteria (GDC) 10, the rod insertability requirements of GDC 27, and the core coolability requirements of GDC 35 (Reference 1.4). To accomplish these objectives the fuel is designed to meet the acceptance requirements outlined in Standard Review Plan (SRP), Section 4.2 (Reference 1.3), relative to:

1. No calculated fuel system damage for normal operation and anticipated operational transients, which includes no predicted fuel rod failure (defined as a breach of fuel rod cladding), fuel system dimensions remaining within operational tolerances, and fuel system functional capabilities not reduced below those assumed in the safety analysis; and

2. Retention of fuel coolability and control rod insertion when required during postulated accidents which includes retention of rod-bundle geometry with adequate coolant channels to permit removal of residual heat considering the potential for cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation and extreme co-planar fuel rod ballooning.

The mechanical integrity design criteria are provided in three categories in this document:

1. General design criteria to assure that all required fuel system damage, fuel rod failure, and fuel coolability issues are addressed for new assembly designs and design changes,

2. Specific design criteria for the assembly components other than fuel rods to assure that the general design criteria are satisfied, and

3. Specific design criteria for the fuel rods to assure that the general design criteria are satisfied.

Discussions of the design criteria are provided in those cases for which clarification is considered necessary.

3.1 Design Criteria, General

3.1.1 Normal Operation and AOOs

Criterion

The fuel assembly shall be designed to avoid fuel damage during normal operation including anticipated transients. The term "fuel damage" refers to fuel rod failure leading to release of radioactive material, mechanical failure of fuel assembly components, or gross geometric distortions which make the assembly unsuitable for continued operation.

Discussion

The goal is zero failures. The design approach to achieve zero failures is to identify and eliminate to the greatest extent possible all causes of failure by establishing conservative design criteria and confirming that these criteria are satisfied. Sections 3.2 and 3.3 provide fuel assembly mechanical Design Criteria for assembly components other than fuel rods and as for the fuel rods, respectively. These design criteria are provided for normal operations and Anticipated Operational Occurrences (AOOs) to assure that this general criterion is satisfied.

3.1.2 Accident Conditions

The fuel assembly shall be designed to avoid unacceptable damage and maintain coolability during design basis accidents. This general criterion is satisfied by meeting the following specific criteria:

3.1.2.1 Fuel Rod Mechanical Fracture

Criterion

Mechanical fracture refers to fuel rod failure caused by external loads such as hydraulic loads and earthquakes. The fuel assemblies must withstand these external loads without fracturing the fuel cladding or causing unacceptable distortions.

Discussion

The methodology for evaluating fuel assembly performance and in illustration the performance of the SVEA-96 assembly for mechanical fracture under seismic/LOCA external loads are described in Reference 3.1.

3.1.2.2 Fuel Coolability

Criterion

The fuel assembly design must be such that the fuel assembly retains its rod-bundle geometry with adequate clearances to permit removal of residual heat. In order to meet this general criterion, the following specific criteria are established:

1. Cladding Embrittlement is limited by requiring that the Peak Clad Temperature (PCT) during a postulated LOCA be less than 1204°C (2200 °F).

2. The fuel assembly design must be such that unacceptable melting, fragmentation, and dispersal of the fuel does not occur during a postulated Control Rod Drop Accident (CRDA). Specifically, the radially averaged peak fuel enthalpy must be less than 280 calories/gram during a postulated CRDA.

3. Fuel rod ballooning must be limited such that unacceptable flow blockage does not occur during a postulated LOCA.

4. The spacer grids must be such that large distortion or failure does not occur under a postulated seismic plus LOCA event.

Discussion

During normal operation and anticipated transients the maintenance of a coolable geometry is assured by the conformance with the design criteria in Sections 3.2 and 3.3.

The ABB methodology for evaluating fuel coolability during postulated LOCAs is described in Reference 3.2 and Reference 3.3.

The ABB methodology for evaluating the consequences of a BWR CRDA and in illustration the performance of the SVEA-96 assembly during a CRDA are described in Reference 3.4.

The ABB methodology for evaluation the consequences during a seismic plus LOCA event are given in Reference 3.1.

3.1.2.3 Clad Bursting

Criterion

Unacceptable rupture of the cladding shall not occur during a postulated LOCA.

Discussion

The ABB methodology for evaluating fuel rupture during postulated LOCAs is described in Reference 3.2 and Reference 3.3.

3.1.2.4 Excessive Fuel Enthalpy

Criterion

The number of fuel rods predicted to reach assumed fuel failure thresholds during a CRDA will be input to a radiological evaluation. The assumed failure thresholds are a radially averaged fuel rod enthalpy greater than 170 calories/gram at any axial location for zero or low power initial conditions, and fuel cladding dryout for rated power initial conditions.

Discussion

The ABB methodology for evaluating the consequences of a BWR CRDA and in illustration the performance of the SVEA-96 assembly during a CRDA are described in Reference 3.4.

The ABB methodology for treating dryout in a BWR is described in Reference 1.1, and the methodology for evaluating SVEA-96 Critical Power is described in Reference 3.5.

3.1.3 Evaluation Methodology

Criterion

The methodology utilized for evaluation of the fuel assembly and fuel rod mechanical performance of the assembly relative to the design bases will be provided to the NRC for review and approval.

Discussion

The policy of NRC review of design bases and evaluation methodology is identified in the SRP and is consistent with past practice.

3.1.4 New Design Features

Criteria

All new designs and design features will be evaluated with the methodology accepted by the NRC relative to the approved design bases.

Significant new design features will be tested prior to full reload application.

The NRC will be notified of the first application of new fuel designs prior to loading into a reactor. New fuel designs and design features will be provided to the NRC for information as supplements to this topical report.

Discussion

New design features will be tested with out-of-reactor prototype testing, with Lead Fuel Assemblies, or with a combination of both approaches. As illustrated in Section 7, ABB practice is to utilize Lead Fuel Assembly programs extensively to confirm satisfactory performance of new designs and design features.

3.1.5 Post-Irradiation Fuel Examination

Criterion

Sufficient post-irradiation fuel examination will be performed to confirm that the fuel, including fuel assemblies with new design features, are operating as expected.

Discussion

The post-irradiation surveillance program described in Section 9 has been fashioned to meet the guidance provided in the SRP. As illustrated by the extensive inspections of the SVEA-96/100 design to date discussed in Section 7, the primary thrust has been on a generic post-irradiation inspection program for the SVEA-96 design.

3.1.6 New Safety Issues

Criterion

Each new safety issue identified by ABB or the NRC, which is related to fuel, will be evaluated relative to the existing ABB design criteria and methodology to confirm that it is properly addressed. If the new issue is not properly addressed, new criteria or revised methodology will be submitted to the NRC for review.

3.1.7 Failure to Satisfy Criteria

Criterion

Any new fuel design feature which does not meet the approved design criteria will be submitted to the NRC for review.

Discussion

Any additional information required for the review of the non conforming feature will be submitted to assist NRC staff review.

3.1.8 Burnup

Criterion

Assembly and fuel rod burnups shall be limited. Burnup limits are based on operational experience or experimental data which are sufficient to demonstrate the satisfactory performance of the assemblies to those burnups or confirm the satisfactory application of the analytical models to those burnups.

Discussion

An important aspect of the ABB mechanical design evaluation methodology is the use of experimental and plant operating data to support analytical modeling and direct confirmation of adequate performance of the design to specific burnup values. ABB design burnup limits are established based on in-plant experience typically utilizing Lead Fuel Assemblies. Prototype ex-core testing is utilized to augment the in-reactor program in supporting analytical predictions with a firm experimental data base.

3.2 Design Criteria, Fuel Assembly Components

This section provides design criteria for fuel assembly components and combinations of components. Design criteria for the fuel rods themselves are provided in Section 3.3.

3.2.1 Compatibility with Other Fuel Types and Reactor Internals

Criterion

The external envelope and positioning of the fuel assembly shall be mechanically compatible with other fuel types as well as core components such as control rods, the fuel support piece, and the core grid. "Mechanical compatibility" is defined as that characteristic of the assembly which assures that the other fuel assembly types and the core components shall not damage or be damaged by the presence of the assembly. Compatibility must be maintained for the design life of the fuel.

The fuel assembly must also be compatible with plant fuel storage facilities and handling equipment.

3.2.2 Geometric Changes in the Assembly During Operation

Criterion

Changes in the geometry of the fuel assembly components must not cause unacceptable interferences or impair the performance of the assembly. Dimensional changes of the assembly and its components as a function of burnup must be included in the design analysis. The effects of irradiation induced growth of fuel rods and channels, growth resulting from loads, bowing, spring relaxation, and creep are included. The mechanical and thermal-hydraulic functions of the bundle must not be impaired by geometrical distortions. The design shall provide sufficient space for unrestricted growth to occur.

The design shall provide sufficient clearances to accommodate differential axial growth of the fuel rods for the design life of the assembly.

3.2.3 Transport and Handling Loads

Criterion

The assembly design shall be such that shipping and handling loads, including acceleration loads, do not cause damage to the fuel assembly. The spacer grids and fuel pellets shall not be significantly affected when transport and handling procedures are complied with.

3.2.4 Hydraulic Lifting Loads During Normal Operation and AOOs

Criterion

The maximum hydraulic lift loads on the assembly during normal operations and AOOs shall not exceed the hold down capability of the fuel assembly.

Discussion

Assembly lifting loads resulting from accident conditions are addressed in Reference 3.1.

3.2.5 Stress and Strain During Normal Operation and AOOs

Criterion

Mechanical failure of assembly components shall not occur. Assembly component dimensions must be maintained within operational tolerances, and functional capabilities shall not be reduced below those assumed in the safety analysis. This criterion is implemented by establishing design limits for stresses in accordance with Reference 1.2 to assure that failure does not occur and that component dimensions and functional capabilities remain within acceptable limits.

Discussion

Specific stress limits are based on the Reference 1.2. Strain limits are not identified specifically for components other than the fuel rod cladding but are implicit in the stress limits as well as the functional design requirements on compatibility and dimensional changes stated in Sections 3.2.1 and 3.2.2.

3.2.6 Fatigue of Assembly Components During Normal Operation and AOOs

Criterion

The design criterion on assembly component fatigue is that fatigue failure of assembly components shall not occur.

3.2.7 Fretting Wear of Assembly Components

Criterion

Fretting wear at contact points on the structural members of the assembly should be limited in an environment free of foreign material such that the function of the assembly is not impaired. No specific design limit is applied, but any significant component wear must be accounted for in evaluating the component relative to stress and fatigue limits.

Fuel rod failure due to fretting in an environment free of foreign material shall not occur.

Discussion

The primary fretting wear concern is fuel rod wear. However, this design criterion is also applied to the other assembly components to assure that this aspect is addressed in evaluating new designs and design changes.

This design criterion is primarily intended to provide that the design of the fuel rods and spacer grids shall be such that damaging wear is avoided, and failures due to fretting wear between fuel assembly components is precluded.

This design criterion does not address fretting wear due to foreign material in the reactor.

3.2.8 Corrosion of Assembly Components

Corrosion of structural assembly components must be accounted for when evaluating the functionality, stress, and dimensional design criteria.

The impact of corrosion products (crud) on assembly components should be limited to avoid undue radioactive contamination of the primary system.

Discussion

The impact of crud formation on the assembly components must also be addressed in the thermal-hydraulic evaluation. This effect is addressed in Reference 1.1.

3.2.9 Hydriding of Zircaloy Assembly Components other than Fuel Rods

Criterion

Hydriding of Zircaloy structural components should be limited to avoid unacceptable strength losses. The impact of hydriding on evaluated stresses in structural components shall be addressed.

3.3 Design Criteria, Fuel Rods

3.3.1 Rod Internal Pressure

Criterion

The design criterion for fuel rod internal pressure requires that the internal pressure of the fuel rod shall not exceed a value which would cause the outward cladding creep to increase the diametral fuel pellet-cladding gap. This value of fuel rod internal pressure is defined to be that internal pressure which causes the outward cladding creep rate to exceed the fuel effective swelling rate. This requirement is referred to as "the lift-off criterion."

Discussion

This criterion is based on the recognition that the physical phenomenon to be avoided is an increase in the pellet-to-cladding gap at high burnups which could cause a rapid fuel pellet temperature increase and fission gas release resulting from the thermal feedback mechanism associated with an increasing gap. This criterion is believed to meet the intent of the SRP guidance. The fuel rod internal pressure must be limited to avoid an increase in gap size which could cause positive thermal feedback and rapidly increasing pellet temperatures. The ABB criterion is considered to more directly address this issue than the requirement suggested in the SRP that fuel and burnable poison rod internal gas pressure remain below the nominal system pressure during normal operation.

3.3.2 Cladding Stresses

Criterion

Fuel rod stresses must be maintained within acceptable limits. This criterion is implemented by establishing design limits for stresses in accordance with Reference 1.2 to assure that failure does not occur and that stresses on the fuel rod remain within acceptable limits.

3.3.3 Cladding Strain

Criterion

The uniform (i.e. total effective) cladding strain should not exceed 1%. In this context, "uniform strain" is the elastic and plastic strain which can occur during normal operation and AOOs excluding the effects of steady-state creep down and irradiation growth.

In addition, the maximum permanent end-of-life cladding strain including the effects of cladding creep down shall be less than 2.5 percent.

Discussion

These criteria result from the requirement that the fuel rods shall not be damaged due to excessive fuel cladding strains. The 1% limit on cladding strain is in compliance with Section 4.2 of the SRP. The 2.5% end-of-life limit including the effects of creep and irradiation growth is an ABB criterion applied to avoid excessive deformation of the cladding.

3.3.4 Hydriding

Criterion

Clad hydriding from waterside and internal sources shall be maintained sufficiently low that premature cladding failure shall not occur due to hydrogen embrittlement.

Discussion

This design criterion augments the 1% uniform strain criterion by providing a limitation on the loss of ductility at high burnups. Excessive loss of ductility at high burnups could in principal allow fuel rod failure without exceeding the 1% uniform strain criterion. Limitation of the clad hydriding limits the loss of ductility associated with hydriding and eliminates this concern.

3.3.5 Cladding Corrosion

Criterion

Clad corrosion must be limited to assure that excessive cladding corrosion does not lead to premature fuel rod failures due to excessive metal thinning or excessive cladding temperatures. The effect of cladding corrosion shall be included in the thermal-mechanical evaluation of the cladding.

3.3.6 Cladding Collapse (Elastic and Plastic Instability)

Criterion

Cladding collapse shall not occur during the design life of the fuel rod. Cladding collapse, or "elastic and plastic instability", refers to the pressure across the tubing walls at which the cladding will buckle in the elastic and plastic ranges.

3.3.7 Cladding Fatigue

Criterion

Cladding fatigue shall not cause fatigue damage during normal operation and AOOs. The fatigue evaluation shall account for the effects of cladding corrosion.

3.3.8 Cladding Temperature

Criterion

Cladding overheating during normal operation and AOOs shall not cause fuel rod failure.

3.3.9 Fuel Temperature

Criterion

The maximum centerline pellet temperature shall remain below the melting temperature of the fuel during normal operation and AOOs.

3.3.10 Fuel Rod Bow

Criterion

Excessive fuel rod bowing shall be precluded for the design life of the fuel assembly. Fuel rod bowing shall be evaluated, and any significant impact shall be accounted for in the thermal and mechanical evaluation of the fuel rods and the assembly.

4 DESIGN METHODOLOGY AND APPLICATION

This section provides the ABB methodology for evaluation of fuel assembly mechanical integrity for normal operation and AOOs relative to the design criteria given in Section 3. The evaluation methodology for accident conditions is covered in References 3.1 through 3.4 and summarized in Reference 1.1.

An evaluation of the fuel assembly relative to the design criteria provided in Section 3 is performed for each plant application. If appropriate conditions such as plant operating conditions, burnup requirements, and assembly design do not change, a single evaluation can be applied to all cycles for a given plant for many of the criteria. Therefore, whenever possible, bounding conditions are assumed for a specific plant to accommodate conditions from cycle-to-cycle.

In addition to the methodology description, the ABB methodology described in this report is applied to the SVEA-96 design as an illustration. This illustration is provided to help the reader understand the methodology and to provide an indication of the margins relative to the design criteria inherent in the SVEA-96 design.

The sample design evaluation demonstrates that the criteria are satisfied up to a [Proprietary Information Deleted] These burnup values will be extended only as sufficient justification, such as operating data and ex-core test results, are obtained to justify use of the evaluation methodology to higher burnups.

This section is organized in the same manner as Section 3. The evaluation methodology and sample application to SVEA-96 are provided in Sections 4.2 and 4.3 for each of the specific criteria in the order in which they appear in Sections 3.2 and 3.3. Correspondence between the subsection numbers in Sections 3.2/4.2 and 3.3/4.3 have been maintained consistent. Supporting information in Section 4.3 which does not directly correspond to any criteria in Section 3.3 has been provided in section 4.3.0.

Mechanical Properties

The materials used in the SVEA-96 BWR fuel assembly are identified in Section 5. As indicated in Section 7, these materials are proven and have had extensive in-reactor experience in domestic and foreign BWRs.

The ABB practice is to utilize the best available mechanical property data for the various materials in the assembly for the design evaluations. The mechanical properties utilized in the design evaluations are based on open literature sources, such as those given in Reference 4.1, SVEA-96 materials specifications, ABB measurement data, and data provided by suppliers. The material properties for the fuel cladding and UO_2 and Gd_2O_3 fuel pellets used in the fuel rod performance evaluations are discussed in Reference 4.2. Typical properties currently used for the fuel assembly design evaluations are provided in Table 3.1.

[Proprietary Information Deleted]

Design Stress Intensities

Mechanical properties, such as those discussed in Table 3.1, are used to establish stress limits defined by the design bases for the design evaluations of the assembly and assembly components.

Stress limits are based on Reference 1.2. [Proprietary Information Deleted]

The design stress intensity, S_m, for [Proprietary Information Deleted]

The design stress intensity, Sm, [Proprietary Information Deleted]

$Rp_{0.2}$ is the 0.2% offset yield strength. [IProprietary Information Deleted]

The specified minimum tensile and yield strengths at material temperature are used unless specific data are available to support the use of less conservative values. For example, at the present time ABB is utilizing [Proprietary Information Deleted]

Design stress intensities, S_m, are shown in Table 3.1 and are derived in this manner and based on the mechanical properties which also are shown in Table 3.1.

The fuel assembly structural component stresses under accident conditions are evaluated using the methods outlined in Appendix F Reference 1.2. The stress intensities (S_m) are defined in accordance with the rules described above for normal operating and anticipated operational transient conditions. [Proprietary Information Deleted]

These limits need not be satisfied at a specific location if it can be shown that the design loadings do not exceed two-thirds of the test collapse load determined in compliance with Section III of the Reference 1.2.

Unless otherwise stated, stress intensities are calculated with the Tresca criterion specified in the Reference 1.2:

S=Maximum{$|\sigma_1-\sigma_2|$, $|\sigma_1-\sigma_3|$, $|\sigma_2-\sigma_3|$}, where the σ_i are the principal stresses.

Under certain circumstances, which are identified in the text, stress intensities are calculated with the Von Mises criterion:

$$S = 1/\sqrt{2}\ [(\sigma_1 - \sigma_2)^2 + (\sigma_1 - \sigma_3)^2 + (\sigma_2 - \sigma_3)^2]^{1/2}$$

Design Loads

Design loads are established for conservative evaluation of the assembly and fuel rod performance in a given application relative to each design basis to assure that the design basis is satisfied during service. Therefore, design loads are selected to bound the loads in service. Selection of design loads are discussed in the following sections as part of the methodology for evaluating performance relative to each of the applicable design bases.

4.1 Methodology for Evaluation of General Design Criteria

Section 4.2 contains the methodologies for evaluation of fuel assembly components other than an individual fuel rod as well as combinations of components for normal operation, AOOs, and handling and transportation. Section 4.3 contains the methodologies for evaluation of an individual fuel rod for normal operation and AOOs. The methodologies for evaluation relative to the accident design criteria identified in Section 3.1.2 and sample applications are given in References 3.1 through 3.4.

The design criteria in Sections 3.1.3 through 3.1.8 provide controls governing fuel assembly design evaluation. These controls are administrative, and identification of technical methods for their evaluation is not applicable.

4.2 Methodology and Application - Fuel Assembly Components

4.2.1 Compatibility with Other Fuel Types and Reactor Internals

Methodology

For each plant application of an ABB fuel assembly type (e.g. SVEA-96) and each application involving a mixed core with fuel other than that fuel assembly type (e.g. fuel manufactured by a different vendor), an evaluation is performed to confirm compatibility with other fuel types and reactor internals. Specifically, this evaluation addresses the following compatibility considerations for the design lifetime of the assembly:

1. Geometrical Compatibility with Other Fuel Types in the Core

A systematic evaluation of the relative positions of the ABB fuel assembly type and other resident adjacent fuel assembly types over the design life of both fuel assembly types is performed. [Proprietary Information Deleted]

2. Geometrical Compatibility with Control Rods and Detectors

Clearances to control rods and in-core detectors are evaluated for the design lifetime of the fuel. Satisfactory clearances to or interferences with control rods and detectors are specifically confirmed. [Proprietary Information Deleted]

Creep Deformation

[Proprietary Information Deleted]

Channel Bow

The effect of channel bow is explicitly included in evaluating clearances to control rods, in-core instrumentation and adjacent assemblies. The impact of channel bow on thermal performance is evaluated as discussed in Reference 1.1.

[Proprietary Information Deleted]

A feature of the ABB methodology for the treatment of channel bow is to utilize materials and manufacturing processes to minimize the impact of channel bow.

3. Geometric Compatibility with Other Core Components

The compatibility of the fuel assembly with the fuel support piece and upper core plate is specifically confirmed.

4. Geometric Compatibility with Storage Facilities

The available space in the new fuel storage facility is compared with the BOL envelope for the fuel assembly. The EOL envelope of the fuel assembly based on upper limit channel growth, channel bow, and channel bulge is compared with the available space in the spent fuel facility to confirm that discharged fuel dimensions will be compatible with the spent fuel racks.

5. Geometric Compatibility with Handling Equipment.

A complete review of site equipment and clearances relative to procedures for fuel assembly handling and channeling is performed for any new application prior to shipment. For example, the following items are checked to confirm compatibility with site handling equipment:

[Proprietary Informatic . Deleted]

Sample Application

This section contains an example of the methodology for evaluating compatibility in a mixed core by evaluating the SVEA-96 assembly in a C-lattice BWR/5 type plant equipped with 3810 mm (150-inch) active fuel. The resident fuel to which the SVEA-96 fuel must be compatible is referred to as the "non-SVEA" fuel assembly.

1. Geometrical Compatibility with Other Fuel Types in the Core

[Proprietary Information Deleted]

The SVEA-96 channel leaf spring provides a nominal force of [Proprietary Information Deleted] This corresponds to a stress of [Proprietary Information Deleted] which is well below the yield stress of [Proprietary Information Deleted] shown in Table 3.1. [Proprietary Information Deleted]

This example demonstrates the compatibility of the SVEA-96 assembly with the non-SVEA assembly over the design life of the assemblies. The conclusions regarding compatibility are typical of those for various non-SVEA fuel designs.

2. Geometrical Compatibility with Control Rods and Detectors

The SVEA-96 assembly and control rod orientation for a full core of SVEA-96 fuel in a C-Lattice plant is shown in Figure 2.3a. In Figure 2.3a, the in-core detectors are located below the intersection of the upper core grid plates and have a typical diameter of [Proprietary Information Deleted] The available minimum space for the detector is [Proprietary Information Deleted] when surrounded by SVEA-96 assemblies. The width of the control rod is 7 mm.

As noted above, the maximum SVEA-96 channel dimension on a side at BOL is [Proprietary Information Deleted] From Figure 2.3a, this maximum dimension provides at least [Proprietary Information Deleted] Therefore, adequate clearances are available at BOL to avoid interference.

The effects of irradiation on the SVEA-96 channel dimensions and the resulting effects on compatibility with the control rods and detectors are considered by evaluating the channel bulge and bow.

Channel Bulge

The following example indicates the impact of channel bulge due to the pressure differential across the channel to a bundle burnup [Proprietary Information Deleted]

The SVEA channel has very favorable creep properties. The support of the channel walls by the water cross reduces creep deformation and stresses associated with deformation. [Proprietary Information Deleted]

Channel Bow

[Proprietary Information Deleted]

Therefore, this example demonstrates compatibility of the SVEA-96 assembly with BWR/4 and BWR/5 control rods and detectors. Similar compatibility evaluations are performed for each new plant application and have demonstrated compatibility with other plant types including BWR/6 units and D-lattice plants.

3. Geometrical Compatibility with Other Core Components

Compatibility with the fuel support piece is assured by the design of the lower nozzle which has been specifically designed to match the fuel support piece design in U.S. BWRs.

[Proprietary Information Deleted]

When it is required, custom design changes to the channel are made to assure proper orientation. For example, some plants are equipped with an upper core grid with a larger internal span than the standard C-lattice upper core grid and a C-lattice lower core plate. Under these circumstances, an assembly equipped with the standard channel appropriate for a "pure" C-lattice plant would tilt. [Proprietary Information Deleted]

In this manner compatibility of the SVEA-96 assembly with the upper core grid and fuel support piece is assured.

4. Geometric Compatibility with Storage Facilities

[Proprietary Information Deleted]

4.2.2 Geometric Changes in the Assembly During Operation

Methodology

For each plant application of an ABB fuel assembly type (e.g. SVEA-96), an evaluation is performed to confirm that the assembly and assembly components will not experience dimensional changes which will impair the performance of the assembly. The scope of this evaluation can depend on the assembly design. [Proprietary Information Deleted]

3. The following assembly components are evaluated to assure that their intended function is maintained during operation in the reactor and effects associated with operation in the reactor do not adversely affect assembly performance during the design life of the assembly:

   a. Upper and Lower Tie Plates

      [Proprietary Information Deleted]

   b. Assembly Handle Configuration

      [Proprietary Information Deleted]

   c. Spacer Capture Rod

      [Proprietary Information Deleted]

   d. Spacer

      [Proprietary Information Deleted]

   e. External Compression Spring

      [Proprietary Information Deleted]

A feature of the ABB methodology when applied to ABB designs to avoid unacceptable interactions of assembly and assembly components is to utilize materials for which excessive relaxation, growth or differential growth is avoided. Proven corrosion-resistant materials are utilized for all components to the greatest extent possible. Continuing post-irradiation examinations are utilized to conf. m or update expected performance of components with burnup and identify any adverse trends which could impact performance.

For non-ABB designs, publicly available information or data obtained from the fuel vendor or the utility are utilized. The level of conservatism in the application of these data is based on the quality and completeness of the data.

Sample Application

This section contains an example of the methodology for evaluating the interference of SVEA-96 assembly components as a function of burnup. [Proprietary Information Deleted]

1. Subbundle Growth

   The differential growth between the SVEA-96 channel and subbundles based on the most current data base can be summarized as follows:

   [Proprietary Information Deleted]

2. Differential Fuel Rod Growth

   An application of the methodology for evaluating the differential growth of the fuel rods based on typical rod growth data can be summarized as follows:

   [Proprietary Information Deleted]

   b. End Plug Extension Engagement

      [Proprietary Information Deleted]

3. Performance of upper and lower tie plates, assembly handle configuration, spacer capture rod, spacer, and external compression springs:

   a. Upper and Lower Tie Plates

      [Proprietary Information Deleted]

   b. Assembly Handle Configuration

      [Proprietary Information Deleted]

   c. Spacer Capture Rods

      The spacer-capture function must not be impaired for the lifetime of bundle by hydraulic forces, neutron irradiation, or corrosion.

      [Proprietary Information Deleted]

   d. Spacer

      [Proprietary Information Deleted]

      Spacers with the same general design and the same material as the SVEA-96 spacer have been used in 8x8 assemblies and in SVEA-64 assemblies. Reactor experience with over 11,500 assemblies has not shown any indication of stress corrosion cracking or fatigue failure. [Proprietary Information Deleted] Furthermore, laboratory tests described in Section 8 demonstrate that the spacer can withstand repeated seismic-type loads. [Proprietary Information Deleted]

Therefore, reactor experience with the SVEA-96 spacer, as well as very similar spacer designs for the 8x8 and SVEA-64 fuel, has confirmed that operation in the reactor will not impair the capability of the spacers to accomplish their function of maintaining the rod spacing during the design life of the fuel.

e. External Compression Spring

The same external compression spring design and spring material has been utilized in the 8x8 and SVEA-64 designs as well as in the SVEA-96/100 assemblies. Therefore, as indicated in Section 7, experience with these springs has been extensive. [Proprietary Information Deleted] Therefore, reactor experience with the SVEA-96 external compression springs has confirmed that operation in the reactor will not impair the capability of the springs to accomplish their function of maintaining the spacing between the end plug shoulders and the upper tie plate.

4.2.3 Transport and Handling Loads

Methodology

For each ABB fuel assembly type, an evaluation is performed to confirm that the assembly and assembly components will not be damaged during transportation or handling at the plant site.

[Proprietary Information Deleted]

Shipping

Special over-the-road shipping tests are performed to confirm that damage to the fuel assembly will not occur for loads less than the design shipping load. These tests are performed under the following circumstances:

[Proprietary Information Deleted]

Handling

A stress evaluation is performed for assembly components which experience potentially limiting loads during handling operations. The potential impact of thinning due to corrosion is included in the evaluation.

Stresses induced by these loads are compared with stress intensity limits (S_m) established in accordance with Reference 1.2. [Proprietary Information Deleted]

Sample Application

The current design loads for shipping and handling of SVEA-96 fuel for U.S. applications can be summarized as follows:

[Proprietary Information Deleted]

Sample Evaluation of Response to Shipping Loads - SVEA-96

Shipping tests have been performed in both the U.S. and Europe for the current shipping method for SVEA-96 assemblies [Proprietary Information Deleted]

The routes for these tests were selected to be representative of challenging roads over which shipments could pass. [Proprietary Information Deleted]

Sample evaluation to Response to Handling Loads - SVEA-96

The evaluation of the SVEA-96 assembly for design handling loads addresses the stresses in the channel assembly, the lifting handle, the tie plates, and the tie rods.

Channel

[Proprietary Information Deleted]

Handle

[Proprietary Information Deleted]

Tie Plates

[Proprietary Information Deleted]

Tie Rods

[Proprietary Information Deleted]

Therefore, margins to very conservative stress limits for the tie rods during handling operations are substantial.

4.2.4 Hydraulic Lifting Loads During Normal Operation and AOOs

Methodology

Hydraulic lift loads on the assembly during normal operation and AOOs are evaluated to assure that vertical liftoff forces are not sufficient to unseat the assembly bottom nozzle from the fuel support piece. The impact of these hydraulic lift loads on the subbundles are also evaluated to confirm that they are insufficient to unseat the subbundles from the lower support piece in the bottom nozzle. The methodology for addressing this circumstance under accident conditions (seismic/LOCA loads) is discussed in Reference 3.1.

[Proprietary Information Deleted]

Sample Application

[Proprietary Information Deleted]

4.2.5 Assembly Stress and Strain During Normal Operation and AOOs

A stress evaluation is performed for assembly components which experience potentially limiting loads during normal operation and AOOs. [Proprietary Information Deleted]

Sample Application

The sample application provided is for SVEA-96 assemblies in a 764-assembly BWR/5.

Stresses in SVEA-96 fuel assembly components have been evaluated for operating loads during normal operation and AOOs for a wide variety BWR plants. [Proprietary Information Deleted]

Spacer and External Compression Springs

[Proprietary Information Deleted]

Channel

[Proprietary Information Deleted]

It is concluded that the stress limits are met at both BOL and EOL conditions. It is also concluded that the deflections are small and are negligible with respect to a potential impact on function of the assembly. The design requirements are therefore met.

4.2.6 Fatigue of Assembly Components

Methodology

Each assembly design is evaluated for each plant application to identify any components which could experience damage or fail as a result of fatigue during normal operation and AOOs. A fatigue analysis is performed for each of the components for which there is a potentially adverse impact due to fatigue for each unique plant application. [Proprietary Information Deleted]

Component stresses are calculated for the assumed loads. Alternating stress intensities are established from the calculated stresses in accordance with the guidance in the Reference 1.2. The fatigue usage factor for the i^{th} load cycle is given by $\frac{n_i}{N_i}$, where:

n_i = number of cycles for the i^{th} load cycle,

N_i = the allowed number of cycles for the i^{th} load cycle from Reference 4.3 or from specific test data obtained and evaluated in accordance with Reference 1.2. Therefore, N_i includes the more limiting of a factor of two on stress and a factor of 20 on the number of cycles as well as the effects of non-zero mean stress.

$$\text{Cumulative Usage Factor} = \sum_{1}^{m} \frac{n_i}{N_i}$$

where m is the number of load cycles.

The Cumulative Usage Factor must be less than 1.0. The potential impact of thinning due to corrosion is included in the evaluation. Mechanical test results or operational experience may be utilized in place of, or to augment, the fatigue analysis to confirm satisfactory response to operational loads.

Sample Application

The only SVEA-96 components which experience appreciable fatigue loads during normal operations and AOOs are the fuel rods and the channel. The fuel rods are addressed in Section 4.3, and this section provides a sample evaluation for the SVEA-96 channel.

[Proprietary Information Deleted]

4.2.7 Fretting Wear of Assembly Components

Methodology

The assembly components are evaluated for their potential for fretting wear during normal operations and AOOs, and strategies for avoiding wear in any component with the potential for fretting wear are implemented.

[Proprietary Information Deleted]

Sample Application

The potential for damaging wear in the SVEA-96 design has been minimized by retaining features from previous designs for which the effectiveness in minimizing wear has been demonstrated. In addition, both SVEA-96 prototype loop tests and post irradiation examinations of SVEA-96 fuel have demonstrated that wear of SVEA-96 components is minimal and does not impair the function of the assemblies.

[Proprietary Information Deleted]

4.2.8 Corrosion of Assembly Components

Methodology

The methodology for minimizing and treating fuel rod cladding corrosion is addressed in Section 4.3.5. The methodology for treatment of corrosion in the remaining assembly components is provided in this section.

The assembly components are evaluated for their corrosion potential, and measures for avoiding excessive corrosion which could cause an unacceptable impact on the mechanical or thermal-hydraulic performance of the assembly are implemented as required. [Proprietary Information Deleted]

The impact of corrosion products (crud) on radioactive contamination of the primary system assembly components is limited to the extent that this buildup is affected by the design of assembly components.

The ABB methodology for minimizing the impact of corrosion and evaluating its effect on assembly components of ABB design is as follows:

1. [Proprietary Information Deleted]

Evaluation of the potential for component corrosion in non-ABB fuel is based on test data and post irradiation examination results for that fuel provided by the utility or the fuel vendor.

Sample Application

Based on industry data and ABB experience with the component materials used in the SVEA-96 design (Section 5.2.2), the SVEA-96 assembly components for which the potential for corrosion must be specifically addressed are:

1. [Proprietary Information Deleted]

[Proprietary Information Deleted] A summary of the operating experience and recent inspections are provided in Section 7.

Corrosion of the fuel rod cladding is addressed separately in Section 4.3.5. [Proprietary Information Deleted]

Assembly component corrosion is also maintained at a low level to keep the contribution to coolant activity by the assembly at a level which is as low as reasonably achievable. A related program to meet this goal is utilization of low-cobalt material. ABB has maintained an ongoing program over the past 20 years to minimize cobalt concentration in core components, including fuel assembly components, as a means of reducing personnel exposures. Particular emphasis has been placed on reducing cobalt concentrations in those components which represent relatively large potential sources of cobalt to the coolant. As a result, cobalt concentrations in ABB fuel assembly components are maintained at a relatively low level as shown in the following table.

[Proprietary Information Deleted]

4.2.9 Hydriding of Zircaloy Assembly Components other than Fuel Rods

Methodology

The methodology for treating fuel rod cladding hydriding is addressed in Section 4.3.4. The methodology for treatment of hydriding in the remaining Zircaloy assembly components is provided in this section.

[Proprietary Information Deleted]

The following measures are taken to minimize the impact of Zircaloy hydriding and to support the evaluation of its effect on structural assembly components for assemblies of ABB design:

1. [Proprietary Information Deleted]

Evaluation of the potential for hydriding of Zircaloy in non-ABB fuel is based on test data and post irradiation examination results for that fuel provided by the utility or the fuel vendor.

Sample Application

[Proprietary Information Deleted]

4.3 Methodology and Application - Fuel Rods

This section contains the methodologies for evaluation of the individual fuel rods in the assembly for normal operation and AOOs. Sections 4.3.1 through 4.3.11 describe the methodologies and provide a specific application to SVEA-96 for evaluation relative to the design criteria described in Sections 3.3.1 through 3.3.10. In addition, Section 4.3.11 is a discussion of ABB measures to minimize the probability of Pellet Cladding Interaction (PCI).

4.3.0 Fuel Rod Power Histories

Evaluation of the fuel rods for compliance with some of the design criteria in Section 3.3 requires the assumption of a specific fuel rod power history. Therefore, ABB has established a systematic approach for assuring that the most limiting power distributions for normal operations and anticipated AOOs are considered.

Methodology

[Proprietary Information Deleted]

Sample Application

Design Power History

[Proprietary Information Deleted]

Plant- and Cycle-Specific Power Histories

[Proprietary Information Deleted]

The sample applications shown in Section 4.3.1 through 4.3.10 which depend on fuel rod power history utilize the DPH and these limiting plant-specific fuel rod power histories.

4.3.1 Rod Internal Pressure

Methodology

For each plant application, End-of-life (EOL) fuel rod internal pressures are calculated using a fuel performance code accepted for referencing in licensing applications by the NRC. [Proprietary Information Deleted]

The upper limit EOL fuel rod pressure encompassing all significant uncertainties is compared with the minimum pressure required to achieve lift-off.

Sample Application

[Proprietary Information Deleted]

4.3.2 Cladding Stresses

Methodology

For each plant application detailed stress analysis of the fuel rod is performed [Proprietary Information Deleted] The analysis is performed to confirm that failure will not occur and that stresses in the fuel rods are within design limits defined in accordance with the ASME Boiler and Pressure Vessel Code, Section III (Reference 4.2).

[Proprietary Information Deleted]

Design stress intensities are established in accordance with the ASME Boiler and Pressure Vessel Code (B&PV), Section III, as described in Section 4. These design stress intensities are compared with maximum stress intensities computed according to the Tresca criterion as described in Section 4.

[Proprietary Information Deleted]

Sample Application

[Proprietary Information Deleted]

As discussed in Section 4, minimum yield and tensile strengths were utilized to establish the ASME limits. These minimum values are about one-half of the actual best estimate values introducing a further conservatism of about a factor of two. Therefore, it is concluded that margins to stress limits for the SVEA-96 fuel rod will be acceptable in any credible BWR application.

4.3.3 Cladding Strain

Methodology

For each plant application, peak cladding strain is evaluated as a function of fuel rod burnup for the design life of the cladding using a fuel performance code accepted for referencing in licensing applications by the NRC. [Proprietary Information Deleted]

Sample Application

[Proprietary Information Deleted]

4.3.4 Hydriding

Methodology

The methodology for treating hydriding of assembly components other than the fuel rod cladding is addressed in Section 4.2.9. The methodology for treatment of hydriding in fuel rod cladding is provided in this section.

The level of hydriding during the design life of the fuel rod is established. [Proprietary Information Deleted]

Evaluation of the potential for cladding corrosion in non-ABB fuel is based on test data and post irradiation examination results provided by the utility or the fuel vendor.

Sample Application

[Proprietary Information Deleted]

Zircaloy cladding accumulates hydrogen during BWR reactor operation. This hydrogen pick-up leads to the formation of zirconium hydride. The main source of hydrogen in the cladding is the corrosion reaction of zirconium and water. A secondary potential source of hydrogen is moisture or hydrogen inside the fuel tube.

[Proprietary Information Deleted]

No failures due to [Proprietary Information Deleted]

4.3.5 Cladding Corrosion

Methodology

The methodology for minimizing and treating the corrosion of assembly components other than the fuel rod cladding is addressed in Section 4.2.8. The methodology for treatment of corrosion of the fuel cladding is provided in this section.

The fuel rod cladding is evaluated for the potential for corrosion for each plant application for the design life of the cladding. In addition, measures for avoiding excessive corrosion which could cause an unacceptable impact on the mechanical or thermal-hydraulic performance of the cladding are implemented as required. [Proprietary Information Deleted]

The ABB methodology for minimizing the impact of corrosion and evaluating its effect on fuel rod performance for the ABB-designed fuel assemblies is as follows:

1. [Proprietary Information Deleted]

Sample Application

[Proprietary Information Deleted]

4.3.6 Cladding Collapse (Elastic and Plastic Instability)

Methodology

For each plant application, cladding collapse is evaluated as a function of fuel rod burnup for the design life of the cladding using cladding collapse methods accepted for referencing in licensing applications by the NRC.

[Proprietary Information Deleted]

Sample Application

It should be emphasized that cladding collapse is a highly improbable event since the occurrence of open axial gaps between the pellets is very unlikely. The high resintering stability of modern fuel prevents this effect. [Proprietary Information Deleted]

The current design limits for SVEA-96 fuel can be summarized as follows:

[Proprietary Information Deleted]

Utilizing the data in Section 5 and specification tolerances, the following parameters were selected to represent bounding conditions for cladding collapse:

[Proprietary Information Deleted]

4.3.7 Cladding Fatigue

Methodology

For each plant application, clad fatigue is evaluated for the design life of the cladding. The effect of clad fatigue is calculated for alternating stress on the cladding resulting from [Proprietary Information Deleted]

Alternating stress intensities are calculated in accordance with Reference 1.2. A Zircaloy fatigue design curve based on the work by

O'Donnel and Langer in Reference 4.3 is used to calculate the fatigue usage factors. This design fatigue curve includes the more conservative of a factor of two on the stress amplitude or 20 on the number of cycles. The sum of individual usage factors represents the cumulative usage factor over the life of the fuel rod. The calculated cumulative usage factor must be less than 1.0 for the design life of the fuel.

Fatigue Due to Fuel Rod Power Changes

Clad fatigue due to fuel rod power changes is evaluated for the design life of the cladding using a fuel performance code accepted for referencing in licensing applications by the NRC.

[Proprietary Information Deleted]

The VIK-2 code described in Reference 4.2 is utilized to calculate the bending stresses associated with the amplitudes calculated with this correlation. [Proprietary Information Deleted]

Sample Application

[Proprietary Information Deleted]

Example of Fatigue Calculation Due to Fuel Rod Power Changes

[Proprietary Information Deleted]

The results of these calculations and the cumulative usage factors can be summarized as follows.

[Proprietary Information Deleted]

These results demonstrate that, even with the very conservative assumptions utilized, the SVEA-96 fuel rod is characterized by very wide margins to fatigue failure for power changes for any plausible plant situation.

[Proprietary Information Deleted]

4.3.8 Cladding Temperature

Methodology

The ABB methodology for evaluating the potential for cladding failure due to overheating follows the traditional practice of assuming that failures will not occur if adequate margin to boiling transition is maintained. Margin to boiling transition is addressed in terms of the minimum critical power ratio (MCPR) as discussed in Reference 1.1.

The MCPR correlation for SVEA-96 fuel is documented in Reference 3.5.

4.3.9 Fuel Temperature

Methodology

For each plant application, fuel pellet temperatures are calculated at BOL and as a function of fuel rod burnup. Fuel pellet temperatures are calculated using a fuel performance code accepted for referencing in licensing applications by the NRC.

[Proprietary Information Deleted]

Sample Application

[Proprietary Information Deleted]

The results of these calculations can be summarized as follows:

[Proprietary Information Deleted]

As shown in this table, substantial margins to fuel melt are available for the SVEA-96 assembly. This is primarily due to the 10x10 design which distributes the bundle power over a relatively large number of fuel rods. Substantial margin to fuel melt is available even for the [Proprietary Information Deleted]

4.3.10 Fuel Rod Bow

Methodology

The potential for bowing of the fuel rods is evaluated to confirm that excessive bowing shall not occur during the design life of the fuel. Excessive bowing is defined as that degree of fuel rod bowing which leads to fuel rod damage or significantly impacts the nuclear or thermal-hydraulic performance of the assembly.

The assembly is evaluated to identify the potential for rod bow during the design life of the fuel for each plant application. [Proprietary Information Deleted]

Evaluation of the potential for fuel rod bow in non-ABB fuel is based on test data and post irradiation examination results for that fuel provided by the utility or the fuel vendor.

Sample Application

Features are specifically incorporated into the SVEA-96 design to preclude fuel rod bow. Based on ABB experience, as well as PWR and

BWR industry experience, the following phenomena are believed to be the prime contributors to fuel rod bow:

[Proprietary Information Deleted]

As discussed in Section 7, ABB maintains a very aggressive post irradiation examination program. [Proprietary Information Deleted]

4.3.11 Pellet-Cladding Interaction

Methodology

As stated in the Standard Review Plan (Reference 1.3), there is no specific NRC criterion for fuel failure due to Pellet-Cladding Interaction (PCI). In accordance with the guidance in the SRP, design criteria limiting the uniform cladding strain to 1% (Section 3.3.3) and precluding fuel melting (Section 3.3.9) are applied which reduce the potential for fuel failure due to PCI. No specific design criterion in addition to these criteria is applied to PCI. [Proprietary Information Deleted]

Sample Application

The sample application is for the SVEA-96 fuel rod used in BWR/2-6 plants.

[Proprietary Information Deleted]

5 TECHNICAL DATA

The data in this table is typical for BWRs. Differences between the listed values and that for specific plants are expected to be minor. For example, bundle mass will change as the Gd_2O_3 design changes.

5.1 Fuel Rods

5.1.1 Pellets

5.1.1.1 Pellet Dimensions

A. UO2 pellets for natural uranium blankets

[Proprietary Information Deleted]

B. UO_2 and gadolinia pellets for enriched parts of the rods

[Proprietary Information Deleted]

5.1.1.2 Pellet Data

[Proprietary Information Deleted]

5.1.1.3 Pellet Densification

[Proprietary Information Deleted]

5.1.1.4 Burnable Poison Pellet

ABB utilizes gadolinia (Gd_2O_3) as a burnable poison. The pellets are a mixture of Gd_2O_3 and UO_2 with a [Proprietary Information Deleted]

5.1.2 Fuel Rod Cladding

5.1.2.1 Cladding Dimensions

[Proprietary Information Deleted]

5.1.2.2 Cladding Chemical and Physical Properties

[Proprietary Information Deleted]

5.1.3 Fuel Rod Length

[Proprietary Information Deleted]

5.1.4 Fuel Rod Miscellaneous Data

[Proprietary Information Deleted]

5.1.5 Fuel Rod Materials

[Proprietary Information Deleted]

5.1.6 Typical Fuel Rod Weights

[Proprietary Information Deleted]

5.1.7 Spacer Grid

[Proprietary Information Deleted]

5.1.8 External Spring

[Proprietary Information Deleted]

5.2 Fuel Assembly Data

5.2.1 Fuel Assembly Miscellaneous Data

[Proprietary Information Deleted]

5.2.2 Fuel Assembly Materials

[Proprietary Information Deleted]

5.2.3 Typical Fuel Assembly Weights

[Proprietary Information Deleted]

6 CODE DESCRIPTION

This section contains a brief description of the computer codes used by ABB in mechanical design calculations. More detailed descriptions of the fuel rod design codes are contained in Reference 4.2.

The VIK-2 code is a collection of the models and formulae used to calculate cladding stresses at the beginning of life. STAV6.2 is the principal code for fuel performance analysis. COLLAPS-II is used to calculate cladding ovality as a function of irradiation.

ABB utilizes the finite element code ANSYS for stress analysis of the SVEA-96 fuel assembly. This code is well known in Europe and the U.S., and has been used routinely for reactor design and analysis.

6.1 VIK-2

VIK-2 calculates the cladding stresses at the beginning-of-life (BOL) for a fuel rod. The code consists of subroutines which calculate different stresses on the cladding. These individual stresses are subsequently added according to the Tresca rule or the "Von Mises" rule and compared to the allowed stresses specified by the appropriate design criterion. Standard analytical expressions are used to calculate the stresses. The source of each stress component and the model used to calculate it can be summarized as follows:

<u>Cladding Internal and External Pressure</u>

The stresses caused by loading of a fuel rod by internal gas pressure and external coolant pressure are calculated. [Proprietary Information Deleted]

<u>Pressure at End Plug</u>

Stress components caused by the pressure at the end plug of a fuel rod are calculated. [Proprietary Information Deleted]

<u>Ovality</u>

The initial ellipticity of the cladding under uniform external pressure gives rise to tangential and axial stresses in the fuel cladding. [Proprietary Information Deleted] The model assumes that there is an initial deviation from a perfect circular form in the shape of the cladding tube. Upon loading the non-circular tube with pressure, the further flattening of the tube as a result of pressures on the tube is calculated.

Radial Temperature Gradient

The stresses caused by a radial temperature distribution within the cladding are calculated. [Proprietary Information Deleted]

Azimuthal Temperature

VIK-2 includes a model for calculating the effect on the cladding temperature distribution of asymmetric positioning of the fuel pellets.

Springs

The axial stresses on the fuel rod caused by the internal and external springs are calculated.

Rod Bending

The stresses exerted on the fuel rods caused by flow-induced vibrations are calculated. The model describes the rod as a straight beam clamped at both ends.

Spacer Grid

The stresses applied to the cladding by the spacer grids consist of three components: spacer membrane, spacer bending, and spacer beam bending. The spacer membrane and spacer bending stresses are caused by the spacer springs. The spacer beam bending stresses arise from the bending in a portion of the rod between the spacer and supports. The model describes the cladding and the spacer as cylindrical shells with closed ends supported at the ends.

End Plug Temperature Gradient

The heat transferred from the UO_2 pellet to the bottom end plug causes thermal expansion of this plug. This expansion loads the cladding on the circumferential bottom end plug weld. The axial and radial temperature gradients in the fuel rod are calculated, [Proprietary Information Deleted]

End Plug Angle

Misalignment of the holes in the tie plates and the end plug extensions combined with maximum unfavorable tolerances in tie plates can be postulated to lead to a bending moment in the fuel rods. [Proprietary Information Deleted]

6.2 STAV6.2

The STAV6.2 code is used by ABB for BWR and PWR fuel rod performance analyses. This report addresses the application of

STAV6.2 in the United States for BWR applications only. STAV6.2 is used in Europe for both BWR and PWR applications. STAV6.2 offers a best-estimate analytical tool for predicting steady-state fuel performance for operation of Light Water Reactor (LWR) fuel rods including UO_2-Gd_2O_3 fuel.

STAV6.2 calculates the variation with time of all important fuel rod performance quantities including fuel and cladding temperatures, fuel densification, fuel swelling, gaseous fission product release, rod pressure, gas gap conductance, cladding stresses and strains due to elastic and thermal creep and plastic deformations, cladding oxidation, and cladding hydriding. Burnup-dependent radial power distributions for both UO_2 and UO_2-Gd_2O_3 fuel, fuel grain growth, and helium release are modeled in the code.

The fuel rod geometric parameters, the actual or projected irradiation history, and the core thermal and hydraulic conditions are the required input for initialization of the code.

Rod Geometric Parameters and Modeling

The fuel rod in STAV6.2 is a typical Light Water Reactor (LWR) fuel rod with an active fuel length consisting of fuel pellets enclosed in Zircaloy cladding. A plenum for accommodation of released fission gases is above the fuel stack.

The Zircaloy cladding is modeled as a tube concentric with the fuel pellet column. The cladding material can be either fully-recrystallized Zircaloy-2 or cold-worked and stress-relieved Zircaloy-4.

The active fuel length is separated into axial segments. The plenum region is treated independently as an additional node. The fuel pellets are right circular cylinders.

The code takes into account the void volume of the rod due to pellet dishing, chamfering, and stacking faults. The fuel rod can be pressurized or unpressurized. The fill gas can be any combination of helium, nitrogen, argon, and xenon. Complete and instantaneous mixing of gases in the fuel rod in the void volume is assumed.

Input Power and Fast Flux Histories

The fuel rod power history given by the local linear heat generation rate (LHGR), as a function of burnup or time, can be supplied either from the output of reactor physics codes or input directly. Fast neutron flux is calculated from the power history and a burnup or time-dependent input factor.

Thermal Hydraulic Parameters

The fuel rod pitch, coolant inlet temperature, coolant pressure, and coolant mass flow rates are supplied as inputs for the cladding outer surface temperature calculation.

The STAV6.2 calculational path starts from the coolant with thermal and hydraulic calculations and extends to cladding strain stress calculations and fission gas release calculations.

The heat transfer between the cladding and the coolant is modeled with either single phase convection, subcooled boiling, or saturated flow boiling. [Proprietary Information Deleted]

An important quantity in STAV6.2 is the heat transfer across the pellet-cladding gap and the pellet-cladding contact pressure. The model is phenomenological, is quite detailed, and is interactive with a pellet cladding mechanical model.

The fuel swelling model in STAV6.2 consists of empirical models for the contributions of: [Proprietary Information Deleted]

The fission gas release (FGR) model in STAV6.2 consists of an athermal FGR model and a thermal FGR model. The athermal FGR model accounts for the contribution to FGR by the knock-out process from regions close to surfaces of the pellet periphery and internal cracks. [Proprietary Information Deleted]

6.3 COLLAPS-II

COLLAPS-II is used for prediction of cladding ovality in BWR fuel rods as a function of irradiation time.

The COLLAPS-II code models the cladding as a long, thin cylindrical tube which is subject to creep as a result of a uniform net external pressure. The cross section of the tube is assumed to have a slight initial deviation from circularity. The standard assumptions appropriate to creep deformation analysis of shells are utilized in the COLLAPS-II code.

COLLAPS-II calculates the following quantities as a function of irradiation time:

- Cladding ovality,
- Creep down strain and total axial strain of the cladding, and
- Bending moments of the cladding.

6.4 ANSYS

ANSYS is a large-scale, general purpose finite-element code. The code's capabilities include:

- Static and dynamic structural analysis, with linear and nonlinear transient methods, harmonic response methods, mode-frequency method, modal seismic method, and vibration analysis,
- Buckling and stability analysis with linear and nonlinear buckling,
- Heat transfer analysis with transient capability and coupled thermal-structural capabilities,
- nonlinear material properties such as plastic deformation, creep, and swelling,
- Fracture mechanics analysis.

The ANSYS element library consists of 78 distinct element types. However, many have option keys for further element specialization, effectively increasing the size of the element library.

The reliability and accuracy of ANSYS software is maintained by a rigorous quality assurance program. A library of verification problems, now numbering over 2000, is used for verification of new versions, and is continuously updated to reflect new features in the program.

7 OPERATING EXPERIENCE

7.1 History

The evolution of the ABB BWR fuel designs is shown in Figure 7.1. ABB started out with an 8x8 lattice design instead of the 7x7 lattice, and then went directly to 10x10 instead of the intermediate 9x9 lattice. The trend towards longer cycles and higher burnups combined with plant uprating made 10x10 the optimum choice.

ABB started manufacturing and delivering 8x8 BWR fuel in 1967. First cores and reload quantities of 8x8 fuel have been delivered to all eleven ABB built BWR plants in Sweden and Finland. In addition, 8x8 Lead Fuel Assemblies have been delivered to two Siemens built plants. Fuel performance and reliability of the ABB 8x8 fuel has been excellent. The last 8x8 fuel was manufactured in 1987.

The second generation of ABB fuel designs, SVEA-64, has four 4x4 subbundles and a watercross in the center. Lead testing of SVEA-64 occurred from 1981 to 1985. Since 1984, SVEA-64 fuel has been delivered to nine ABB built plants, one GE plant and three Siemens plants.

The design of the top handle in the SVEA-64S fuel, which is used in Swedish and Finnish reactors, is slightly different from the SVEA-64C fuel used in non-ABB built reactors. These differences are required primarily to adapt the design to existing fuel handling equipment and core internals. Therefore, the experience gained from SVEA-64S fuel is also valid for SVEA-64C fuel. The SVEA-64C design with Zircaloy spacers was introduced to the U.S. as the QUAD+ assembly by Westinghouse.

The third evolutionary generation, SVEA-96/SVEA-100, has four 5x5 subbundles and a watercross using the same channel design as SVEA-64. SVEA-96 fuel is very similar to the SVEA-100 fuel. [Proprietary Information Deleted]

The other components in the SVEA-96 and the SVEA-100 designs are the same with the exception that SVEA-96 has four 5x5-1 subbundles versus the 5x5 subbundles for SVEA-100. [Proprietary Information Deleted] Therefore, operating experience gained with SVEA-100 is applicable to SVEA-96 as well.

Reload quantities of SVEA-96/100 have been delivered to three ABB built plants, two Siemens reactors, and one Swiss GE BWR/6 reactor. The deliveries of SVEA fuel are summarized in Figure 7.2 and in Table 7.1. This experience base is steadily increasing and as of November 1993, ABB has delivered eight reloads of SVEA-100 and seven reloads of SVEA-96 fuel, for a grand total of 2,100 SVEA-96/SVEA-100 fuel assemblies. As of February 1994, ABB has also accumulated orders for 44 more reloads including five in the United States.

7.2 Experience

7.2.1 SVEA-64

The first four SVEA-64 Lead Fuel Assemblies (LFAs) were loaded into the Ringhals 1 reactor in 1981. Two of these were discharged in 1987 after six years of operation with a burnup of [Proprietary Information Deleted], and the other two in 1988, also with a burnup of 35 MWd/kgU after their seventh cycle. In Oskarshamn 2, one SVEA-64 assembly reached [Proprietary Information Deleted] and another SVEA-64 assembly reached [Proprietary Information Deleted]. Since 1981, SVEA-64 assemblies have been loaded into Swedish reactors on an annual schedule. In 1985, SVEA-64 fuel was loaded into the Finnish reactor TVO II. Since 1986, SVEA-64 fuel assemblies have been loaded into the German reactors Krümmel, Philippsburg 1, Brunsbüttel, and the Swiss reactor Leibstadt. A survey of the SVEA-64 burnup statistics as of September 1993 is given in Table 7.2. As of September 15, 1993, a total of 5,804 SVEA-64 assemblies have been delivered.

7.2.2 SVEA-96/SVEA-100

In 1986, the first SVEA-100 Lead Fuel Assemblies were loaded, four into the Oskarshamn 3 and two into the Forsmark 3 reactors. Since 1986, additional SVEA-100/SVEA-96 Lead Fuel Assemblies have been loaded into Swedish reactors on an annual schedule. SVEA-96/100 Lead Fuel Assemblies also have been loaded into Finnish and German reactors. In 1990 the first full SVEA-100 reload consisting of 100 assemblies, was loaded into Oskarshamn-3. In 1993, the German reactors Philippsburg 1 and Isar 1, loaded reload quantities of SVEA-96. The Finnish reactor TVO II will received a reload of SVEA-100 fuel in 1993, which will be loaded into the core in 1994. In addition, the German Brunsbüttel reactor continues to load SVEA fuel transitioning to reload quantities of SVEA-96.

In 1990, 116 SVEA-96 fuel assemblies were delivered to the Swiss Leibstadt reactor (a General Electric BWR/6 plant). As of September 1993, 330 SVEA-96 assemblies have been loaded into the core. Since September 1993, an additional 120 SVEA-96 assemblies have been delivered to the Leibstadt reactor. Furthermore, Leibstadt has extended its orders for reload quantities of SVEA-96 through 1996.

In 1990 and 1991 sixteen SVEA-96 Lead Fuel Assemblies were installed in four U.S. GE BWR reactors. The first four U.S. LFAs were placed into the Supply System WNP-2 plant in 1990. Based in part on the success of the LFA program, the Supply System has ordered five reloads of SVEA-96 fuel starting with deliveries in 1996.

Lead Fuel Assemblies installed in 1991 are currently operating in Fermi 2, Peach Bottom 2, and Limerick 2 (all BWR/4s). Fuel Surveillance and Post Irradiation Examination programs are ongoing or planned to verify acceptable performance of the SVEA-96 fuel. There have been no problems encountered with fuel receipt, handling, or insertion at any of the sites.

The Spanish Cofrentes plant (a GE built BWR/6) will install four Lead Fuel Assemblies in 1994, which shall include the ABB debris filter (see Section 2.5.1).

A total of 58 reloads of SVEA-96/100 have been delivered or are on order (see Table 7.1). Over 20 percent of the reloads are for GE built reactors.

A survey of SVEA-100 and SVEA-96 burnup statistics, as of September 1993, is given in Table 7.3. As of November, 1993, a total of 2,100 SVEA-96/SVEA-100 have been delivered (see Table 7.1 and Figure 7.2). A total of 1568 SVEA-96/100 assemblies have been loaded into 13 different reactors.

7.3 Fuel Reliability

7.3.1 General

As can be seen on Table 7.4 and Figure 7.3, ABB BWR fuel performance over the period 1981 through 1993 has been excellent, with no fuel failures due to manufacturing problems or Crud Induced Localized Corrosion (CILC). Note that these data are based on failed fuel rods, not assemblies. It is ABB practice to identify the cause of all fuel failures to the greatest extent possible. To this end many of the failed rods have been taken to a hot cell for investigation. The majority of the remaining unidentified cases are believed to be debris failures.

7.3.2 8x8

Fuel performance for ABB 8x8 fuel has been good with the majority (71 percent) of the failures known to be debris related. Fuel reliability per cycle is 99.9982 percent when debris related failures are considered and is at least 99.9995 percent when known debris related failures are excluded from the data base. Many, if not all, of the unknown failures are also probably debris related, so that actual fuel performance for the 8x8 is even better than stated above.

Incorporating the PCI lessons learned with 7x7 fuel in the design of ABB fuel assemblies, in addition to conservative utility operation, helped prevent massive PCI failures in the 8x8 (or in any other) ABB fuel. High quality manufacturing techniques and processes used since early in ABB's history have also avoided substantial numbers of manufacturing related failures. CILC failures have also not occurred in ABB 8x8 fuel, [Proprietary Information Deleted]

7.3.3 SVEA-64

Fuel performance following the introduction of ABB SVEA-64 fuel remained excellent, even with uprating and extended operating flexibility introduced for most of the plants being supplied with ABB fuel. The majority of failures again are debris related. Fuel reliability per cycle for ABB 8x8 and SVEA-64 fuel is 99.9980 percent when all failures are considered and is at least 99.9992 percent when the known debris and dryout event related failures are removed from the data base. The unknown failures are suspected to be debris related. Hence the actual fuel performance is even better than stated above.

As can be seen in Table 7.4, four of the SVEA-64 failures were caused by the "Dryout Event" at Oskarshamn 2. [Proprietary Information Deleted]

7.3.4 SVEA-96/SVEA-100

One of the driving forces behind ABB's choice of the 10x10 array (SVEA-96/100) was increased fuel reliability via a substantial reduction in fuel rod duty. The logic behind the change with respect to fuel reliability is as follows:

- Investigations into fuel rod corrosion have found that corrosion rates are strongly heat flux dependent (the higher the heat flux, the faster corrosion builds up on the rods, thus degrading rod performance and shortening its dependable life).

- The total heat transfer surface area of a 10x10 array is 25 percent greater than in a 8x8 array; therefore, the heat flux for a SVEA-96 assembly will be about 75 percent of that for an 8x8 assembly. Computer simulations indicate a 40 percent reduction in corrosion buildup with the lower surface heat flux.

- The smaller diameter of the 10x10 fuel rod relative to 8x8 fuel results in lower fuel temperatures. The lower fuel temperatures associated with the 10x10 lattice results in much less fission gas release because the end-of-life internal rod pressure remains well below the coolant pressure even at very high burnups. This allows much higher discharge burnups (hence better economics) or inherently more margin to potential failure mechanisms when compared to 8x8 fuel.

- [Proprietary Information Deleted]

Fuel performance for ABB SVEA-96/100 fuel is consistent with the past experience of ABB 8x8 and SVEA-64. As of December, 1993, three failures have occurred in over 164,000 irradiated fuel rods. Two failures have been confirmed to be debris related and the other one is awaiting further investigations. Hence, the SVEA-96/100 fuel reliability per cycle is 99.9989 percent when all failures are considered and is at least 99.9994 percent when the known debris related failure is removed from the data base. The actual fuel performance of the SVEA-96/100 should be better, once the other failure is examined. This reliability for our advanced fuel design is based on operation of 1568 irradiated bundles in 13 reactors (six ABB, five GE, and two Siemens) from 1986 through the present.

7.3.5 Reliability Improvement

Even with the excellent record of ABB fuel, improvements in both actual performance and resistance to failures are desirable. [Proprietary Information Deleted]

7.4 Inspections

7.4.1 SVEA-64

ABB maintains ongoing post irradiation examination programs to confirm the acceptable operation of the fuel and identify potential design improvements. This section provides a discussion of the recent inspection program for SVEA fuel. Inspection programs of this scope are anticipated for the future as well and are discussed in Section 9.

Over 120 SVEA-64 assemblies have had detailed inspections performed during refueling outages since 1982. The poolside inspections and measuring programs have verified equipment and procedures for safe handling of irradiated SVEA fuel assemblies. In addition, a substantial operating data base has been established.

[Proprietary Information Deleted]

The results of these inspections indicate excellent fuel performance. The behavior of the SVEA-64 fuel assemblies is completely within expectations.

[Proprietary Information Deleted]

7.4.2 SVEA-96/SVEA-100

Inspections similar to those performed for SVEA-64 fuel have been done on the lead SVEA-96/100 fuel assemblies in each year from 1987 through 1993 (see Table 7.6). The results of the SVEA-96 fuel inspections show that the fuel is in excellent condition with little differential growth and no indications of fretting.

From a mechanical perspective, the SVEA-96 fuel is designed according to the same principles as the SVEA-64 fuel. [Proprietary Information Deleted] Therefore, the experience gained from operation of SVEA-64 supports that for SVEA-96.

As shown in Table 7.6, the first Post Irradiation Examination of SVEA-96 fuel in the U.S. was performed [Proprietary Information Deleted]

As shown in Table 7.6, two leading SVEA-100 assemblies have achieved a burnup of about [Proprietary Information Deleted] and were inspected in August 1993. This inspection verified good performance as predicted, with respect [Proprietary Information Deleted]

8 PROTOTYPE TESTING

ABB has a continuing program to perform prototype testing for all of their fuel assembly designs. Tests have been performed on the ABB 8x8 assembly, the SVEA-64 design, the SVEA-96 design, and the SVEA-100 design. The types of testing include seismic testing of the assemblies, strength tests on individual components, fretting tests, and hydraulic endurance and performance tests. This section describes some of the tests that have been performed which support the SVEA-96 design and design evaluation.

This information is provided to supplement the analytical and operating experience bases of the SVEA-96 design. A discussion of in-reactor experience, which includes inspection data from Lead Fuel Assemblies at various plants in addition to reload quantities of SVEA-96 assemblies, is provided in Section 7.

8.1 Fretting Tests

Full scale fretting tests have been performed on SVEA-96 fuel in the FRIGG loop at the ABB laboratory. The intent of this test was to verify that fretting damage would not occur under operating conditions. Conditions for the tests are described in the table below. [Proprietary Information Deleted]

The conclusions from the fretting tests are that the mechanical behavior of the SVEA-96 fuel is satisfactory and that reactor operation without failures caused by fretting can be expected.

8.2 Pressure Cycling Test

A pressure cyclic test has been performed [Proprietary Information Deleted]

8.3 Lateral Load Cycling Test, Channel and Spacer Grid

Lateral load cycling tests have been performed on [Proprietary Information Deleted]

The tests were performed at room temperature, and scaling factors are used to translate test results to operating conditions in accordance with ASME III, Appendix II-1520. The scaling factors include the effect of the temperature and irradiation as well as a statistical margin.

The tests have verified that the spacer grids and channel welds will withstand the following lateral seismic type loads at operating conditions without failure and with negligible deformation:

[Proprietary Information Deleted]

8.4 Spacer Capture Rod Test

The ability of the spacer capture rod to maintain the position of the spacers has been tested on SVEA-96 spacer capture rods and spacer grids.

A test has been performed to determine the spacer capture force for grid passage. [Proprietary Information Deleted]

8.5 Handle Tension Test

[Proprietary Information Deleted]

The handles were fastened to the tension testing machine with screws, fitting the holes for the channel screws and the upward force on the handle beam was applied by a simulated fuel grapple. These test results are also valid for the SVEA-96 handle since all the relevant dimensions are the same as for the SVEA-64C handle. [Proprietary Information Deleted]

8.6 Tension Test on Screw Mounted in Channel

[Proprietary Information Deleted]

8.7 Top Tie Plate Load Test

A test was performed to determine the loading at which permanent deformation of the top tie plate occurred and the amount of the deformation. The top tie plate was supported at the tie rod position and clamped to the table of the test machine. Vertical loads were then applied at the same positions as used by the normal handling of the fuel subbundle. The loads were applied underneath the top tie plate.

[Proprietary Information Deleted]

9 TESTING, INSPECTION, AND SURVEILLANCE PLANS

9.1 Testing and Inspection of New Fuel

Figure 9.1 is a diagram of the ABB-CENO (Combustion Engineering Nuclear Operations) quality assurance program. This figure shows the identification and hierarchy of the various program manuals, and procedures manual for controlling engineering and manufacturing of nuclear fuels. These documents are available for evaluation. CENPD-210A is a topical report which has been approved by the U.S. NRC.

ABB fuel manufacturing facilities are currently located in Windsor, Connecticut, Västerås, Sweden, and Hematite, Missouri BWR fuel pellets and assemblies can be manufactured in either Hematite or Västerås with some components manufactured in Windsor. [Proprietary Information Deleted]

9.1.1 Inspection and Testing Associated with Manufacturing

The specific manufacturing inspections and tests are continually updated to improve manufacturing processes and product quality. A general summary of typical inspections and tests performed as part of the fabrication process is provided to give an indication of the general scope and nature of manufacturing tests and inspections.

Fuel Rods

[Proprietary Information Deleted]

Fuel Subbundles

[Proprietary Information Deleted]

Fuel Channel

[Proprietary Information Deleted]

Handle

[Proprietary Information Deleted]

Fuel Assembly

[Proprietary Information Deleted]

9.2 On-Line Fuel System Monitoring

On-line monitoring is plant specific. It is addressed in the applicants FSAR.

9.3 Post-Irradiation Surveillance

As illustrated in Section 7, ABB considers inspection of ABB fuel assemblies a crucial aspect part of the goal to achieve zero failures. Specific post irradiation examination programs depend on the design and the application. A general overview is provided in this section.

[Proprietary Information Deleted]

The data from these examinations, plus historical records are collected, summarized, documented, stored and readily retrievable by ABB in Europe and the U.S. The information is available to users. Lessons learned are fed back into the design to improve the fuel performance, decrease the risk, and to reduce cost. ABB has performed fuel surveillance on irradiated SVEA-100 fuel in Swedish reactors during outages every year since 1987. The experience with SVEA-100 fuel is directly applicable to SVEA-96. Furthermore, ABB has performed examination of SVEA-96 fuel in ABB Nordic plants, Siemens plants in Germany, GE plants in Switzerland and the U.S. This work has included dismantling of SVEA assemblies and subbundles and inspection of fuel rods and spacer capture rods.

ABB has routinely inspected, and performed operations on 8x8 fuel since the early 1970's and on SVEA fuel since 1982. ABB has performed most of the fuel surveillance in Sweden and Finland.

Surveillance work may include any or all of the following:

[Proprietary Information Deleted]

Additional details on inspections of SVEA fuel is given in Section 7.4. This experience provides ABB with a very solid record of fuel performance.

10 REFERENCES

1.1 "Reference Safety Report for Boiling Water Reactor Reload Fuel," ABB Report CENPD-300-P-A (proprietary), CENPD-300-NP-A (non-proprietary), July 1996.

1.2 ASME Boiler and Pressure Vessel Code, Section III.

1.3 "Fuel System Design," U.S. NRC Standard Review Plan Section 4.2, NUREG-0800, Rev. 2, July 1981.

1.4 Code of Federal Regulations, Section 10, Energy, Part 50, Appendix A.

3.1 "ABB Seismic/LOCA Evaluation Methodology for Boiling Water Reactor Fuel," ABB Report CENPD-288-P-A (proprietary), CENPD-288-NP-A (non-proprietary), July 1996.

3.2 "Boiling Water Reactor Emergency Core Cooling System Evaluation Model," ABB Reports RPB 90-93-P-A and RPB 90-94-P-A, October, 1991.

3.3 "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: SVEA-96 LOCA Sensitivity Studies," ABB Report CENPD-283-P-A (proprietary), CENPD-283-NP-A (non-proprietary), July 1996.

3.4 "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification," ABB Report CENPD-284-P-A (proprietary), CENPD-284-NP-A (non-proprietary), July 1996.

3.5 "SVEA-96 Critical Power Experimentation on a Full Scale 24 Rod Subbundle," ABB Report UR 89-210-P-A, June 1994.

4.1 MATPRO Revision 11-Version 2, "A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behaviour", NUREG/CR-0497, Tree-1280.

4.2 "Fuel Rod Design Methods for Boiling Water Reactors," ABB Report CENPD-285-P-A (proprietary), CENPD-285-NP-A (non-proprietary), July 1996.

4.3 W. J. O'Donnel and B. F. Langer, "Fatigue Design Basis for Zircaloy Components," Nuc. Sci. and Eng., Vol. 20, pg. 1-12 (1964).

4.4 "QUAD+ BWR Critical Power Correlation Development Report," Westinghouse Report WCAP-11369, September 1986.

4.5 "ABB Nuclear Design and Analysis Programs for Boiling Water Reactors: Program Description and Qualification," ABB Report BR 91-4C2-P-A, May, 1991.

4.6 Timoshenko and Gere, "Theory of Elastic Stability", McGraw-Hill, 1963

4.7 M. P. Paidoussis, "An Experimental Study of Vibration of Flexible Cylinders Induced by Nominally Axial Flow," Nucl. Sci. and Eng., Vol. 30, pg. 121, 1969.

4.8 "The Effect of Reduced Clearance and Rod Bow on Critical Power in Full-Scale Simulations of 8x8 BWR Fuel", ASME Publication 75-HT-69, 1975.

4.9 "The Effect of Reduced Clearance and Rod Bow on Critical Power in Simulated Nuclear Reactor Bundles," Paper No. 5, ANS Reactor Heat Transfer Meeting, Karlsruhe, October, 1973.

TABLE 2.1 AND TABLE 7.1

Proprietary Information Deleted

TABLE 7.1

SVEA-96/SVEA-100 FUEL DELIVERIES

Status January 1994

Proprietary Information Deleted

Grand Total: [#] Reloads and [#] Lead Fuel Assemblies

RL = SVEA-96/100 Reload

\# Proprietary Information Deleted

TABLE 7.1 (CONTINUED)

SVEA-96/100 FUEL DELIVERIES

Status November 1993

Proprietary Information Deleted

Grand Total: [#] assemblies delivered by December 1993

Proprietary Information Deleted

TABLE 7.2 THROUGH TABLE 7.6

Proprietary Information Deleted

FIGURE 2.1a THROUGH FIGURE 4.3.11-1

Proprietary Information Deleted

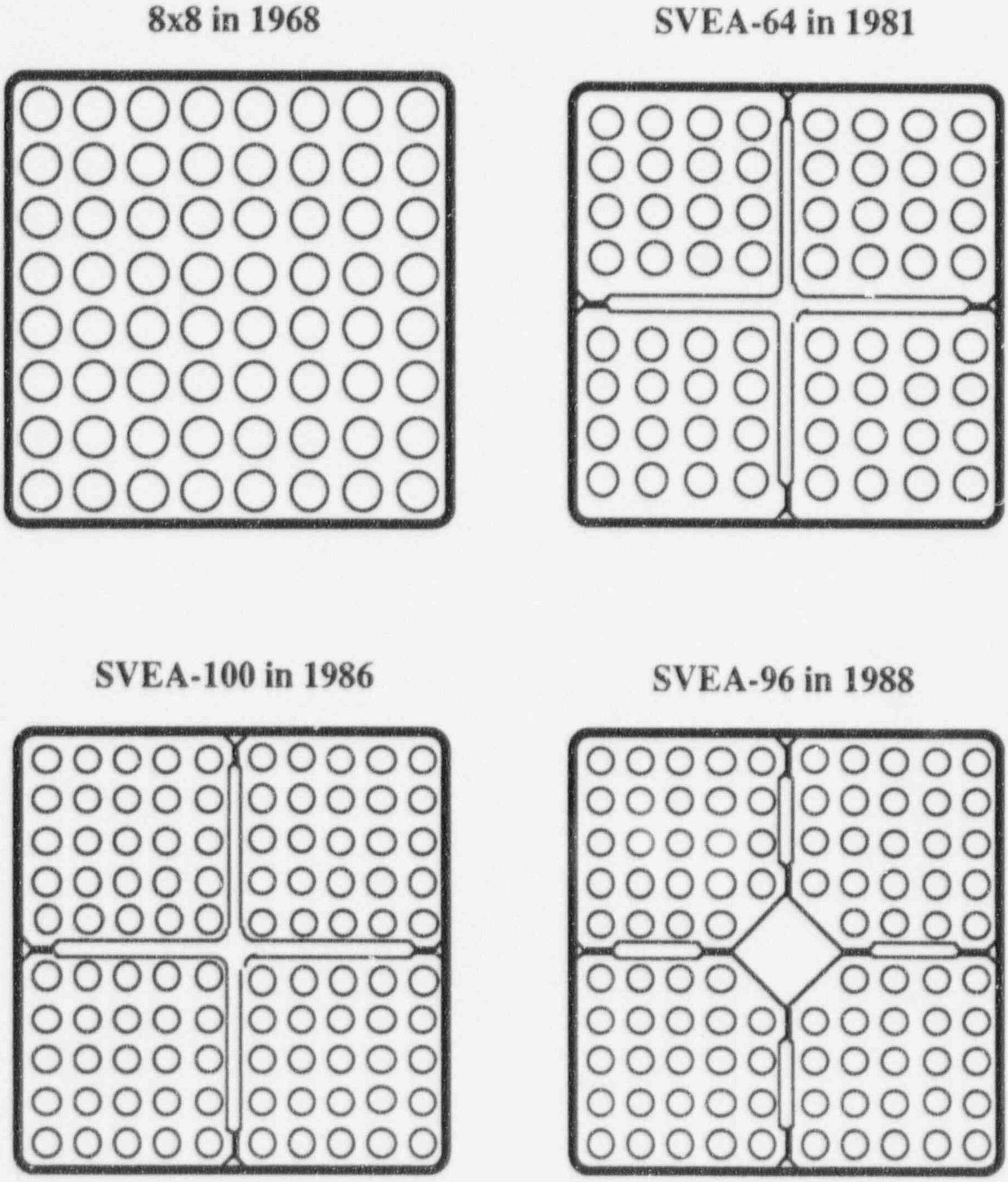

Figure 7.1 ABB Fuel Design Evolution

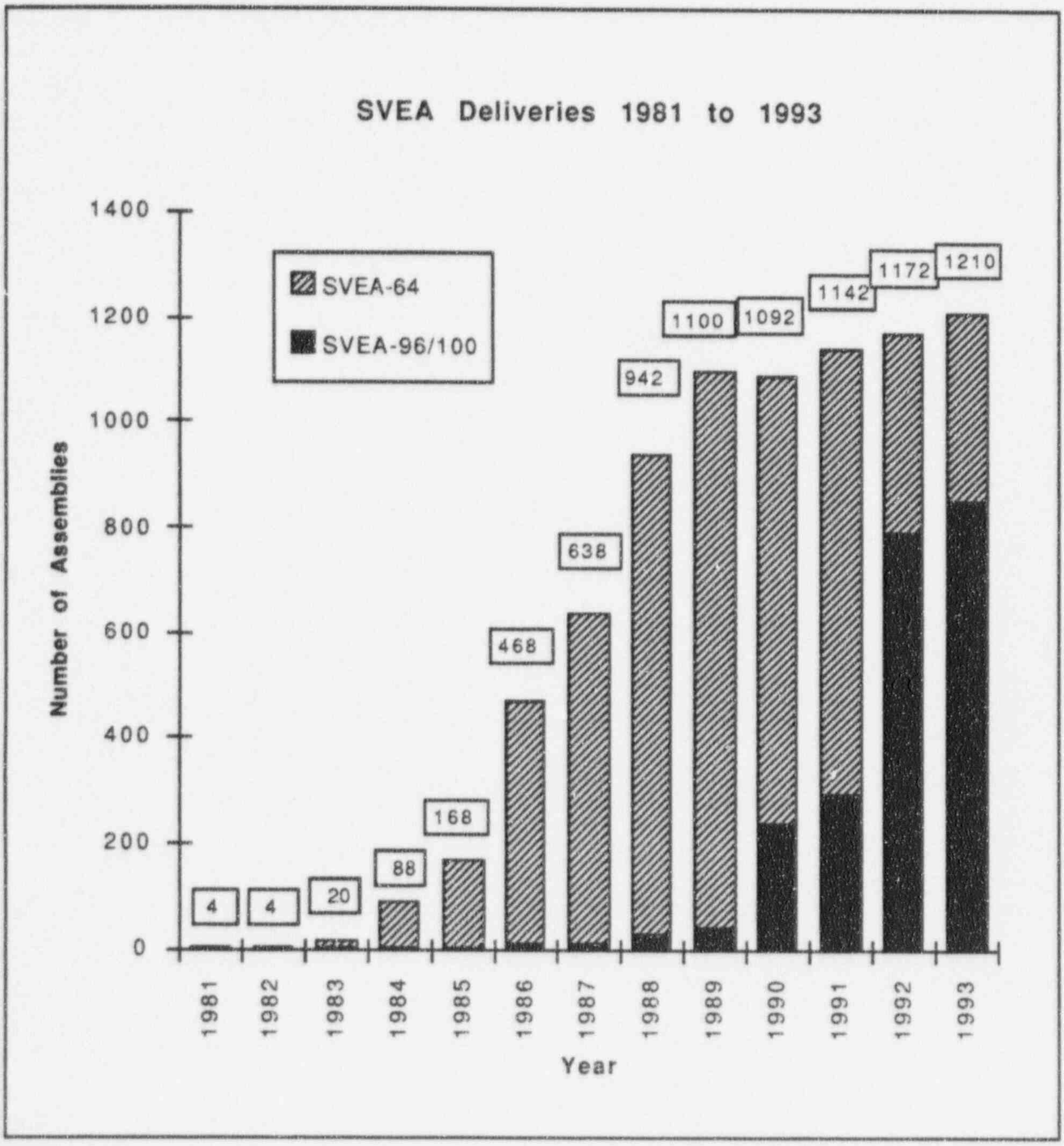

Figure 7.2 SVEA Fuel Deliveries - 1981 through 1993.

FIGURE 7.3

Proprietary Information Deleted

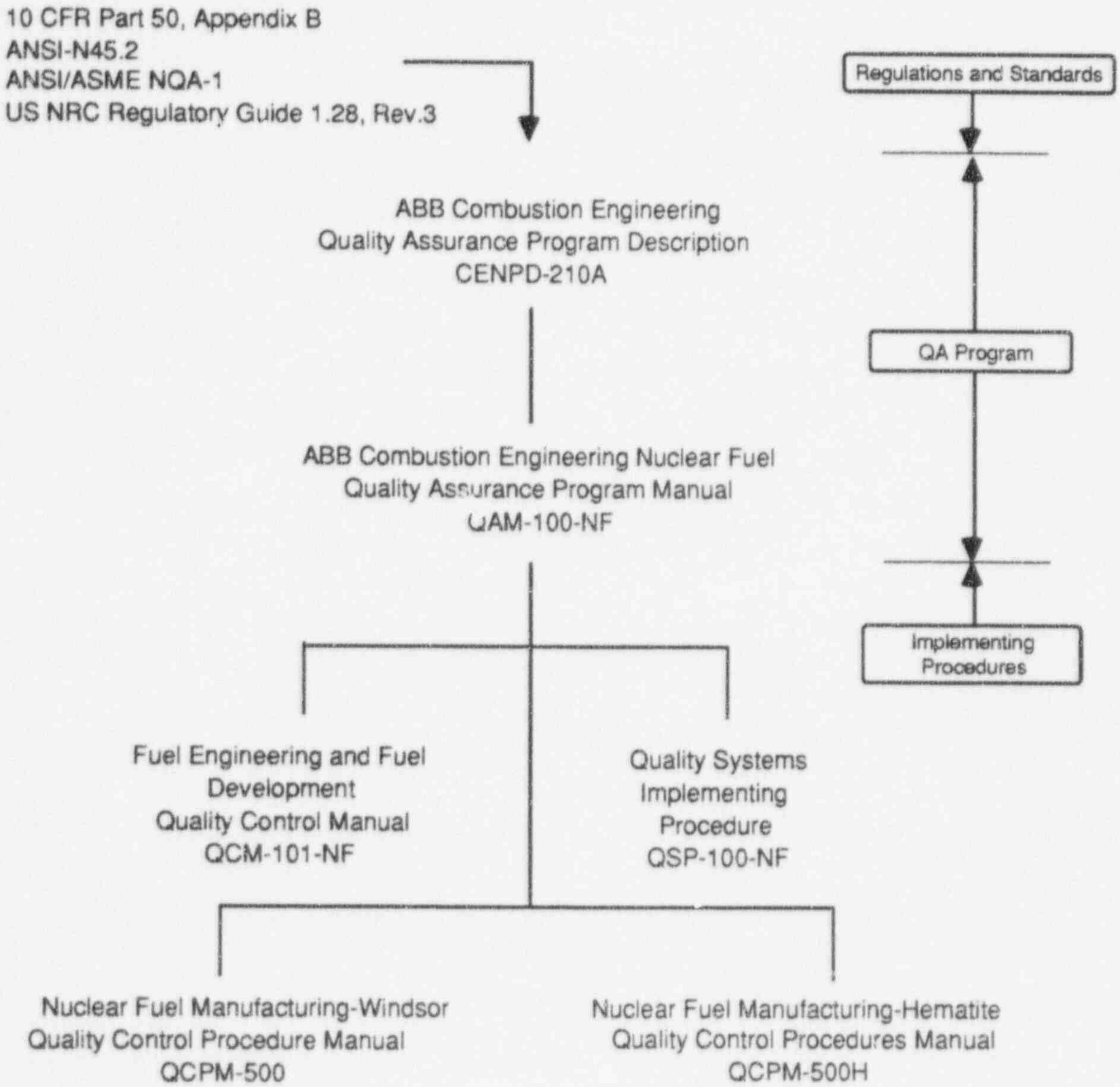

Figure 9.1 ABB CENO Quality Assurance Program

APPENDIX A: RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

A.1 Introduction

This appendix contains responses to the NRC Request for Additional Information regarding Reference A1.

Reference A1 contains the ABB methodology for the fuel assembly and fuel rod mechanical evaluation identified in Section 4.2 of the Standard Review Plan, NUREG-0800. It also contains an application of that methodology to the ABB SVEA-96 fuel assembly which demonstrates that the SVEA-96 assembly satisfies the ABB design criteria.

Typographical errors have been identified in the original submittal of Licensing Topical Report, CENPD-287-P (Reference A1). The errata are listed in Table A.1-1. The corrections identified in Table A.1-1 have been made to the main body of this report.

A.2 Questions and Responses

NRC Question A1 (General Question)

Several figures are given in this report that are plotted as a function of burnup, e.g., Figures 2.15 a&b, 4.2.1, 4.2.2, 4.2.3, & 4.2.8, that are not defined as to whether they are nodal, rod average or assembly average burnups. Please provide this information.

ABB Response to Question A1

The burnup referred to in Figures 2.15 a and b, 4.2.1, 4.2.2, 4.2.3, and 4.2.8 is assembly average burnup.

NRC Question A2 (General Question)

Please provide the STAV 6.2 application methodology for steady-state initialization of accidents and transients such as for LOCA. Also include how STAV 6.2 code and power uncertainties are applied in these applications.

ABB Response to Question A2

Fuel rod performance parameters are calculated and used as input to various transient and accident analyses. [Proprietary Information Deleted]

Fast Transient Analysis

[Proprietary Information Deleted]

Example of Core Average and Hot Channel h_{gap} for Fast Transients

[Proprietary Information Deleted]

CRDA Analysis

[Proprietary Information Deleted]

LOCA Analysis

[Proprietary Information Deleted]

Example of Conservatism in STAV6.2 Input to LOCA Analysis

[Proprietary Information Deleted]

Stability Analysis

[Proprietary Information Deleted]

Dose Calculations

[Proprietary Information Deleted]

Summary

The methodology for determining these parameters for each type of analysis is also discussed in the response to Question A1 on CENPD-285-P-A (Reference A3) and summarized in the table below:

Analysis Type	**Parameter Required**	**Where Methodology Described***
Transient	h_{gap}	Ref. A5, Sect 3.4.3
LOCA	h_{gap} & Gap Gas Composition, Volumes, and Dimensions	Ref. A5, Sect 3.4.5
CRDA	h_{gap}	Ref. A12, Response A19
Stability	h_{gap}	Ref. A5, Sect 3.4.7

* Modifications to the sections referenced will be implemented as required to make them consistent with the methodology described in this response.

NRC Question A3 (Regarding Section 4)

From examination of all subsections in Section 4.0 it appears that unirradiated values of yield and tensile strength are used to determine stress limits for assembly components and cladding; however, the beginning of Section 4.0 implies that irradiation effects are included. Please clarify this contradiction for stress limits on assembly components and fuel cladding.

ABB Response to Question A3

The stress evaluation must demonstrate that stress limits are satisfied at all burnups. Therefore, ABB utilizes yield and ultimate stress values for Zircaloy components which are burnup dependent as required as discussed in Section 4.0 within the constraint that stress limits will be satisfied with sufficient conservatism.

[Proprietary Information Deleted]

The general trends in yield and ultimate strength as a function of irradiation given in Reference A3 are currently utilized to conservatively demonstrate that stress limits are satisfied for the channel. When unirradiated values are not utilized for irradiated components, the effects of irradiation are treated conservatively. For example, when comparing to stress limits, minimum values of yield and tensile strengths are used. An example of this approach is provided in Section 4.2.5 of Reference A1. The table in section 4.2.5 provides BOL stresses in the channel at various locations and the corresponding BOL allowable values. It is noted that conservative estimates of the increase in outer channel and watercross peak stresses associated with wall thinning due to corrosion are [Proprietary Information Deleted], respectively. However, the yield and tensile strengths are expected to increase by factors of [Proprietary Information Deleted]

The influence of irradiation on stainless steel and nickel-based alloy strength parameters is not significant. Therefore, unirradiated strength values are used at all burnups.

<u>NRC Question A4 (Regarding Section 4.2.1)</u>

Channel bow in a D lattice is mentioned but no data are provided for this application. Are these D lattice applications expected? If so, please provide this data and how it will be applied. If such applications are not expected, but should D lattice applications arise in the future, the channel bow data will need to be submitted at that time.

<u>ABB Response to Question A4</u>

SVEA-96 reload applications in D-lattice plants are anticipated.

[Proprietary Information Deleted]

<u>NRC Question A5 (Regarding Section 4.2.1)</u>

Examination of Figure 4.2.3 shows that the variance in the channel growth data increases with increasing burnup and there is very little data above 40 GWd/MTU. What is the 95/95 upper bound of this growth data (rod average) as a function of burnup taking into account the increase in variability with burnup and the limited data above 40 GWd/MTU?

<u>ABB Response to Question A5</u>

[Proprietary Information Deleted]

NRC Question A6 (Regarding Section 4.2.2.2)

Please provide the differential rod growth data used to predict end plug shoulder-to-lower tie plate gap as a function of burnup. Also what are the fabricated tolerances (maximum and minimum) for the end plug shoulder-to-tie plate gap?

ABB Response to Question A6

ABB engages in ongoing programs of in-reactor inspections and laboratory testing to monitor the performance of its BWR fuel. The purpose of this program is to confirm that the fuel is behaving as expected and to identify and correct any potentially undesirable behavior. The most current available data are utilized to evaluate performance relative to design criteria. Evaluation of differential rod growth provides a good example of this approach.

[Proprietary Information Deleted]

Figure A6-3 shows the current SVEA-96/100 differential rod growth data base in terms of clearance between the end plug shoulder and the upper tie plate. [Proprietary Information Deleted]

NRC Question A7 (Regarding Section 4.2.2.3)

The maximum fast fluence level data for spacer spring relaxation of the SVEA-96/100 design (Figure 4.2.11) is approximately a factor of three less than the goal burnup requested in this submittal. However, it is stated that there is extensive experience with this same material and general spacer design on other SVEA fuel assembly designs. What are the maximum fast fluence and corresponding burnup levels of this other experience and are spacer spring relaxation data available? How does this data compare to the data in Figure 4.2.11? Also what are the similarities and differences between the SVEA-96/100 spacer springs and the other SVEA assembly designs?

ABB Response to Question A7

[Proprietary Information Deleted]

The SVEA-96/100 spacer design is based on earlier ABB 8x8 and SVEA-64 spacer designs and utilizes the [Proprietary Information Deleted] and the same fabrication technique as the earlier designs. The spacers are fabricated from [Proprietary Information Deleted]

The spacers for the ABB 8x8, SVEA-64, and SVEA-96/100 assemblies have the same general design. [Proprietary Information Deleted] In effect, the SVEA-96/100 spacer spring is a refined version of the SVEA-64 and 8x8 spacer spring design.

Minor differences in the spacer designs for the different fuel types are required due to differences in fuel rod pitch, number of rods, and diameter and weight of the rods. For example, the type of differences are illustrated for the SVEA-96/100 and SVEA-64 designs in Table A7-1. In addition, drawings of the SVEA-96 and the SVEA-64 spacer cells, showing the similarities and the differences in spacer spring design, are provided in Figures A7-1 and A7-2. As shown by these comparisons, the general spacer and spring designs are very similar, and [Proprietary Information Deleted]

Operating experience with this general spacer design has been extensive. This experience is reflected in Figure A7-3 which shows the number of assemblies as a function of assembly average burnup. Post irradiation examinations have not indicated fuel rod to spacer spring gaps or evidence of fuel rod fretting which might be expected if gaps had developed.

[Proprietary Information Deleted]

NRC Question A8 (Regarding Section 4.2.2.3)

It is also stated that the same material and basic design of the external compression springs for the SVEA-96/100 design have been used on all SVEA designs and, therefore, there is extensive experience. What is the maximum fast fluence levels of the compression spring experience for other SVEA designs and how does this compare to the maximum fluence levels that can be achieved at the goal burnups requested in this submittal?

ABB Response to Question A8

The same experience reflected in Figure A7-3 applies to the external compression springs. The same external compression spring design and spring material has been utilized in the 8x8 and SVEA-64 designs as well as in the SVEA-96/100 assemblies.

The maximum fast fluence (greater than 1 Mev) in the vicinity of the external compression spring is expected to be about [Proprietary Information Deleted] Figure A7-3 indicates that the anticipated application falls within the existing SVEA-96/100 experience base.

NRC Question A9 (Regarding Section 4.2.5)

The channel edge of the water cross appears to be over stressed and the flat part of the water cross is very close to the stress limits. Of *additional concern is the fact that these analyses are extrapolations of an earlier analysis at lower differential pressures rather than actual calculations at the maximum differential pressures. Please address these concerns.*

ABB Response to Question A9

The stresses in the channel during operation with a maximum internal overpressure of [Proprietary Information Deleted] are discussed in Section 4.2.5 of Reference A1. All stresses in the outer channel and the water cross compare favorably with the allowable values identified in the ASME Code, Sect III at 300° C. For example, the stresses at BOL calculated without the effect of wall thinning and the allowable values at BOL are shown in the first table in Section 4.2.5 of Reference A1. The stresses in the edge of the cross and flat part of the cross are less than [Proprietary Information Deleted] of the allowable stress, respectively, relative to the ASME code allowable stresses at 300° C. The allowable values based on the ASME code include the safety margins needed for each stress category. Therefore, if the stresses in the channel do not exceed the ASME limits, they will be well below the stresses which would be required to over stress the channel. [Proprietary Information Deleted]

As discussed on page 64 of Reference A1, and clarified in the response to Question A3 in this document, the impact of maximum wall thinning due to corrosion of the channel on margins to stress limits is more than offset by the increase in yield and ultimate strength as a function of irradiation. Therefore, the margins to stress limits as the channel is irradiated will be at least as great as those calculated at BOL.

[Proprietary Information Deleted]

NRC Question A10 (Regarding Section 4.3.0)

It is stated that the SVEA Design Power History (DPH) in Figure 4.3-1 has been derived based on the most limiting design bases in Section 3.3 and calculations with the computer codes in CENPD-285-P. Have code uncertainties been included in determining the DPH and is the DPH for peak node or rod average linear heat generation rates? The individual power histories (Group A, B and C) are based on peak burnup at end-of life (EOL). This will not necessarily pick the peak LHGRs that occur in a given cycle, and the latter may be more important to maximizing the EOL rod pressure. Please revisit the power history examinations, find rods with peak-in-cycle LHGRs, and calculate the FGR for these rods.

ABB Response to Question A10

The design power history (DPH) is provided to the plant operator in terms of a Linear Heat Generation Rate (LHGR) operating limit which should not be exceeded during normal steady-state operation. It is referred to as the Thermal-Mechanical Operating Limit (TMOL) in Reference A5. [Proprietary Information Deleted]

The methodology described above has been used for a 3486 MWth BWR/5. The bundles selected and their descriptions are provided as an illustration of the methodology in Table A10-1. The rods selected from these bundles and considered to be most limiting are listed in Table A10-2. These power histories are shown in Figures A10-2 through A10-5.

NRC Question A11 (Regarding Section 4.3.0)

It is also stated that each of the power histories is multiplied by a factor that accounts for uncertainties in the nuclear calculation of the power histories, possible anticipated operational occurrences (AOOs) and the fuel performance calculational uncertainties. However, this factor does not appear to be large enough to account for calculated temperature uncertainties in STAV 6.2 for transient initialization or for fuel melting. Please provide an estimate of STAV6.2 calculated thermal uncertainties.

ABB Response to Question A11

[Proprietary Information Deleted]

The impact of an AOO on fuel temperature is evaluated as discussed in the response to Question A13. The level of conservatism in the analyses of a postulated LOCA using the STAV6.2 input is discussed in the response to Question A2.

NRC Question A12 (Regarding Section 4.3.1)

Please provide values for the standard deviations used in the sample application and resulting delta pressures calculated for each of the uncertainties.

ABB Response to Question A12

The values used in the nominal calculation and the corresponding perturbations are provided in the table below:

[Proprietary Information Deleted]

The individual results of the sample application calculations using the nominal values and each of the perturbations are as follows:

[Proprietary Information Deleted]

NRC Question A13 (Regarding Section 4.3.1)

The power histories are increased by a value that is intended to cover uncertainties in determining steady state rod powers and transients during AOOs. This value is treated as an uncertainty in the rod pressure calculation; however, transients during AOOs are considered by the NRC as actual events and should not be considered as an uncertainty. Transients during AOOs need to be included explicitly in the steady-state power histories.

ABB Response to Question A13

[Proprietary Information Deleted]

NRC Question A14 (Regarding Section 4.3.1)

A value of the internal rod pressure required for cladding lift-off plus uncertainty is provided when rod pressures exceed system pressure, but no basis is given for how these values were determined. In addition, the uncertainty value appears to be very low. What fuel swelling and cladding creep models and calculational assumptions were used to determine the lift-off pressure? Also, what are the uncertainties in the fuel swelling and cladding creep models? Please provide the data used to develop the swelling and creep model uncertainties.

ABB Response to Question A14

[Proprietary Information Deleted]

NRC Question A15 (Regarding Section 4.3.1)

The application of rod pressures above system pressure raise other fuel performance concerns: 1) cladding tensile stresses that result in hydride reorientation in the radial direction resulting in cracking and 2) DNB propagation due to fuel rods ballooning during transients and accidents. Please address these issues also in relation to fuel rod overpressure.

ABB Response to Question A15, Concern 1

[Proprietary Information Deleted]

Radial hydride precipitation in SVEA-96 fuel rods due to tensile hoop stresses during cool down is very unlikely. The BWR conditions under which the SVEA-96 assemblies operate are such that significant radial hydride reorientation is not expected even for rods exceeding system pressure. Furthermore, the number of fuel rods for which the internal pressure could exceed system pressure is very small.

The potential for radial hydride reorientation in a SVEA-96 fuel rod with an internal pressure equal to the conservative lift-off pressure of 17.7 MPa can be addressed by considering the predicted cladding conditions at EOL for the initial and boundary conditions described in the Response to Question A14. The initial conditions and boundary conditions as well as the resulting calculated conditions of the cladding at the fuel rod EOL are shown in Table A15-1. The SVEA-96 fuel rod is assumed to have the minimum wall thickness, maximum internal gas pressure, and maximum linear heat generation rate. [Proprietary Information Deleted]

It should also be noted that the number of SVEA-96 fuel rods which could exceed system pressure is very low for any anticipated reactor application. For example, an evaluation of a Nordic core containing SVEA-64 (ABB 8x8 water cross fuel design) fuel assemblies demonstrated that less than 0.1 % of the SVEA-64 fuel rods would exceed system pressure at end-of-life(EOL). This evaluation assumed that every fuel rod operated at 6% power above its nominal value. The analysis utilized a Monte Carlo technique to systematically include the effects of uncertainties in parameters having a significant impact on EOL fuel rod pressure. Specifically, the impact of uncertainties in cladding inner diameter, pellet densification, pellet outer diameter, pellet density, and initial helium pressurization were included. The power histories led to a peak rod nodal burnup of 45 MWd/kgU.

SVEA-96 assemblies can be operated to somewhat higher burnups than SVEA-64 assemblies. For example, a peak pellet burnup of 65 MWd/kgU corresponds to a peak nodal burnup of about 56 MWd/kgU in the assembly. However, experience with actual reactor applications has demonstrated that the higher burnup allowed for SVEA-96 fuel rods is more than compensated by the reduction in fuel rod power caused by the increased number of fuel rods. Therefore, fission gas releases in SVEA-96 fuel are generally less than in SVEA-64 assembles at EOL. Consequently, it is expected that less than 0.1% of the SVEA-96 fuel rods would exceed system pressure at EOL.

ABB Response to Question A15, Concern 2

Boiling transition propagation due to fuel rod ballooning assumes that the fuel rod cladding expands to reduce the pitch between adjacent fuel rods sufficiently to cause degraded boiling transition performance. The effect is postulated to propagate by successive fuel rods overheating, ballooning, and causing boiling transition in adjacent fuel rods. As discussed in Section 4.3.10 of Reference A1, significant degradation of boiling transition performance due to a reduction of fuel rod pitch in a BWR is expected only if adjacent fuel rods actually contact each other. In this situation, the maximum decrease in MCPR has been shown to be less than 15%. Furthermore, BWR assemblies are equipped with fuel channels which are separated by interassembly gaps. The presence of the fuel channels would be expected to reduce the probability of boiling transition propagating from the fuel rods in one assembly to those in an adjacent channel to a negligible value.

[Proprietary Information Deleted]

One of the design concerns during a postulated Loss of Coolant Accident (LOCA) is to maintain fuel coolability. The United States Atomic Energy Commission, predecessor to the Nuclear Regulatory Commission, in their review of the acceptance criteria for ECCS (Reference A16), concluded that maintaining the peak cladding temperature below 2200°F and maintaining less than 17 percent local cladding oxidation will ensure that sufficient ductility of the cladding remains during the quenching process.

Therefore, the core fuel structure will remain intact and amenable to long-term cooling when the temperature and oxidation design criteria are met. Furthermore current best estimate analyses of postulated LOCAs demonstrate that substantially lower temperatures would be predicted.

Experimental tests have shown that even with a substantial percentage of flow blockage, sufficient coolant flow is maintained to ensure fuel cooling. For example, the amount of rod swelling and flow area reduction was specifically evaluated in the BWR FLECHT test ZR-2 reported in Reference A9. These tests allowed for fuel rod temperatures up to about 2200 °F for an extended length of time. Therefore, these tests were very conservative simulations of LOCAs which might actually occur in a BWR. The BWR FLECHT test ZR-2 demonstrated that even at the elevation of maximum flow area reduction, the region near the channel wall and outer rods remained free from significant flow area reduction leaving at least 50% of the available flow area. Therefore, it was concluded that the BWR ECCS core-spray will not be significantly impaired by the flow blockage even at the worst elevation. This test result also demonstrates under accident conditions the effect of the BWR channels to limit any CPR propagation. Consequently, BWR ECCS testing has confirmed that boiling transition propagation due to fuel rod ballooning would not remove the capability to maintain a coolable geometry.

NRC Question A16 (Regarding Section 4.3.1)

What are the standard deviations of the fuel swelling and cladding creep rates used in calculating cladding strains?

ABB Response to Question A16

The upper and lower bound values for the pellet swelling and cladding creep rates are provided in the table below. [Proprietary Information Deleted]

NRC Question A17 (Regarding Section 4.3.4)

Correlating cladding hydrogen pickup as a function of burnup alone is unusual because it is generally considered to be a function of both cladding oxidation and cladding material. Cladding oxidation is in turn dependent on material, heat rating, coolant temperature, coolant chemistry, and exposure time. Please demonstrate that the intended applications in terms of material type, linear heat ratings, coolant temperatures and chemistry, and exposure times of the data bound the intended applications in U.S. plants. Also, what is the hydrogen pickup fraction of this data in terms of cladding corrosion? Also, how are the uniform average hydrogen levels determined (measured) for the data presented in Figure 4.3.4-1?

ABB Response to Question A17

[Proprietary Information Deleted]

NRC Question A18 (Regarding Section 4.3.4)

Figure 4.3.5-1 provides two sets of cladding corrosion data. The first set is for average oxide thickness. The second is for maximum oxide thickness from eddy current measurements. How are the average oxide thicknesses determined, and are the maximum thicknesses measured maximum individual measurements?

ABB Response to Question A18

[Proprietary Information Deleted]

NRC Question A19 (Regarding Section 4.3.5)

A design limit on oxide layer thickness is provided. Is this limit for maximum or average oxide thickness? If it is intended for average oxide thickness, please justify because fuel rods will most likely fail at their weakest or most embrittled location.

ABB Response to Question A19

[Proprietary Information Deleted]

ABB engages in ongoing programs of in-reactor inspections and laboratory testing to monitor the performance of its BWR fuel. The purpose of this program is to confirm that the fuel is behaving as expected and to identify and correct any potentially undesirable behavior. The most current available data are utilized to evaluate performance relative to design criteria. [Proprietary Information Deleted]

NRC Question A20 (Regarding Section 4.3.6)

How are cladding temperatures determined for input to the COLLAPS-II creep collapse calculation? Please provide an example. It is not clear whether the uncertainties in the creep model are included in the COLLAPS-II calculation of creep collapse. If uncertainties are not included please justify and provide an example application when uncertainties are included in calculating cladding collapse.

ABB Response to Question A20

[Proprietary Information Deleted]

If the analysis is performed utilizing conservative input parameters as shown in the example in Section 4.3.6 of Reference A1, the parameters listed above are treated as follows:

[Proprietary Information Deleted]

These assumptions are considered to be very conservative. [Proprietary Information Deleted]

NRC Question A21 (Regarding Section 4.3.10)

Please provide data on measured fraction of rod-to-rod channel closure based on the nominal as-fabricated channel width as a function of fast fluence. Also, based on this same data, please estimate maximum channel closure at a 95% confidence level as a function of fast fluence at the axial location with maximum bow.

ABB Response to Question A21

Table A21-1 shows results of pool side examinations of SVEA-96/100 fuel rods for evidence of fuel rod bow. [Proprietary Information Deleted]

A.3 References

A1. Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, ABB Report CENPD-287-P (proprietary), CENPD-287-NP (non-proprietary), June, 1994.

A2. Letter Shih-Liang Wu (NRC) to D.B. Ebeling-Koning (ABB), "Request for Additional Information on CENPD-287-P, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors,"(TAC NO. M90189), May 9, 1995

A3. Fuel Rod Design Methods for Boiling Water Reactors, ABB Report CENPD-285-P-A (proprietary), CENPD-285-NP-A (non-proprietary), July 1996.

A4. MATPRO Revision 11-Version 2, "A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behaviour", NUREG/CR-0497, Tree-1280.

A5. Reference Safety Report for Boiling Water Reactor Reload Fuel, ABB Report CENPD-300-P-A (proprietary), CENPD-300-NP-A (non-proprietary), July 1996.

A6. "ROPE-I: The Studsvik BWR Rod Overpressure Experiment", Schrire, et.al., 1994 International Topical Meeting on Light Water Reactor Fuel Performance, West Palm Beach, Florida, April 17 through April 21, 1994, p. 212

A7. "Redistribution of Hydrogen in Zircaloy", K. Forsberg and A.R. Massih, Journal of Nuclear Materials, 172, 1990, page 130-134.

A8. "Corrosion and Hydriding Performance of Zircaloy-2 and Zircaloy-4 Cladding Materials in PWRs", Limbach et. al., Proceedings - 1994 International Topical Meeting on Light Water Reactor Fuel Performance, West Palm Beach, Florida, April 17-21, 1994

A9. J.D. Duncan and J.E. Leonard, "Emergency Cooling in Boiling Water Reactors under Simulated Loss-of-coolant Conditions (BWR-FLECHT Final Report)",General Electric Report GEAP-13197, June 1971.

A10. Kearns, J. J., Terminal Solubility and Partitioning of Hydrogen in the Alpha Phase of Zirconium, Zircaloy-2 and Zircaloy-4, J. Nucl Materials 22 (1967) 292-303.

A11. Northwood, D. O. and Kosasih, U., Hydrides and Delayed Hydrogen Cracking in Zirconium and its Alloys, International Metals Reviews, 1983, Vol. 28, No. 2 92-121.

A12. Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification, ABB Report CENPD-284-P-A (proprietary), CENPD-284-NP-A (non-proprietary), July 1996.

A13. D. Hardie and M.W. Shanahan, Journal of Nuclear Materials, 1974, 55, page 1-13

A14. J. Kearns and C.R. Woods, Journal of Nuclear Materials, 1966, 20, page 241-261

A15. Thermal-Hydraulic Stability Methods for Boiling Water Reactors, ABB Report CENPD-294-P-A (proprietary), CENPD-294-NP-A (non-proprietary), July 1996.

A16. United States Atomic Energy Commission, "Opinion of the Commission -- Acceptable Criteria for Emergency Core Cooling Systems -- Light-Water Cooled Nuclear Reactors," Docket No. RM-50-1, December 28, 1974.

A17. Supplement 1 to the Technical Report of Densification of General Electric Reactor Fuels, US AEC Regulatory Staff, December 14, 1973.

A18. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors"

A19. U.S. Nuclear Regulatory Commission Standard Review Plan, NUREG-0800.

A20 Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling Water and Pressurized Water Reactors"

A21. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Supplement 1 to Code Description and Qualification, ABB Report CENPD-293-P-A (proprietary), CENPD-293-NP-A (non-proprietary), July 1996.

TABLE A.1-1

ERRATA LIST FOR CENPD-287-P

Location	Correction Description
Page 76, Section 4.3.0	Last sentence of second paragraph should read: [Proprietary Information Deleted]
Page 78	Definition of Group C should read: [Proprietary Information Deleted]

TABLE A7-1 THROUGH TABLE A10-2

Proprietary Information Deleted

TABLE A13-1

POSTULATED BWR ANTICIPATED OPERATIONAL OCCURRENCES AND PLANT MANEUVERS AFFECTING ROD INTERNAL PRESSURE

Description	Frequency During Assembly Residence in the Core (6 years)
1. Startup from Cold Conditions to Full Power	45
2. Reduction to 75% Power and Return to Full Power	1500
3. Reduction to 50% Power and Return to Full Power	300
4. Control Rod Pattern Exchange	60
5. Fast and Slow Anticipated Operational Occurrences (e.g. Turbine Trip, Turbine Generator Trip, Loss of Feedwater Heating)	18

TABLE A15-1 THROUGH TABLE A21-1

Proprietary Information Deleted

FIGURE A2-1 THROUGH FIGURE A13-1

Proprietary Information Deleted

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