CENPD-287-NP

FUEL ASSEMBLY MECHANICAL DESIGN METHODOLOGY FOR BOILING WATER REACTORS

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]

[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]

[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]

<u>Cladding</u>

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.. [Proprietary Information Deleted]

[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]

[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]

[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. Correspondance 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]

<u>Design Stress Intensities</u>

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. [Proprietary 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]

   [Proprietary Information Deleted]

   <u>Creep Deformation</u>

   [Proprietary Information Deleted]

   <u>Channel Bow</u>

   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 Information 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.

<u>Channel Bulge</u>

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]

[Proprietary Information Deleted]

<u>Channel Bow</u>

[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]

1. [Proprietary Information Deleted]

2. [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 confirm 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]

[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]

[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]

[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]

[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.

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:

[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:

[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]

[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.

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:

[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]

[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]

[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]

[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:

> [Proprietary Information Deleted]

[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]

[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]

[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 BWR's. 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]

[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:

Cladding Internal and External Pressure

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

Pressure at End Plug

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

Ovality

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]

[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]

[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]

[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]

[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]

[Proprietary Information Deleted]

8.3 Lateral Load Cycling Test, Channel and Spacer Grid

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

[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]

[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]

[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 Vasteras with some components manufactured in Windsor. [Proprietary Information Deleted]

[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, Oct. 1994.

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, May 1994.

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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 91-402-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

SVEA FUEL WITH LINER CLADDING

PROPRIETARY INFORMATION DELETED

TABLE 3.1

TYPICAL FUEL ASSEMBLY MATERIAL PROPERTIES

PROPRIETARY INFORMATION DELETED

TABLE 7.1

SVEA-96/SVEA-100 FUEL DELIVERIES

Status January 1994

PROPRIETARY INFORMATION DELETED

Grand Total: [Proprietary Information Deleted] Reloads and [Proprietary Information Deleted] Lead Fuel Assemblies

RL = SVEA-96/100 Reload

Status November 1993

PROPRIETARY INFORMATION DELETED

Grand Total: [Proprietary Information Deleted] assemblies delivered by December 1993

TABLE 7.2

SVEA-64 BURNUP STATISTICS AS OF SEPTEMBER 15, 1993

PROPRIETARY INFORMATION DELETED

TABLE 7.3

SVEA-96/SVEA-100 BURNUP STATISTICS AS OF SEPTEMBER 15, 1993

PROPRIETARY INFORMATION DELETED

TABLE 7.4

FUEL FAILURE DATA FOR TIME PERIOD 1981 - 1993

Fuel Type: 8 x 8

Total Rods placed in service 1979 through November 1993: approx. [Proprietary Information Deleted]

Proprietary Information Deleted

Fuel Type: SVEA-64

Total Rods placed in service 1981 through November 1993: [Proprietary Information Deleted]

Proprietary Information Deleted

Fuel Type: SVEA-96/100

Total Rods placed in service 1986 through November 1993: [Proprietary Information Deleted]

Proprietary Information Deleted

Proprietary Information Deleted

Data as of January 1994

TABLE 7.5

SVEA-64 HIGH BURNUP ASSEMBLIES INSPECTIONS

PROPRIETARY INFORMATION DELETED

TABLE 7.6

SVEA-96/100 FUEL INSPECTIONS

PROPRIETARY INFORMATION DELETED

[Proprietary Information Deleted]

Figure 2.1a Style 1 Fuel Assembly

[Proprietary Information Deleted]

Figure 2.1b Style 2 SVEA-96 Fuel Assembly

[Proprietary Information Deleted]

Figure 2.2 SVEA-96 Fuel Assembly Cross Section

[Proprietary Information Deleted]

Figure 2.3a SVEA-96 Assembly and Control Rod Orientation in a C-lattice Plant, Style 1

[Proprietary Information Deleted]

Figure 2.3b SVEA-96 Configuration in D-lattice Plant [Proprietary Information Deleted]

[Proprietary Information Deleted]

Figure 2.4 Fuel Assembly Lattice

[Proprietary Information Deleted]

Figure 2.5a Style 1 Fuel Subbundle

[Proprietary Information Deleted]

Figure 2.5b Style 2 Fuel Subbundle

[Proprietary Information Deleted]

Figure 2.6a Style 1 SVEA-96 Channel

[Proprietary Information Deleted]

Figure 2.6b Style 2 SVEA-96 Channel

[Proprietary Information Deleted]

Figure 2.7 SVEA-96 Handle and Leaf Spring Configuration

[Proprietary Information Deleted]

Figure 2.8a Standard Fuel Rod, Style 1

[Proprietary Information Deleted]

Figure 2.8b Standard Fuel Rod, Style 2

[Proprietary Information Deleted]

Figure 2.9a Tie Rod, Style 1

[Proprietary Information Deleted]

Figure 2.9b Tie Rod, Style 2

[Proprietary Information Deleted]

Figure 2.10a Spacer Capture Rod, Style 1

[Proprietary Information Deleted]

Figure 2.10b Spacer Capture Rod, Style 2

[Proprietary Information Deleted]

Figure 2.11 Internal Compression Plenum Spring

[Proprietary Information Deleted]

Figure 2.12 Enriched UO_2 and UO_2-Gd_2O_3 Pellet Dimensions

[Proprietary Information Deleted]

Figure 2.13 SVEA-96 Spacer

[Proprietary Information Deleted]

Figure 2.14 SVEA-96 Debris Filter

[Proprietary Information Deleted]

Figure 2.15a Channel Corrosion Results - Maximum Oxide Thickness

[Proprietary Information Deleted]

Figure 2.15b Channel Corrosion Results - Average Oxide Thickness

[Proprietary Information Deleted]

Figure 4.2.1 SVEA-64 Channel Creep Deformation

[Proprietary Information Deleted]

Figure 4.2.2 SVEA Channel Bow in Nordic Reactors

[Proprietary Information Deleted]

Figure 4.2.3 Channel Growth Data from European Reactors

[Proprietary Information Deleted]

Figure 4.2.4 Compatibility of SVEA-96 Assembly (BOL) and non-SVEA Assembly (BOL)

[Proprietary Information Deleted]

Figure 4.2.5 Compatibility of SVEA-96 Assembly (BOL) and non-SVEA Assembly (EOL)

[Proprietary Information Deleted]

Figure 4.2.6 Compatibility of SVEA-96 Assembly (EOL) and non-SVEA Assembly (BOL)

[Proprietary Information Deleted]

Figure 4.2.7 Compatibility of SVEA-96 Assembly (EOL) and non-SVEA Assembly (EOL)

[Proprietary Information Deleted]

Figure 4.2.8 SVEA-96/100 Fuel Rod Growth

[Proprietary Information Deleted]

Figure 4.2.9 SVEA-96 Clearance Between Subchannel and Handle

[Proprietary Information Deleted]

Figure 4.2.10 SVEA-96 Differential Fuel Rod Growth Limiting Situations

[Proprietary Information Deleted]

Figure 4.2.11 SVEA-96 Spacer Spring Relaxation

\B\BC\[Proprietary Information Deleted)

Figure 4.2.12 FEM Model For SVEA-96 Channel Stress Calcualtions

[Proprietary Information Deleted]

Figure 4.2.13 Calculated SVEA-96 Channel Deflections

[Proprietary Information Deleted]

Figure 4.3.0-1 Typical SVEA-96 Design Power History

[Proprietary Information Deleted]

Figure 4.3.0-2 SVEA-96 Power History 2C-46

[Proprietary Information Deleted]

Figure 4.3.0-3 SVEA-96 Power History 2A-36

[Proprietary Information Deleted]

Figure 4.3.0-4 SVEA-96 Power History 1B-38

[Proprietary Information Deleted]

Figure 4.3.3-1 Maximum SVEA-96 Cladding Strain for the DPH times 1.12

[Proprietary Information Deleted]

Figure 4.3.3-2 Maximum SVEA-96 Cladding Strain for SVEA-96 Power History 2A-36 times 1.12

[Proprietary Information Deleted]

Figure 4.3.3-3 Maximum SVEA-96 Cladding Strain for SVEA-96 Power History 2C-46 times 1.12

[Proprietary Information Deleted]

Figure 4.3.3-4 Maximum SVEA-96 Cladding Strain for SVEA-96 Power History 1B-38 times 1.12

[Proprietary Information Deleted]

Figure 4.3.4-1 Typical ABB BWR Cladding Hydriding

[Proprietary Information Deleted]

Figure 4.3.5-1 Typical ABB BWR Cladding Oxide Measurements

[Proprietary Information Deleted]

Figure 4.3.6-1 Calculated Worst-case Ovality as a Function of Time

[Proprietary Information Deleted]

Figure 4.3.9-1 Calculated Maximum BOL UO_2 Fuel Pellet Temperatures versus Fuel Rod LHGR

[Proprietary Information Deleted]

Figure 4.3.9-2 Calculated Maximum BOL UO_2-Gd_2O_3 Fuel Pellet Temperatures versus Fuel Rod LHGR

[Proprietary Information Deleted]

Figure 4.3.11-1 SVEA-96 PCI Threshold for Unlined Fuel

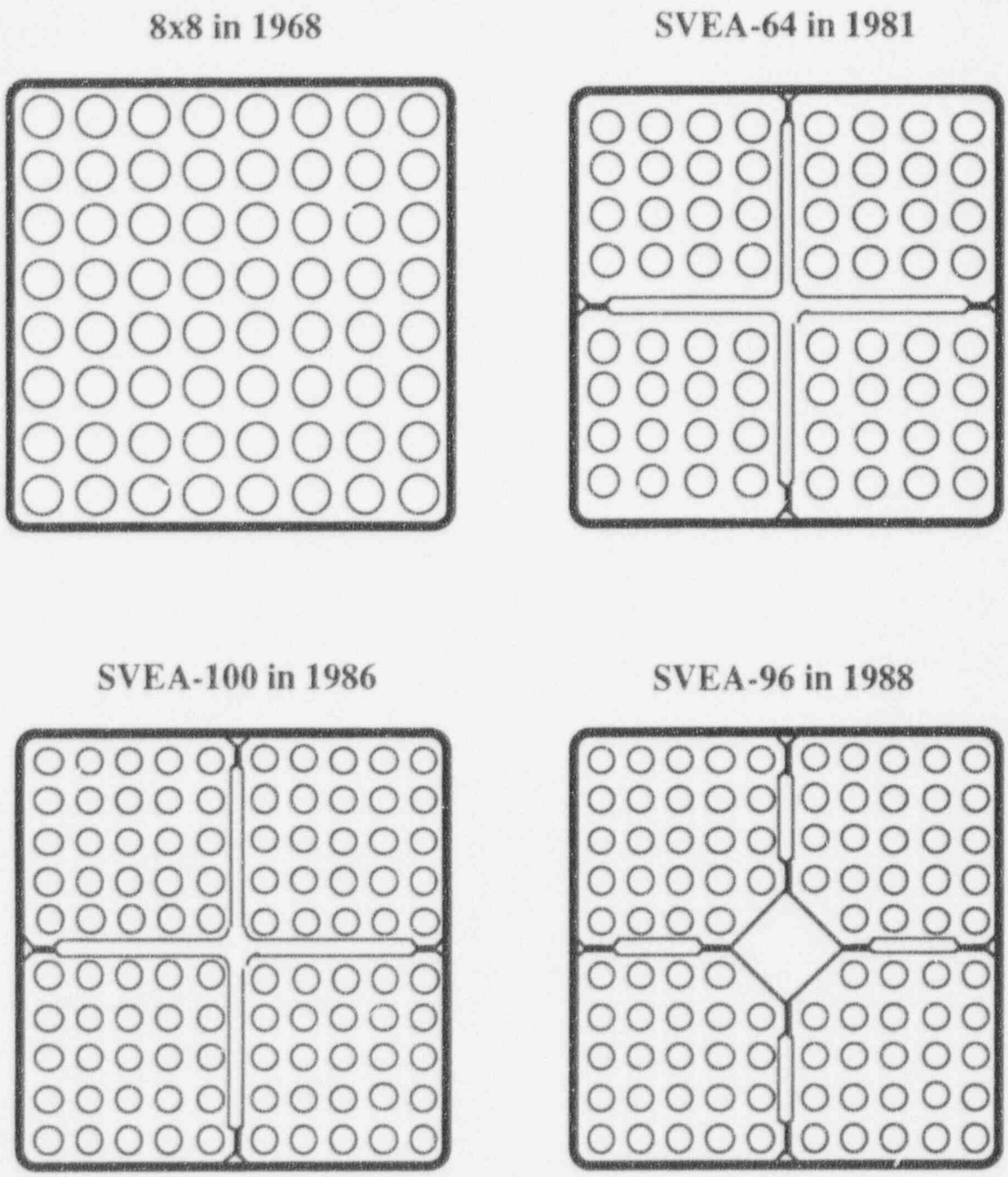

Figure 7.1 ABB Fuel Design Evolution

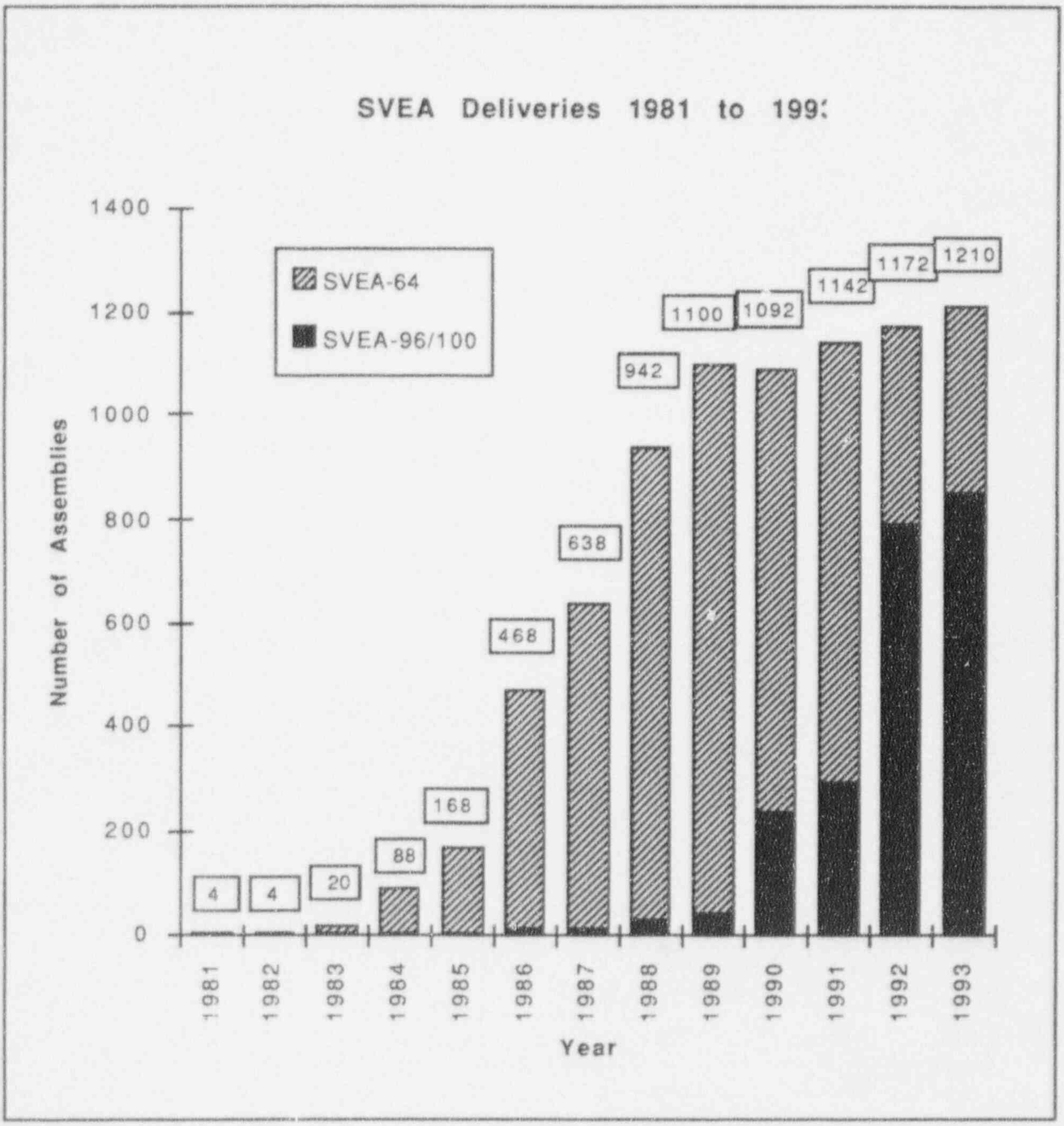

Figure 7.2 SVEA Fuel Deliveries - 1981 through 1993.

[Proprietary Information Deleted]

Figure 7.3 BWR Fuel Rod Failures in Time Frame 1981 through 1993.

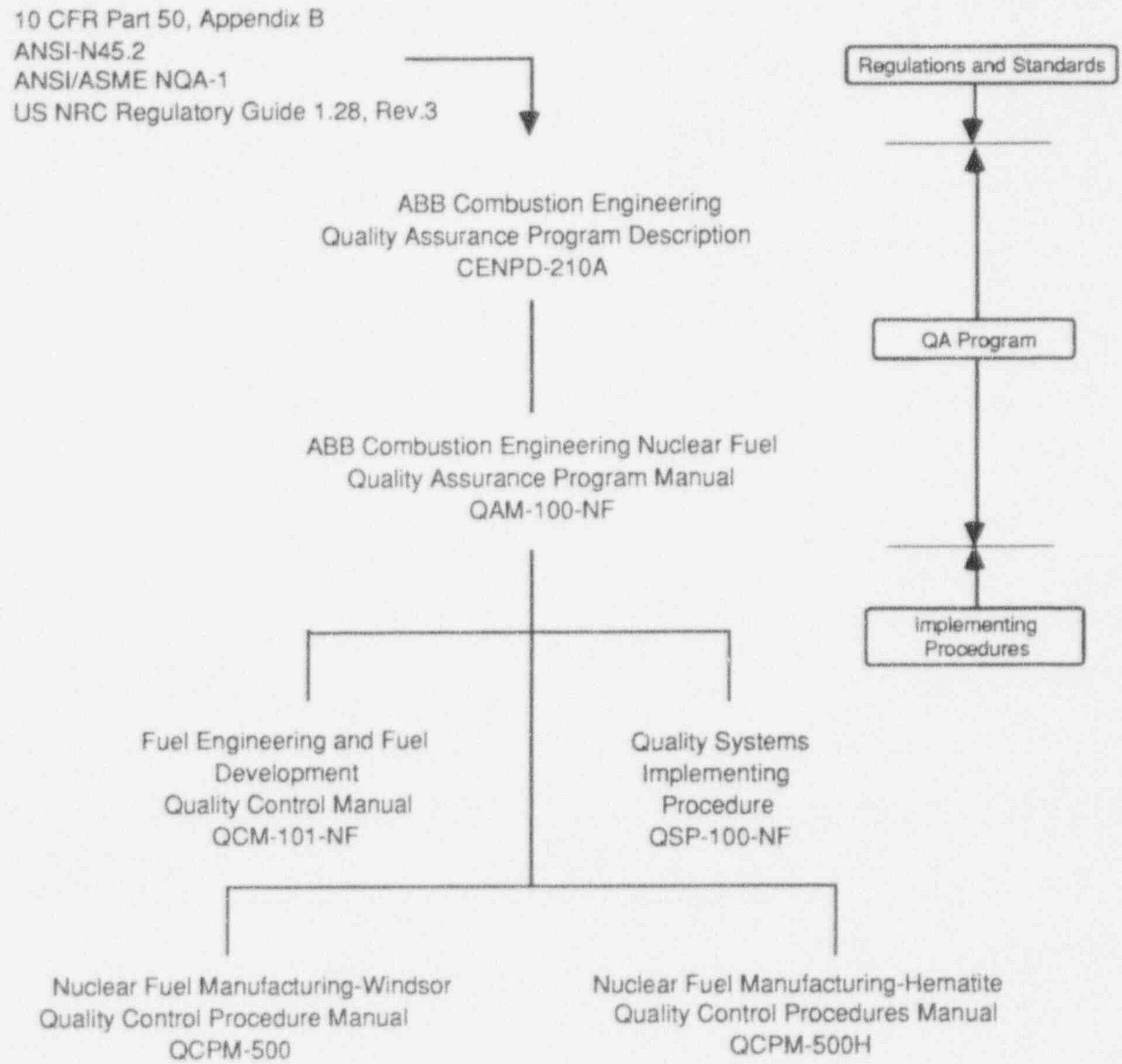

Figure 9.1 ABB CENO Quality Assurance Program

7

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