Westinghouse Non-Proprietary Class 3

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Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors Supplement 1 to CENP-287

Westinghouse

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1 SUMMARY AND CONCLUSIONS

1.1 SUMMARY

This Licensing Topical Report is a Supplement 1 to Reference 1.0, which contains the Westinghouse methodology for the fuel assembly and fuel rod mechanical evaluation identified in Section 4.2 of the Standard Review Plan, NUREG-0800 (Reference 1.4). This supplement describes improvements to the methodology contained in Reference 1.0 associated with adoption of the latest versions of the STAV, VIK and COLLAPS codes (STAV7.2, VIK-3, and COLLAPS-3.3D). Description and qualification of the STAV7.2, VIK-3, and COLLAPS-3.3D codes were submitted for NRC review in Reference 1.2. Reference 1.2 provides a description of the revised models implemented in these latest code versions along with the qualification actions which demonstrate that these codes are qualified for fuel rod design and safety analyses to a rod average burnup of 62 MWd/kgU.

Utilization of these improved code versions, i.e., STAV7.2, VIK-3, and COLLAPS-3.3D, in conjunction with the required revisions to the methodology associated with the use of these codes, as described in Reference 1.0, justifies an extension of the burnup range for which the methodology can be applied to a rod-average burnup of 62 MWd/kgU.

The methodology described in Reference 1.0 continues to be acceptable to a rod-average burnup of 50 MWd/kgU when used in conjunction with the STAV6.2, VIK-2 and COLLAPS-II codes as described in Reference 1.0.

This report also contains an application of the updated methodology to the latest version of the Westinghouse SVEA-96 fuel assembly referred to as SVEA-96 Optima2. In conjunction with an expanded fuel rod and assembly inspection data base and test basis, this sample application demonstrates that the SVEA-96 Optima2 assembly satisfies the Westinghouse design criteria to a rod-average burnup of 62 MWd/kgU for the sample plant application. As discussed in Reference 1.0, satisfaction of the Westinghouse 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 1.5). Similar information supporting the thermal-hydraulic, nuclear, and safety analyses evaluations are provided in Reference 1.1.

The SVEA-96 Optima2 fuel assembly contains part length rods in addition to full-length rods. As discussed in Section 4, the maximum rod-average burnup of 62 MWd/MtU to which the methodology and codes are considered to be qualified in this report refers to the rod-average burnup of full-length fuel rods. Further discussion of this position is provided in Section 2.

The burnup limit for licensing analysis is expressed in terms of rod-average burnup. In this report, performance data are sometimes provided in terms of assembly-average burnup. The precise relationship between rod-average and assembly-average burnup depends on the assembly and cycle nuclear design. However, for comparative purposes, a rod-average burnup of 62 MWd/MtU generally corresponds to an assembly-average burnup of about 55 MWd/kgU.

Design criteria and methods in Reference 1.0, which have not been changed, will continue to be used in the updated methodology. Therefore, this supplement relies on Reference 1.0 for design criteria and

methodology, which have remained unchanged. In general, descriptions of design criteria and methodologies described in Reference 1.0 which have not been revised have not been repeated in this document in order to focus on the differences relative to Reference 1.0. In some cases, however, design criteria and methodologies which have not changed relative to Reference 1.0 are clarified by providing updated descriptions. In Sections 2 through 10 of this report, in order to clearly discriminate descriptive text regarding such subjects as design criteria, methodology, and sample analyses from editorial comments relating this document to Reference 1.0, descriptive text is provided in normal font while editorial comments are given in italics.

The numbering of sections in this document follows that of Reference 1.0 in order to assist the reader in relating this supplement to Reference 1.0. However, equation, table, figure, and reference numbering in this supplement is independent of the numbering in Reference 1.0.

The contents of this report can be summarized as follows:

1. Description of the Westinghouse SVEA-96 Optima2 BWR watercross fuel assembly design as well as advanced design features being considered for this design,

2. Minor clarifications of the Westinghouse fuel assembly and fuel rod mechanical design criteria described in Reference 1.0,

3. Modifications, relative to the methodology described in Reference 1.0 of the Westinghouse design methodology for evaluation of performance relative to those criteria for normal operations and Anticipated Operational Occurrences (AOOs) associated with the improved code versions,

4. Sample application of the Westinghouse design evaluation methodology demonstrating compliance of the SVEA-96 Optima2 assembly with the design criteria for normal operations and AOOs to a fuel rod burnup of 62 MWd/kgU,

5. An updated summary of the computer codes used in Westinghouse methodology relative to the methodology described in Reference 1.0,

6. An updated description of the manufacturing inspection measures which assure that the assembly is constructed as required by the design specifications relative to the methodology described in Reference 1.0,

7. A summary of the operating experience with the SVEA-96 Optima2 design and similar Westinghouse designs,

8. An updated summary of the ex-core prototype test programs relative to the methodology described in Reference 1.0,

9. An updated summary discussion of ongoing testing, inspection, and surveillance plans relative to the methodology described in Reference 1.0.

Therefore, in conjunction with Reference 1.0, general design criteria as well as the design criteria for the fuel rods and other assembly components are clearly stated. The mechanical design methods used to evaluate assembly and component performance against these design criteria for normal operations and AOOs are then systematically addressed. An illustrative evaluation of the SVEA-96 Optima2 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.

1.2 CONCLUSIONS

The information contained in this report in conjunction with References 1.0 and 1.2 support the following conclusions:

1. The design bases identified are sufficient to assure that the requirements and guidelines identified in Section 4.2 of NUREG-0800 10CFR50, Appendix A and Section III of the ASME Code (Reference 1.3) will be satisfied.

2. The methodology for evaluating fuel assembly and fuel rod mechanical behavior relative to the design bases is acceptable for design and licensing applications to a rod-average burnup of 62 MWd/kgU, and

3. The methodology sample application to the SVEA-96 Optima2 BWR fuel assembly provides an illustration of the methodology to be utilized for each application to BWR fuel assemblies. These evaluations also demonstrate the capability of this assembly to satisfy the fuel performance, mechanical, and materials design bases under normal operation and anticipated operational occurrences to a peak rod-average burnup of 62 MWd/kgU.

NRC review and acceptance of this document for referencing in licensing applications to a rod-average burnup of 62,000 MWd/MtU is requested.

2 GENERAL DESCRIPTION

Section 2 is replaced to include a description of the SVEA-96 Optima2 assembly.

2.1 ASSEMBLY DESCRIPTION

The primary objective of the SVEA design is integrity and reliability of the fuel rod and assembly. To this end, improvements and design features are incorporated with the goal of achieving zero fuel rod failures during reactor operation. The fuel rod mechanical design methodology described in Reference 1.0 was illustrated by an application to the SVEA-96/96+ design. The revised methodology described in this document is illustrated by application to an improved version of the SVEA-96 design referred to as SVEA-96 Optima2. The SVEA-96 Optima2 design represents an evolutionary improvement of the basic SVEA-96 10x10 design in which an optimized lattice based on strategically positioned part-length rods is introduced. This section contains a description of the SVEA-96 Optima2 design. While similarities and differences relative to the SVEA-96 design described in Reference 1.0 are provided, sufficient detail is provided to allow a reader unfamiliar with the original SVEA-96 design to understand the SVEA-96 Optima2 design.

SVEA-96 Optima2 contains evolutionary design features which support high energy cycles with increased power densities and longer duration while maintaining the basic proven SVEA-96 design. Part-length fuel rods have been introduced which provide substantially improved shutdown margin performance, reduced pressure drop, improved reactivity, and improved stability performance. In addition, the dryout performance of the assembly has been improved by increasing the active flow area, improving the spacer design through the addition of mixing vanes, increasing the number of spacers, and optimizing the fuel rod pitch. Stability performance is improved by the reduction of the ratio of two-phase to one-phase pressure drop associated the introduction of the part-length rods. The SVEA-96 Optima2 design is based on well-proven concepts and materials which characterize the original SVEA-96 design. Furthermore, confidence in the robustness of the 10x10 SVEA design family (i.e., SVEA-100/96/96+/Optima/Optima2) and its high degree of reliability is based on extensive reload experience.

The SVEA-96 Optima2 design combines the features of the 10x10 SVEA design with the benefits of strategically positioned part-length rods. While the features of the design will be discussed more fully in the following detailed mechanical design description, it is instructive to summarize some of the major 10x10 SVEA mechanical characteristics which have contributed to the demonstrated reliable operation of the 10x10 SVEA assemblies over the past years.

- As with the original SVEA-96 design, the larger number of fuel rods in an assembly compared to the 8x8 design allows relatively high bundle powers while maintaining modest rod powers. The low linear heat generation rate (LHGR) and increased heat transfer area allows 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.

- SVEA-96 Optima2 utilizes the SVEA-96 channel. The integral construction provided by welding the watercross to the midspan of the outer channel results in substantially enhanced channel dimensional stability. Westinghouse BWR reactor experience has shown that the SVEA channels are less susceptible to channel bulge and bow than channels without this feature. 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.

- As explained in Reference 1.0, 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 channel rather than the fuel rods carry the tensile load associated with the assembly weight during fuel handling.

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

- The 10x10 SVEA cladding is recrystallized low corrosion Zircaloy-2. The cladding used for the 10x10 SVEA design over the past years has been optimized to provide optimum corrosion resistance. The result of this optimization process is a product referred to as "LK3" cladding. This fully annealed recrystallized Zircaloy-2 cladding has been demonstrated to exhibit excellent resistance to uniform corrosion while providing good nodular corrosion resistance as well as resistance to Crud Induced Localized Corrosion (CILC). There is also 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 10x10 design is a major contributing factor to its observed high level of corrosion resistance.

The SVEA-96 Optima2 fuel design with its 5x5-1 sub-bundle lattice and three strategically positioned optimized part-length rods in each sub-bundle represents a new generation of the SVEA-96 fuel.

2.1.1 Background

The SVEA-96 Optima2 fuel assembly consists of three basic components:

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

The SVEA-96 Optima2 description in this section and Section 5 provides typical numerical data for the SVEA-96 Optima2 design for a [ ]^{a,c} active fuel length assembly for a C-lattice plant. The SVEA-96 Optima2 fuel assembly design is shown in Figure 2-1 and the cross section dimensions are shown in Figure 2-2. Figure 2-1 is based on a [ ]^{a,c} active fuel length.

The fuel bundle consists of 96 fuel rods, arranged in four 5x5-1 sub-bundles. The sub-bundles are separated by a cruciform internal structure (water cross) in the channel. The water cross has a square central canal and smaller water channels in each of the four wings for non-boiling water during operation.

As with SVEA 96, the sub-bundles are inserted into the channel from the top and are [

]a,c. This design principle has always been used in Westinghouse BWR fuel and eliminates any leakage flow ambiguities at the bottom end of the channel. This feature also avoids stresses in the tie-rods during normal fuel handling since the weight of the fuel bears on the channel and bottom support. The bottom support with the integrated debris filter is designed to prevent potentially damaging debris from entering the fuel bundle. The fuel assembly is lifted by a handle, connected to the top end of the channel, and supported against adjacent assemblies in the core module by the double leaf spring.

As with the original SVEA-96 design, the sub-bundles are freestanding inside the channel. There is sufficient space for sub-bundle growth at the top of the assembly to avoid restriction due to differential growth between the fuel bundles and the channel. Furthermore, the bottom of the transition piece, or "nose piece," seats in the fuel support piece. The top end of fuel assembly is 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.

The original SVEA-96 design was equipped with [

]a,c

As shown in Figure 2-2, [

]a,c These fuel rod positions have been chosen to maximize the shut down margin improvement with a minimum number of part-length rods.

All rods in the SVEA-96 Optima2 design have the [

]a,c

These dimensions were optimized to achieve optimum uranium content while preserving acceptable fuel rod thermal-mechanical performance.

The SVEA-96 Optima2 spacers [

]a,c

[

]^{a,c}

Relative to the SVEA-96 design, the SVEA-96 Optima2 top and bottom tie plates have been adapted to accommodate the part-length rods and lateral variation in fuel rod pitch.

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 Optima2 fuel assemblies in a C-lattice plant are shown in Figures 2-3a and 2-3b. These gap widths provide adequate clearances to the control blades and rollers. The SVEA-96 Optima2 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. As with the SVEA-96/SVEA-96+ designs discussed in Reference 1.0, the SVEA-96 Optima2 transition piece (bottom nozzle) can be modified to offset the assembly toward the control rod gap in a D-lattice configuration as discussed in Section 2.4.

Reference is 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 10x10 SVEA design which has been optimized for use in Nordic BWRs built by Westinghouse. The SVEA-96 design has been optimized for reactors designed by General Electric and Siemens. The Nordic BWRs have an assembly pitch slightly greater than 152.4 mm. These reactors are operating in Sweden and Finland. Due to the slightly "wetter lattice" and 15 percent stronger control rods, the optimum 10x10 SVEA design for these reactors contains four 25-rod sub-bundles. The watercross in the SVEA-100 design is a cruciform water channel, without the large central section, which is welded to the outer channel in the same manner as the SVEA-96 design. The fuel rods are identical for the two designs. The spacers are identical with the exception of the internal corner of the sub-channel which supports an extra rod in the SVEA-100 design. The fuel rod pitch for SVEA-100 and SVEA-96 is [

]^{a,c} 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.

The SVEA-96S design is a SVEA-96+ assembly in a Nordic BWR. Therefore, the SVEA-96S channel [

]^{a,c}

For SVEA-96 Optima2, [

]^{a,c}

Reference is also made in this document to the SVEA-64 design. The SVEA-64 design utilizes a SVEA-100 channel and a 4x4 fuel rod array. Therefore, the fuel channel mechanical design and the

materials used in SVEA-64 are virtually identical to those utilized in the SVEA-96/100 designs. The SVEA-64 fuel rods have a larger diameter than the SVEA-96/100 fuel rods. While the same materials and conceptual design are utilized for the SVEA-64 spacer as those in the SVEA-96/100 design, the mechanical configuration of the spacers differ.

SVEA-64 assemblies with a [ ]^{a,c} developed for the U.S. market were marketed in the U.S. under the Westinghouse trade name "QUAD+."

The SVEA-96 Optima design is a forerunner to the SVEA-96-Optima2 design. Relative to the SVEA-96 design, it has eight part-length rods adjacent to the center water channel. In order to preserve Uranium mass, the [ ]^{a,c}.

2.1.2 Handle with Spring

The handle and leaf spring design for the SVEA-96 Optima2 assembly is the same as the handle and leaf spring for the SVEA-96/SVEA-96+ design discussed in Reference 1.0. As shown in Figure 2-7, the handle and leaf spring configuration is fitted to the top end of the channel. [ ]^{a,c}

The handle is designed for lifting with the standard handling equipment at the plant. An individual identification number for the fuel assembly is engraved in the handle.

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.

[ ]^{a,c}

2.1.3 Fuel Transport

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

Westinghouse qualified a transport and handling method for SVEA fuel for shipment of complete fuel assemblies (bundles in channel with handle) for European customers and has been shipping SVEA fuel channeled since the early 1990s in Europe. Transport of complete fuel assemblies is preferred for several

reasons. The sub-bundles are installed in the channel, and the handle is installed on the channel at the fabrication facility using this shipment method. This approach simplifies the process of assuring that the sub-bundles are properly installed in the channel since all work is performed at the fuel fabrication facility. Westinghouse has accumulated extensive experience transporting complete BWR fuel assemblies without plastic inserts between the rods.

[

]^{a,c}

Westinghouse has submitted a License amendment request for shipping fuel channeled to its U.S. customers.

2.1.4 Lattice and Fuel Rod Types

Each SVEA-96 Optima2 sub-bundle is a 5x5-1 lattice. The fuel assembly has [ ]^{a,c} which are mechanically different:

[

]^{a,c}

2.2 FUEL SUB-BUNDLE DESCRIPTION

The fuel sub-bundle design is shown in Figure 2-5. The fuel bundle consists of four subbundles, each with 24 fuel rods arranged in a 5x5-1 lattice. In each sub-bundle, [

]^{a,c}. Each sub-bundle is assembled as a separate unit with its own top and

bottom tie plates and is held together by two tie rods. [

]^{a,c}

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

]^{a,c}

2.2.1 Top and Bottom Tie Plates

The top tie plate of each sub-bundle has an individual identification number engraved on the side. This feature is also used for administrative control purposes to assure the correct placement of the sub-bundle in the channel. [

]^{a,c}

2.2.2 Standard Fuel Rods and Tie Rods

A typical full-length standard fuel rod is shown in Figure 2-8a. The tie rod is shown in Figure 2-9.

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 seal welded at the ends to encapsulate the uranium fuel. [

]^{a,c} They are fabricated from enriched or natural 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 [ ]a,c.

The top of the fuel rod has a plenum to accommodate fission gases as they are released from the pellets during irradiation. An [ ]a,c spring is located in the upper 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 [ ]a,c 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 sub-bundle length. [ ]a,c

Cladding

The cladding tube is an [ ]a,c Zircaloy-2 tube. The final surface treatment of the inner diameter of the tubes is [ ]a,c

Westinghouse introduced the LK3 cladding for BWR fuel rods in 1994. The LK3 cladding was developed based on extensive experience obtained from earlier manufacturing processes for Zircaloy-2 cladding, such as LK0 (employed during the period 1970-1980), LK1 (1981 to 1986), LK2 [ ]a,c

The cladding tubes are manufactured according to specifications and procedures which produce optimum corrosion resistance. In the LK3 process [ ]a,c. The excellent corrosion performance of tubing manufactured with the LK3 process has been verified by several years of operation in a variety of BWRs.

Part-length Rods

As shown in Figure 2-8b, the part-length rod design is essentially the same as the full-length rods shown in Figure 2-8a with two differences:

[

]a,c and may be deleted in a later design.

[

]a,c

2.2.3 Spacer Capture Rods

The spacer capture fuel rod is shown in Figure 2-10. A spacer capture rod is [

]a,c

The tab welding process is performed such that the inside surface of the clad is not affected. This type of welded tab has been routinely 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 Optima2 is designed to [

]a,c A sketch of the fuel pellet is shown in Figure 2-12.

The pellet sintering process utilized by Westinghouse is designed to minimize in-pile fuel pellet densification. [

]a,c

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

]a,c

2.2.5 Spacers

The SVEA-96 Optima2 spacer is shown in Figure 2-13. The spacer design is based on earlier Westinghouse SVEA-96 Optima and SVEA-96/96+ grid cell designs and utilizes [ ]^{a,c}.

The spacer grid consists of [ ]^{a,c}

The spacer is designed [ ]^{a,c}

The lower end of the frame is designed to [ ]^{a,c}

The spacers are fabricated from strip material [ ]^{a,c}

The spacer design is well proven. The basic SVEA [ ]^{a,c} design was used in some 8x8 open-lattice designs and has been used in all SVEA designs. The first four SVEA assemblies (SVEA-64) were loaded in 1981. 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. The differences between the Optima2 spacer and the SVEA-96/96+ spacers are the addition of mixing vanes and dimension changes to accommodate the small change in fuel rod diameter and envelope for the Optima2 design.

2.3 SVEA-96 OPTIMA2 FUEL CHANNEL

The fuel channel consists of a [ ]^{a,c} inlet piece and a Zircaloy channel. The inlet piece is composed of a transition piece and bottom support which can be equipped with an integrated debris filter and is bolted to the channel with four screws made of [ ]^{a,c} material. The Zircaloy channel consists of an outer channel with a square cross section and an internal double-walled, cruciform structure, or "watercross," which forms channels for non-boiling water as shown in Figures 2-2 and 2-6.

The watercross structure is composed of a square central water channel and smaller water channels in each of the four wings as shown in Figure 2-2. The watercross structure, along with the outer channel walls, form four sub-channels in which the sub-bundles are positioned. A screw is welded to the top end of the cross which attaches the handle with the leaf spring to the fuel channel. The wall thicknesses of the outer channel and watercross are [ ]a,c, respectively.

[

]a,c

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 [

]a,c

The channel and inlet transition piece is designed to assure compatibility with the reactor internals as well as other fuel types in the core. The outer envelope of the SVEA-96 Optima2 channel and transition piece provide ample clearance for control rods and in-core instrumentation for the BWR/2 through BWR/6 reactor designs. The improved dimensional stability of the SVEA channel assures that ample clearances are maintained even at high burnups. The length of the assembly is compatible with the relative positions of the fuel support piece and upper core grid. The inlet transition piece matches the existing fuel support piece and has been designed to be compatible with existing refueling equipment. A detailed mechanical compatibility evaluation is performed as part of the standard design process for each application to assure mechanical compatibility.

SVEA channels were first introduced for reactor operation in 1981. An extensive inspection and measurement program has been performed on SVEA channels during the annual reactor shutdowns. These inspections have verified that the SVEA channels behave as expected and have excellent dimensional stability.

2.3.1 Debris Filter

Experience in BWRs over the past several years has demonstrated the need for increased protection from debris fretting. Therefore, Westinghouse has developed an advanced debris filter for SVEA-10x10 fuel referred to as the TripleWave debris filter.

The objective of the debris filter is to prevent debris with a potential for damaging assembly components from entering the fuel bundles, and thus, to minimize the risk of fretting fuel failure. The [

]^{a,c}

The bottom support is designed with [

]^{a,c} which provides a robust and redundant design.

The TripleWave debris filter is designed with an emphasis on capturing [

]^{a,c}

The name TripleWave debris filter refers to [

]^{a,c}

Debris trapping, insulation clogging, and endurance (fretting) tests have been performed to confirm the performance and reliability of the TripleWave debris filter.

The debris trapping efficiency is increased significantly compared with previous filter designs. These tests confirmed that the TripleWave filter [

]^{a,c}

A clogging test in which [

]^{a,c}

Full-scale endurance testing has been performed on a SVEA-96 Optima2 fuel assembly equipped with a TripleWave debris filter. Inspection of the assembly after the testing demonstrated that the [

]^{a,c}

Therefore, it is concluded that the TripleWave debris filter provides very efficient protection from potentially damaging debris and will not negatively impact assembly or reactor operation.

2.4 OFFSET OF THE SVEA-96 OPTIMA2 ASSEMBLY

The SVEA-96 Optima2 assembly may be offset into the control rod gap to provide a more uniform lattice and thereby improve nuclear performance in D-lattice plants. [

]^{a,c}

The increased reactivity and reduced neutron flux gradients associated with the offset provide substantial advantages. Specific mechanical characteristics that support offsetting the channel were incorporated into the SVEA design. [

]^{a,c}

An extensive full-scale test program was performed by Westinghouse. [

]^{a,c}

All SVEA fuel assemblies installed in Swedish and Finnish reactors with asymmetric inter-assembly gaps have been displaced toward the control rod gap. The Nordic experience has been very good. No impact on the control rod motion has been observed. [

]^{a,c}

2.5 ADVANCED FEATURES

Westinghouse engages in a continuing, evolutionary process of improving the BWR fuel assembly design and performance to satisfy changing and more demanding industry requirements. Primary emphasis is placed on improving fuel reliability while enhancing fuel performance.

Section 2.5 of Reference 1.0 described advanced features in the 1994-1995 time frame. Since that time, the debris filter described in Section 2.5.1 of Reference 1.0 has evolved to the TripleWave debris filter described in Section 2.3.1. [

]^{a,c} The SVEA-96+ design described in Section 2.5.4 of Reference 1.0 replaced the earlier SVEA-96 design and has evolved to the SVEA-96 Optima2 design described in Sections 2.1 through 2.4. The [

]^{a,c}. The adoption and improvement of these features, considered to be advanced in the 1994 time frame, as well as the SVEA-96 Optima2 assembly described in Sections 2.1 through 2.4, are examples of the prudent, evolutionary Westinghouse design improvement process based on proven materials and principals.

We anticipate that several advanced features for the SVEA-96 Optima2 design which are in varying stages of testing and qualification will be made available for reload application in the U.S. market within the next few years. Therefore, some of these features are described in the following sections. Each of the features is intended to further improve fuel reliability.

2.5.1 Mechanical Improvements to SVEA-96 Optima2

Measures to improve any design are constantly being evaluated. Based on experience with SVEA-96 Optima2, several relatively minor modifications to the design discussed in Sections 2.1 through 2.4 are currently being considered to improve the performance of the SVEA-96 Optima2 assembly and to improve the manufacturing process in the continuing effort to optimize manufacturing Quality Assurance. Examples of the features being considered can be summarized as follows:

1. [

]^{a,c}

2. Examples of streamlining the design include:

 [

 ]^{a,c}

3. [

]^{a,c}

4. Debris resistant spacer

 This measure introduces some mechanical changes to the spacer to further reduce the potential for trapping debris.

Confirmation and qualification of these improvements is currently in progress. It is anticipated that these improvements will be introduced in the SVEA-96 Optima2 design in the next few years.

2.5.2 Beta-Quenched Fuel Channels

BWR channel bow can reduce thermal margins, negatively impact assembly and core reactivity, and reduce mechanical clearance margins. Therefore, reduction or elimination of BWR channel bow has been a continuing design priority. Historically, channel bow has been reduced by assuring that opposite sides of the channels have matched microstructures and by careful channel management.

Westinghouse has developed a process for using [ ]^{a,c}. The radiation-induced growth of channels fabricated from Beta-quenched Zircaloy is substantially reduced relative to channels without a Beta-quenched microstructure. The Beta-quenched material has a random texture without a preferred direction of irradiation-induced growth. Bowing is primarily caused by differences in growth between opposite sides of a channel. Therefore, channel bow associated with Beta-quenched channels is substantially reduced.

[ ]^{a,c}

2.5.3 Cladding Improvements

Based on experience, Westinghouse engages in a continuing program to improve fuel reliability with a zero failure goal. This section contains some improvements to the cladding currently under consideration.

In the past, Westinghouse has [ ]^{a,c} This characteristic becomes particularly important after a primary failure when there is a potential for relatively large hydrogen generation and pick-up in the cladding. This potential improvement is currently being evaluated, [ ]^{a,c}

As explained in Section 2.2.2, the alloying composition of the Westinghouse LK3 cladding has been [ ]^{a,c}

[

]a,c

2.5.4 ADOPT Pellets

Westinghouse has also been engaged in a continuing program to identify improvements to the current BWR pellet design and UO_2 material characteristics to [

]a,c

In support of this program, [

]a,c

a,c

Figure 2-1 SVEA-96 Optima2 Fuel Assembly

a,c

Figure 2-2 SVEA-96 Optima2 Fuel Assembly Cross Section

a,c

Figure 2-3a SVEA-96 Optima2 Assembly and Control Rod Orientation in a C-lattice Plant

a,c

Figure 2-3b Typical Control Gap Dimensions with SVEA-96 Optima2 Fuel

a,c

Figure 2-4 Fuel Assembly Lattice

a,c

Figure 2-5 SVEA-96 Optima2 Fuel Sub-bundle

a,c

Figure 2-6 SVEA-96 Optima2 Channel

a,c

Figure 2-7 SVEA-96 Optima2 Mounting of Handle with Leaf Spring

a,c

Figure 2-8a Standard Fuel Rod

a,c

Figure 2-8b Part-Length Fuel Rods

a,c

Figure 2-9 Tie Rod

a,c

Figure 2-10 Spacer Capture Rod

a,c

Figure 2-11 Typical Internal Compression Springs Used for the Various Rod Length

a,c

Figure 2-12 UO_2 and UO_2-Gd_2O_3 Pellet Dimensions

a,c

Figure 2-13 SVEA-96 Optima2 Spacer

a,c

Figure 2-14 TripleWave Debris Filter

a,c

Figure 2-15 SVEA-10x10 Channel Bow Measurements in a Symmetric Lattice Plant

3 DESIGN CRITERIA

The only change in this section relative to the corresponding section in Reference 1.0 is a clarification and an update of the reference numbers in the first two sentences to read as follows:

The principal objective of the fuel assembly mechanical design is to meet the acceptable fuel design limits of General Design Criteria (GDC) 10, the rod insert ability requirements of GDC 27, and the core coolability requirements of GDC 35 (Reference 1.5). To accomplish these objectives the fuel is designed to meet the acceptance requirements outlined in Standard Review Plan (SRP), Section 4.2 (Reference 1.4), relative to:

The remainder of the section is unchanged.

3.1 DESIGN CRITERIA, GENERAL

3.1.1 Normal Operations and AOOs

Criterion

This section is unchanged relative to the corresponding section in Reference 1.0.

Discussion

The only change in this section relative to the corresponding section in Reference 1.0 is to change a typographical error as follows:

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 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

This section is unchanged relative to the corresponding section in Reference 1.0.

3.1.2.1 Fuel Rod Mechanical Failure

Criterion

This section is unchanged relative to the corresponding section in Reference 1.0.

Discussion

This section is unchanged relative to the corresponding section in Reference 1.0.

3.1.2.2 Fuel Coolability

Criterion

The changes in this section relative to the corresponding section in Reference 1.0 are modifications of Items 1 and 2 to include the 17% oxidation limit in Item 1 and anticipate a possible change in the CRDA acceptance requirement in Item 2 to read as follows: The remainder of the section is unchanged.

1. Cladding Embrittlement is limited by requiring that the Peak Clad Temperature (PCT) during a postulated LOCA be less than 1204°C (2200°F), and the calculated local oxidation of the cladding shall nowhere exceed 0.17 times the local cladding thickness before oxidation.

2. The fuel assembly design must be such that unacceptable melting, fragmentation, and dispersal of the fuel do not occur during a postulated Control Rod Drop Accident (CRDA). Specifically, limits on the peak fuel enthalpy must be in compliance with NRC requirements.

The remainder of the section is unchanged.

Discussion

The only change in this section relative to the corresponding section in Reference 1.0 is some clarification and the replacement of ABB with "Westinghouse" to read as follows:

During normal operation and Anticipated Operational Occurrences (AOOs) the maintenance of a coolable geometry is assured by the conformance with the design criteria in Sections 3.2 and 3.3.

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

The Westinghouse methodology for evaluating the consequences of a BWR CRDA and an illustrative application for a SVEA-96 core during a CRDA is described in Reference 3.4.

The Westinghouse methodology for evaluation of the consequences during a seismic plus LOCA event is given in Reference 3.1.

3.1.2.3 Clad Bursting

Criterion

This section is unchanged relative to the corresponding section in Reference 1.0.

Discussion

The only change in this section relative to the corresponding section in Reference 1.0 is the replacement of "ABB" by "Westinghouse" to read as follows:

The Westinghouse 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 only change in this section relative to the corresponding section in Reference 1.0 is a modification to anticipate a possible change in the CRDA acceptance requirement to read as follows:

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 threshold(s) must be in compliance with NRC requirements.

Discussion

The only change in this section is elimination of an inappropriate reference to dryout to read as follows:

The Westinghouse methodology for evaluating the consequences of a BWR CRDA and an illustration of the application methodology are described in Reference 3.4.

3.1.3 Evaluation Methodology

Criterion

This section is unchanged relative to the corresponding section in Reference 1.0.

Discussion

This section is unchanged relative to the corresponding section in Reference 1.0.

3.1.4 New Design Features

Criterion

This section is unchanged relative to the corresponding section in Reference 1.0.

Discussion

The only change in this section relative to the corresponding section in Reference 1.0 is the replacement of "ABB" by "Westinghouse" to read as follows:

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, Westinghouse 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

This section is unchanged relative to the corresponding section in Reference 1.0.

Discussion

The only change in this section relative to the corresponding section in Reference 1.0 is updating the reference to fuel assemblies to include recent designs to read as follows:

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 various 10x10 SVEA designs to date discussed in Section 7, the primary thrust has been on a generic post-irradiation inspection program.

3.1.6 New Safety Issues

Criterion

The only change in this section relative to the corresponding section in Reference 1.0 is the replacement of "ABB" by "Westinghouse" to read as follows:

Each new safety issue identified by Westinghouse or the NRC, which is related to fuel, will be evaluated relative to the existing Westinghouse 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

This section is unchanged relative to the corresponding section in Reference *1.0.*

3.1.8 Burnup

Criterion

This section is unchanged relative to the corresponding section in Reference 1.0.

Discussion

The only change in this section relative to the corresponding section in Reference 1.0 is the replacement of "ABB" by "Westinghouse" to read as follows:

An important aspect of the Westinghouse 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. Westinghouse 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 database.

3.2 DESIGN CRITERIA, FUEL ASSEMBLY COMPONENTS

This section is unchanged relative to the corresponding section in Reference 1.0.

3.2.1 Compatibility with Other Fuel Types and Reactor Internals

This section is unchanged relative to the corresponding section in Reference 1.0.

3.2.2 Geometric Changes in the Assembly during Operation

This section is unchanged relative to the corresponding section in Reference 1.0.

3.2.3 Transport and Handling Loads

This section is unchanged relative to the corresponding section in Reference 1.0.

3.2.4 Hydraulic Lifting Loads during Normal Operation and AOOs

This section is unchanged relative to the corresponding section in Reference 1.0.

3.2.5 Stress and Strain during Normal Operation and AOOs

Criterion

The only change in this section relative to the corresponding section in Reference 1.0 is an update of the reference numbers to read as follows.

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 1.3 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 Reference 1.3. 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

This section is unchanged relative to the corresponding section in Reference 1.0.

3.2.7 Fretting Wear of Assembly Components

This section is unchanged relative to the corresponding section in Reference 1.0.

3.2.8 Corrosion of Assembly Components

This section is unchanged relative to the corresponding section in Reference 1.0.

3.2.9 Hydriding of Zircaloy Assembly Components other than Fuel Rods

This section is unchanged relative to the corresponding section in Reference 1.0.

3.3 DESIGN CRITERIA, FUEL RODS

3.3.1 Rod Internal Pressure

Criterion

The design criterion for rod internal pressure is unchanged relative to the criterion in Reference 1.0.

Discussion

The only change in this section relative to the corresponding section in Reference 1.0 is the replacement of "ABB" by "Westinghouse" to read as follows:

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 Westinghouse 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

The only change in this section relative to the corresponding section in Reference 1.0 is updating of a reference:

Fuel rod stresses must be maintained within acceptable limits. This criterion is implemented by establishing design limits for stresses in accordance with Reference 1.3 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 licensing analysis design criterion for cladding strain is unchanged relative to the criterion in Reference 1.0. However, the wording of the 1% cladding strain requirement has been clarified, and the internal Westinghouse guidance on total strain have been deleted, as follows:

The total transient induced elastic and plastic cladding circumferential strain should not exceed 1%. In this context, total transient induced 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.

Discussion

This section has been revised relative to the corresponding section in Reference 1.0 to replace "ABB" with "Westinghouse" and delete reference to the internal Westinghouse guidance on total strain to read as follows:

This criterion results 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.

3.3.4 Hydriding

Criterion

The design criterion for cladding hydriding is unchanged relative to the criterion in Reference 1.0. The wording of this section has been clarified as follows:

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

Discussion

The wording of this section has been clarified relative to the corresponding section in Reference 1.0 as follows:

This design criterion augments the 1% transient 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 cladding oxidation will limit clad hydriding and, concomitantly, limits the loss of ductility associated with hydriding.

3.3.5 Cladding Corrosion

This section is unchanged relative to the corresponding section in Reference 1.0.

3.3.6 Cladding Collapse (Elastic and Plastic Instability)

This section is unchanged relative to the corresponding section in Reference 1.0.

3.3.7 Cladding Fatigue

This section is unchanged relative to the corresponding section in Reference 1.0.

3.3.8 Cladding Temperature

This section is unchanged relative to the corresponding section in Reference 1.0.

3.3.9 Fuel Temperature

This section is unchanged relative to the corresponding section in Reference 1.0.

3.3.10 Fuel Rod Bow

The design criterion for fuel rod bow has been clarified relative to the criterion in Reference 1.0 as follows:

Criterion

Excessive fuel rod bowing shall be precluded for the design life of the fuel assembly. Fuel rod bowing shall be evaluated, and the impact on fuel rod performance shall be accounted for, if necessary, in the thermal and mechanical evaluation of the fuel rods and the assembly. Fuel rod bow shall not lead to loss of integrity due to cladding overheating.

Table 3.1 has been deleted and reinserted as Table 4-1 since it is discussed in Section 4.

4 DESIGN METHODOLOGY AND APPLICATION

Section 4 has been changed relative to Section 4.0 of Reference 1.0 to incorporate editorial changes, such as reference to Westinghouse rather than ABB, to clarify the methodology as necessary, including the accommodation of the RAIs and TER/SER comments and conditions from Reference 1.0, and to provide a sample application of the methodology to the SVEA-96 Optima2 design demonstrating satisfactory performance to 62 MWd/kgU rod-average. In addition, Section 4.4 is a new section relative to Reference 1.0 which has been added to describe the STAV7.2 initialization of accidents and transients provided in RAI A2 of Reference 1.0.

This section provides the Westinghouse 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. Similarly, if appropriate conditions such as core and plant operating conditions and design, burnup requirements, and assembly design do not change substantially, a single evaluation can be applied to more than one plant for many of the criteria. Therefore, whenever possible, sufficiently conservative conditions are assumed for to accommodate conditions from cycle-to-cycle for each plant or for more than one plant.

In addition to the methodology description, the Westinghouse methodology described in this report is applied to the SVEA-96 Optima2 assembly 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 Optima2 design. It should be noted that the design criteria in Section 3 and the methodology in this section are general and can be applied to any BWR assembly for which the supporting information is available.

The sample design evaluations demonstrate that the criteria are satisfied up to a [ ]^{a,c}.

This section is organized in the same manner as Section 3. The evaluation methodology for any assembly design and the sample application to SVEA-96 Optima2 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. The correspondence between the subsection numbers in Sections 3.2/4.2 and 3.3/4.3 are 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 Optima2 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 Westinghouse 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 4.7, Westinghouse materials specifications, Westinghouse measurement data, and data provided by suppliers. The material properties for the fuel cladding and UO_2 and UO_2-Gd_2O_3 fuel pellets used in the fuel rod performance evaluations are discussed in Reference 1.2.

Typical properties for unirradiated Zircaloy and Stainless Steel components currently used for the fuel assembly design evaluations are provided in Table 4-1.

[

]^{a,c}.

When unirradiated values are utilized for irradiated components, the effects of irradiation are treated conservatively. For example, conservative estimates of the increase in outer channel and watercross peak stresses associated with wall thinning due to corrosion are assumed. However, the yield and tensile strengths are expected to increase by factors of [

]^{a,c}

Design Stress Intensities

Mechanical properties, such as those discussed in Table 4-1, 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 1.3. [

]^{a,c}

[

]^{a,c}

The design stress intensity, S_m, for [

]^{a,c}

The design stress intensity, Sm, [

]^{a,c}

$Rp_{0.2}$ is the 0.2% offset yield strength. [

]^{a,c}

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 Westinghouse is utilizing [

]^{a,c}

Sample design stress intensities, S_m, are shown in Table 4-1 and are derived in this manner and based on the mechanical properties which are also provided.

The fuel assembly structural component stresses under accident conditions are evaluated using the methods outlined in Appendix F Reference 1.3. The stress intensities (S_m) are defined in accordance with the rules described above for normal operating and anticipated operational transient conditions. [

]^{a,c}

[

]^{a,c}

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 1.3.

Unless otherwise stated, stress intensities are calculated with the Tresca criterion specified in the Reference 1.3:

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

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 to provide 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. 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

This section is revised to read:

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

This section is unchanged relative to the corresponding section in Reference 1.0.

4.2.1 Compatibility with Other Fuel Types and Reactor Internals

The technical aspects of the methodology in this section are not significantly different than those in Reference 1.0. Editorial changes relative to Section 4.2.1 of Reference 1.0, such as reference to Westinghouse rather than ABB, and some clarifications of the methodology have been incorporated.

The sample evaluation for SVEA-96 in Reference 1.0 is replaced with a sample evaluation for SVEA-96 Optima2.

Methodology

For each plant application of a Westinghouse fuel assembly type (e.g. SVEA-96 Optima2) 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 Westinghouse fuel assembly type and other resident adjacent fuel assembly types over the design life of both fuel assembly types is performed. [

   ]^{a,c}

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. [

   ]^{a,c}

Creep Deformation

[

]^{a,c}

[

]a,c

Channel Bow

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

[

]a,c

A feature of the Westinghouse 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 grid 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:

[

]^{a,c}

Sample Application

This section contains an example of the methodology for evaluating compatibility in a mixed core by evaluating the SVEA-96 Optima2 assembly in a C-lattice in a BWR/6 type plant equipped with 3810 mm (150-inch) active fuel. The resident fuel to which the SVEA-96 Optima2 fuel must be compatible is referred to as the "non-SVEA" fuel assembly.

1. Geometrical Compatibility with Other Fuel Types in the Core

[

]^{a,c}

[

]a,c

[

]a,c

SVEA-96 Optima2 at EOL / Non-SVEA Assembly at EOL

[

][a,c]

Therefore, the SVEA-96 Optima2 assembly is concluded to be compatible with the resident non-SVEA assembly with regard to axial growth.

[

][a,c]

The SVEA-96 Optima2 handle leaf spring provides a nominal force of [ ][a,c]. This corresponds to a stress of [ ][a,c] which is well below the yield stress of [ ][a,c] shown in Table 4-1. [ ][a,c]

[

]a,c

This example demonstrates the compatibility of the SVEA-96 Optima2 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 Optima2 assembly and control rod orientation for a full core of SVEA-96 Optima2 fuel in a BWR/6 C-Lattice plant is shown in Figure 2-3a and 2-3b. 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 [ ]a,c The available minimum space for the detector is [ ]a,c when surrounded by SVEA-96 Optima2 assemblies at beginning-of-life (BOL) (Figure 2-3a). The width of the control blade in this example is 8.33 mm at the blade location and 10.1 mm at the control rod roller location.

As noted above, the maximum SVEA-96 Optima2 channel dimension on a side at BOL is [ ]a,c. From Figure 2.3a, this maximum dimension provides at least [ ]a,c Therefore, adequate clearances are available at BOL to avoid interference.

The effects of irradiation on the SVEA-96 Optima2 channel dimensions and the resulting effects on compatibility with the control rods and detectors are considered by evaluating the channel bulge and bow.

Channel Bulge

The following example illustrates the impact of channel bulge due to the pressure differential across the channel to a bundle burnup [ ]a,c

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. [

]a,c

[

]^{a,c}

Since the correlation has [

]^{a,c}

Application of the creep model described above demonstrates that for the SVEA-96 Optima2 channel the combination of the axial variations of [

]^{a,c}

a,c

[

]^{a,c}

a,c

[

]^{a,c}

Channel Bow

[

]^{a,c}

[

]a,c

While the operational experience of SVEA fuel in asymmetric lattice reactors is extensive with no indications of SVEA fuel jeopardizing control rod maneuverability, the maximum burnup of the measured SVEA channel bow database for asymmetric lattice reactors is not quite as high as it is for symmetric lattice plants.. The Zry-2 channel average bow and standard deviation has been evaluated for the available database for asymmetric reactors. The average channel bow is close to -1 mm (away from the control rod) with a and the standard deviation of 2.5 mm at a burnup of 45 MWd/kgU. This is similar to the symmetric data with the exception of the average bow is closer to -1 mm rather than zero mm.

The SVEA fuel channel bow in asymmetric lattice data are plotted in Figure 4.2-6. Comparing this with the data from symmetric lattice reactors in Figure 2-15 it can be seen that the channel bow behaviour is similar and therefore, the asymmetric data can be extrapolated to higher burnups based on the higher burnup experience from symmetric lattice reactors.

For example, available asymmetric and symmetric lattice data has been used to infer asymmetric lattice bow to an assembly burnup of 55 MWd/kgU. This evaluation that the average bow for symmetric lattice varies very little and is close to zero. For an asymmetric lattice the average bow is close to -1 mm, and based on the experience from symmetric lattice plants it is expected to vary

very little with burnup. From the symmetric lattice data it is also concluded that the spread in channel bow varies very little up to a certain burnup, referred to below as the "break point". Above the break point, the increase in spread (standard deviation) is linear. The break point for symmetric lattice data is at a burnup of approximately 35 MWd/kgU. For asymmetric lattice the break point is at approximately 25 MWd/kgU and the standard deviation is approximately 1 mm. The standard deviation increases to approximately 2.5 mm at a burn up of 45 MWd/kgU. According to experience from bow in symmetric lattice plants, the increase in spread is linear above the break point. The expected channel bow standard deviation for an asymmetric lattice at 55 MWd/kgU can then be determined as follows:

> The increase is from 1 to 2.5 mm for an increase in the burnup from 25 to 45 MWd/kgU. That is, a standard deviation increase with 1.5 mm corresponds to an increase in burnup of 20 MWd/kgU. As the standard deviation is 1 mm at a burnup of 25 MWd/kgU, the expected standard deviation at 55 MWd/kgU will be: 1 + (55-25)/20 x 1.5 = 1 + 2.25 3.5 mm. This expected spread together with the expected average bow, -1 mm, at 55 MWd/kgU could result in 2.5 mm (-1 + 3.5 = 2.5 mm) channel bow toward the control rod. This can be compared with 2 mm bow toward the control rod for the symmetric lattice (see above). As the control rod clearance is more than 0.5 mm larger in an asymmetric lattice, the margin for interference is larger than for a symmetric lattice.

Moreover, the increased dimensional stability of the SVEA-96 Optima2 channel and its greater flexibility substantially reduces the risk of unacceptable control rod interference relative to open lattice 8x8 or 9x9 designs. The reduced frictional force exerted on a control rod by the SVEA channel design relative to traditional open lattice fuel was confirmed by tests performed in the Westinghouse BURE test facility in the early 1980s on a SVEA assembly with a 1.4 mm-thick outer channel and 8x8 open-lattice fuel with a 2.3 mm-thick channel. These tests also demonstrated that very substantial interference would be required for a SVEA channel to affect control rod insertability. For example, it was found in these tests that about 4.5 mm of interference on the control rod guides was required to achieve two-thirds of the force available to withdraw a control rod.

Furthermore, the experience with SVEA fuel and reduced control rod gaps in Westinghouse reactors is very extensive and no case of control rod maneuverability problems due to the SVEA fuel have been indicated or reported. Therefore, it is concluded that SVEA-96 Optima2 fuel in C- and D-lattice BWR reactors will not pose a risk of jeopardizing control rod maneuverability.

The SVEA-96 Optima2 channel could bow sufficiently to contact an instrument guide tube. However, the relatively flexible SVEA channel will not damage the instrument guide tube, and operational experience to date has not indicated that channel bow adversely affects the operation of the in-core instrumentation.

Therefore, these examples demonstrate the compatibility of the SVEA-96 Optima2 assembly with control rods and detectors in symmetric and asymmetric lattice plants. Similar compatibility evaluations are performed for each new plant application.

3. Geometrical Compatibility with Other Core Components

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

[

]a,c

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. [

]a,c

In this manner compatibility of the SVEA-96 Optima2 assembly with the upper core grid and fuel support piece is assured.

4. Geometric Compatibility with Storage Facilities

[

]a,c

4.2.2 Geometric Changes in the Assembly During Operation

Methodology

For each plant application of a Westinghouse fuel assembly type (e.g. SVEA-96 Optima2), 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. The following considerations are typical and address the SVEA-96 Optima2 design for the design lifetime of the assembly:

1. [

]a,c

[

]a,c

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

      [
      ]a,c

   b. Assembly Handle Configuration

      [
      ]a,c

   c. Spacer Capture Rod

      [
      ]a,c

   d. Spacer

      [
      ]a,c

e. External Compression Spring

[

]^{a,c}

A feature of the Westinghouse methodology when applied to Westinghouse 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-Westinghouse 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 Optima2 assembly components as a function of burnup. [

]^{a,c}

The fuel rod growth can be a result of different contributions, e.g. anisotropic creep down, pellet cladding contact, cladding hydriding and stress free growth.

The pellet cladding contact contribution to the rod growth can increase for SVEA-96 Optima2 fuel due to the reduced pellet – cladding gap, the reduced cladding wall thickness and the increased pellet density. This can increase the rod growth and also the differential rod growth compared to SVEA-96/100 fuel at the same conditions. It is conservatively estimated that at an assembly burnup of 70 MWd/kgU the maximum rod growth could increase by as much as 70%, and the differential rod growth by 50% for SVEA-96 Optima2 compared with SVEA-96 for the same conditions.

1. Sub-bundle Growth

The differential growth between the SVEA-96 Optima2 channel and subbundles based on the most current data base can be summarized as follows:

[

]^{a,c}

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 is summarized below and the design limits are shown in Figure 4.2-10:

[

]^{a,c}

[

]^{a,c} Thus, the requirements on space for unrestricted rod growth, and rod guidance in the top tie plate is fulfilled.

b. Standard Fuel Rods

[

]^{a,c} The requirement on space for unrestricted rod growth, and rod guidance is thus fulfilled.

c. Part Length Fuel Rods

[

]^{a,c}

[

]^{a,c}

A similar evaluation demonstrates that the margin for the part length rod (1/3) are larger than those for the two-third length rods.

Also in the worst case according to the above, the requirement on rod guidance in the spacer and absence of fretting is fulfilled.

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

   [

   ]^{a,c}

[

]^{a,c}

b. Assembly Handle Configuration

[

]^{a,c}

c. Spacer Capture Rod

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

[

]^{a,c}

[

]^{a,c}

d. Spacer

[

]^{a,c}

Spacers with the same general design and the same material as the SVEA-96 Optima2 spacer have been used in 8x8, SVEA-64, SVEA-100, SVEA-96/96+ and SVEA-96 Optima assemblies. Reactor experience with over 31,000 assemblies has not shown any indication of stress corrosion cracking or fatigue failure. [

]^{a,c} Furthermore, laboratory tests described in Section 8 demonstrate that the spacer can withstand repeated seismic-type loads. [

]^{a,c}

Therefore, reactor experience with the SVEA-96 Optima2 spacer, as well as very similar spacer designs, 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 general external compression spring design and spring material has been utilized in the 8x8, SVEA-64, SVEA-100, SVEA-96/96+ and SVEA-96 Optima designs as well as

in the SVEA-96 Optima2 assemblies. Therefore, as indicated in Section 7, experience with these springs has been extensive. [

]^{a,c} Therefore, reactor experience with the SVEA-96 Optima2 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 Westinghouse 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.

[

]^{a,c}

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:

[

]^{a,c}

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 1.3. [

][a,c]

Sample Application

The current design loads for shipping and handling of SVEA-96 Optima2 fuel for U.S. applications can be summarized as follows:

a,c

Sample Evaluation of Response to Shipping Loads - SVEA-96 Optima2

Shipping tests have been performed in both the U.S. and Europe to qualify the current shipping methods of SVEA assemblies, [

][a,c]

[

]^{a,c}

Subsequent to the transport test, the inner steel container went through a handling sequence (i.e., shock tests) to verify the acceptable shock limits. These handling test included:

[

]^{a,c}

Prior to testing the fuel assembly components were carefully inspected and characterized for later comparisons. Furthermore, a sub-assembly was disassembled and the spacers were inspected again.

After completion of the test, the subbundles were disassembled, and the spacers and rods were carefully examined. All components were visually inspected, including Gamma scanning of rods and visual inspection of pellets. The examination after these tests showed no indication of unacceptable deformation of the fuel assembly components. Small dimensional changes on the spacers were observed. However, most of these changes were within the same range of dimensional changes introduced by the assembly/disassembly process, and all spacer square dimensions were within drawing tolerances.

[

]^{a,c}

Sample Evaluation to Response to Handling Loads - SVEA-96 Optima2

The evaluation of the SVEA-96 Optima2 assembly for design handling loads addresses the stresses in the channel assembly, the lifting handle, the tie plates, and the tie rods.

Channel

[

]^{a,c}

[

]^{a,c}

a,c

[

]^{a,c}

Handle

The sample evaluation of the handle is performed on a SVEA-96 Optima2 handle design with dimensions typically used in U.S. BWRs. The sample evaluation of the handle is performed on a SVEA-96 Optima2 handle design with dimensions typically used in U.S. BWRs. A tension test has been performed, in accordance with Reference 1.3 (Experimental Analysis), on the SVEA-96 Optima2 handle to verify that the handle meets the design requirements. [

]^{a,c}

[

]^{a,c}

This result demonstrates that the requirements are fulfilled, and the design requirement with respect to mechanical loads are thus met for the handle.

Tie Plates

[

]^{a,c}

Tie Rods

[

]^{a,c}

The stresses are classified as primary membrane, and the values presented in the table demonstrate that the margin to allowable stress is substantial:

Stresses at 1500 N Load Distributed Equally amongst the two Tie Rods

	Maximum Calculated stress N/mm^2	Maximum Allowable Stress (S_m) N/mm^2
Cladding tube	48	142
Girth Weld	61	142
End plug (bottom)	42	129
Top end plug	43	129

[

]a,c

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.

[

]a,c

[

]a,c

Sample Application

[

]a,c

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. [

]a,c

[

]^{a,c}

Sample Application

The sample application provided is for SVEA-96 Optima2 assemblies in a BWR/6 plant.

Stresses in SVEA-96 Optima2 fuel assembly components have been evaluated for operating loads during normal operation and AOOs for a wide variety BWR plants. [

]^{a,c}

Spacer and External Compression Springs

[

]^{a,c}

[

]a,c

Channel

[

]a,c

a,c

[

][a,c]

[

]^{a,c}

a,c

[

]^{a,c}

It is concluded that the stress limits for the overpressure expected to bound most BWR plant applications are satisfied at both BOL and EOL conditions. It is also concluded that the channel deflections are small and are negligible relative to their potential impact on the function of the assembly.

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. [

]^{a,c}

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 1.3. 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 4.8 or from specific test data obtained and evaluated in accordance with Reference 1.3. Therefore, N_i includes the more limiting of a factor of two on stress and a factor of 20 on the number of cycles as well as the effects of non-zero mean stress.

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

where m is the number of load cycles.

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

Sample Application

The only SVEA-96 Optima2 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 Optima2 channel.

[

]a,c

[

]a,c

a,c

[

]a,c

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.

[

]^{a,c}

Sample Application

The potential for damaging wear in the SVEA-96 Optima2 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 Optima2 prototype loop tests and post irradiation examinations of SVEA-96 Optima2 fuel have demonstrated that wear of SVEA-96 Optima2 components is minimal and does not impair the function of the assemblies.

[

]^{a,c}

as the SVEA-64, SVEA-100, SVEA-96/96+ and SVEA-96 Optima designs. Manufacturing and operational experience with over 23000 assemblies utilizing this general spacer design have demonstrated that assemblies can be fabricated with virtually no chance of damaging spacer cells and that very little wear occurs with this type of spacer design.

Careful maintenance of tolerances in the SVEA-96 Optima2 assembly allows critical clearances to be minimized in a manner which reduces lateral vibrations. For example, the radial clearance between fuel rod end plug extensions and guiding holes in the tie plates is minimized and maintained within tight tolerances to provide hydraulic damping and minimize fuel rod lateral vibration. The success of this feature is demonstrated by the fact that fretting wear has not been observed on the fuel rod end plug extensions or the inner surfaces of the tie plate holes.

Similarly, in order to increase hydraulic damping, the distance between the top tie plate and the channel walls is maintained to a tight clearance at operating temperature. The top tie plate is also equipped with side support surfaces which guide the subbundle in the subchannel and provide and efficient hydraulic damping. No potentially damaging wear has been observed on the tie plate sides or the channel walls in post irradiation examinations or examinations following hydraulic endurance tests.

[

]a,c

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. [

]^{a,c}

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

[

]^{a,c}

Evaluation of the potential for component corrosion in non-Westinghouse 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 Westinghouse experience with the component materials used in the SVEA-96 Optima2 design (Section 5.2.2), the SVEA-96 Optima2 assembly components for which the potential for corrosion must be specifically addressed are:

[

]^{a,c} 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. [

]^{a,c}

[

]^{a,c}

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. Westinghouse has maintained an ongoing program over the past 30 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 Westinghouse fuel assembly components are maintained at a relatively low level as shown in the following table.

a,c

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.

[

]a,c

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 Westinghouse design:

[

]a,c

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

Sample Application

[

]a,c

[

]^{a,c}

a,c

It should be noted that hydriding is only important for areas of the channel which are subjected to substantial stresses during handling. These areas are at the top and bottom sections of the channel for the SVEA-96 Optima2 assembly. Since the channel areas with relatively high stresses during handling operations are a relatively small fraction of the total area, it is judged that the probability that the maximum corrosion would occur at a high stress location is relatively low. Furthermore, the limit of 500 ppm hydrogen is considered to be conservative. Therefore, it is concluded that hydriding of the SVEA-96 [

]^{a,c}.

Table 4-1 Typical Fuel Assembly Material Properties

a,c

Table 4-1 Typical Fuel Assembly Material Properties (cont.)

a,c

a,c

Figure 4.2-1 SVEA Channel Growth

a,c

Figure 4.2-2 SVEA-96 Optima2 Assembly (BOL) and non-SVEA Assembly (BOL)

a,c

Figure 4.2-3 SVEA-96 Optima2 Assembly (BOL) and non-SVEA Assembly (EOL)

a,c

Figure 4.2-4 SVEA-96 Optima2 Assembly (EOL) and non-SVEA Assembly (BOL)

a,c

Figure 4.2-5 SVEA-64 Channel Creep Deformation

a,c

Figure 4.2-6 SVEA-10X10 Channel Bow Measurements in Asymmetric Lattice Plants

a,c

Figure 4.2-7 SVEA-96/100 Fuel Rod Growth

a,c

[ ]

Figure 4.2-8a SVEA-96/100 Differential Fuel Rod Growth

[ ]

Figure 4.2-8b SVEA-96/100 Differential Growth of Tie Fuel Rods

a,c

Figure 4.2-9 SVEA-96 Clearance Between Subchannel and Handle

a,c

Figure 4.2-10 Fuel Rod Growth Allowances

a,c

Figure 4.2-11 SVEA-96 Spacer Spring Relaxation

Figure 4.2-12 FEM Model For SVEA-96 Channel Stress Calculations

a,c

Figure 4.2-13 Calculated SVEA-96 Channel Deflections

a,c

Figure 4.2-14a Maximum SVEA Channel Oxide Thickness

a,c

Figure 4.2-14b Average SVEA Channel Oxide Thickness

4.3 METHODOLOGY AND APPLICATION – FUEL RODS

The rod performance analysis methodology described in this section is substantially the same as the methodology described in Reference 1.0 as modified by the changes made during the NRC review of Reference 1.0. It is described in its entirety in this section for clarity purposes and to accommodate minor revisions based on experience and new information obtained since 1995 and to remove features specific to STAV6.2.

The term Design Power History (DPH) used in Reference 1.0 has been replaced with the term Thermal Mechanical Operating Limit (TMOL) in this document.

This section contains the methodologies for fuel rod design evaluations of the individual fuel rods in the assembly for normal operation and AOOs. Sections 4.3.1 through 4.3.10 describe the methodologies and provide a specific application to SVEA-96 Optima2 for evaluation relative to the design criteria described in Sections 3.3.1 through 3.3.10. In addition, Section 4.3.11 provides a discussion of Westinghouse measures to minimize the probability of Pellet Cladding Interaction (PCI).

The treatment of uncertainties may utilize one of [ ]^{a,c}. All proposed approaches are applied in a manner which assures that adequate margins to design limits are maintained.

1. [ ]^{a,c} If this approach is intended to be used at some point, Westinghouse will submit the details and provide justification.

2. [ ]^{a,c}

3. [ ]^{a,c} This approach is deterministic and is the most conservative.

   Westinghouse will apply an [ ]^{a,c} as described in each application.

A description of the procedure for selection of power histories for limiting fuel rod performance is described in Section 4.3.0.

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 application of specific fuel rod power histories. Therefore, Westinghouse has established a systematic approach for assuring that [

]a,c Selection of limiting power histories and the method of analyses are essentially the same as the methodology approved for STAV6.2 in CENPD-287-P-A. The projected fuel rod power histories are established from plant- and cycle-specific calculations utilizing a three-dimensional nodal simulator and lattice physics codes accepted for referencing in licensing applications by the NRC.

Methodology

Individual limiting power histories and a [

]a,c

Sample Application

[

]^{a,c}

Limiting Assemblies

Assuming that the assemblies composing the equilibrium reload cycle are in the core for the cycles N_o, N_{o+1} ... N_L-1, N_L, fuel rod power histories are selected for evaluation and used in design analyses as follows:

[

]^{a,c}

Base Power Histories

From the [

]^{a,c}

[

]^{a,c}

From the [

]^{a,c} The value for a specific plant and feed fuel assembly design application should be established based on the specific application.

These power histories [

]^{a,c}

Limiting Fuel Rods for Specified Acceptable Fuel Design Limit (SAFDL) and Rod Internal Pressure (RIP) Evaluations

The Westinghouse SAFDLs which are based on LHGR protect against excessive cladding strain and fuel temperature. [

]^{a,c} These limiting power histories are used in the evaluation of fuel rod performance under transient conditions associated with plant maneuvers and Anticipated Operational Occurrences (AOOs).

Power History Envelopes

Using the base power histories, [

]^{a,c}

[

]a,c

Thermal Mechanical Operating Limits (TMOLs)

Define the Thermal Mechanical Operating Limits (TMOLs) for [

]a,c The evaluations are performed with computer codes accepted for referencing in licensing applications by the NRC. The performance of the fuel rod for each application is evaluated for the limiting power histories and/or the TMOL relative to the design bases in Section 3.3 which are sensitive to fuel rod power history. The TMOL is provided to the plant operator in terms of a Linear Heat Generation Rate (LHGR) operating limit which should not be exceeded during normal operation.

[

]a,c The enveloping LHGR and 6 SPH for UO_2 fuel rods are shown in Figures 4.3.0-1 through 4.3.0-6. Similarly, the enveloping LHGR and 6 SPH for UO_2-Gd_2O_3 fuel rods are shown in Figures 4.3.0-7 through 4.3.0-12.

Treatment of Part-Length Rods

[

]a,c

Treatment of AOO Power Ramps

[

]a,c

[

]a,c

The sample applications which depend on fuel rod power history and, therefore utilize the TMOL and these limiting fuel rod power histories are described in subsequent sections.

a,c

Figure 4.3.0-1 UO_2 Envelope and TMOL SPH 1

a,c

Figure 4.3.0-2 UO_2 Envelope and TMOL SPH 2

a,c

Figure 4.3.0-3 UO_2 Envelope and TMOL SPH 3

a,c

Figure 4.3.0-4 UO_2 Envelope and TMOL SPH 4

a,c

Figure 4.3.0-5 UO_2 Envelope and TMOL SPH 5

a,c

Figure 4.3.0-6 UO_2 Envelope and TMOL SPH 6

a,c

Figure 4.3.0-7 UO_2-Gd_2O_3 Envelope and SPH 1

a,c

Figure 4.3.0-8 UO_2-Gd_2O_3 Envelope and SPH 2

a,c

[ ]

Figure 4.3.0-9 UO_2-Gd_2O_3 Envelope and SPH 3

a,c

[ ]

Figure 4.3.0-10 UO_2-Gd_2O_3 Envelope and SPH 4

a,c

Figure 4.3.0-11 UO_2-Gd_2O_3 Envelope and SPH 5

a,c

Figure 4.3.0-12 UO_2-Gd_2O_3 Envelope and SPH 6

4.3.1 Rod Internal Pressure

Methodology

For each plant application, maximum fuel rod internal pressure is evaluated to confirm that the lift-off criterion identified in Section 3.3.1 is not violated. The evaluation is a two step process involving:

1. Calculation of the internal fuel rod pressure required to violate the lift-off criterion is performed. This calculation is a burnup-dependent comparison of the outward creep rate of the cladding with swelling rate of the fuel pellets. A cladding creep correlation and pellet swelling rate accepted by the NRC for licensing applications are used for this purpose. Appropriate uncertainties, such as those associated with fuel rod dimensions, clad creep rate and pellet swelling rate, are accounted for in the calculation to assure that the lift-off pressure is not over-estimated.

2. The fuel rod internal pressure is calculated as a function of burnup to End-of-Life (EOL) using a fuel rod performance code accepted for referencing in licensing applications by the NRC. The calculations are performed for the [

   ]^{a,c}.

The dependence of the maximum fuel rod internal pressure on uncertainties in parameters to which the fuel rod pressure is sensitive is established, and an EOL value encompassing the significant uncertainties is established for comparison with the critical pressure required for fuel rod lift-off established in Step 1. The most limiting value of any parameter with a significant impact on fuel rod pressure, which is not included in the uncertainty evaluation, is utilized in the nominal calculation. Uncertainties in the following parameters are typically considered:

[

]^{a,c}

[

]^{a,c}

Sample Application

This sample application uses the STAV7.2 code described in Reference 1.2 and the cladding creep and fuel pellet swelling models described in Reference 1.2 to evaluate the SVEA-96 Optima2 fuel rod design described in Section 2.

Critical Lift-Off Pressure

The BWR cladding creep correlation in Reference 1.2, Section 2.2.3, is applied to the calculation of critical lift-off pressure. The outward cladding creep rate due to internal pressure is not allowed to exceed the fuel pellet swelling rate. Lift-off pressure is based on conservative dimensional and environmental input data and conservative creep and swelling models. [

]^{a,c} This upper bound internal pressure is required to be less than the critical lift-off pressure.

Solid swelling of the fuel pellet is defined in Reference 1.2, Section 2.1.3. Adjusting to the lower bound swelling rate results in

[

]a,c

Maximum Internal Pressure

[

]a,c The results are compared with the critical pressure established for lift-off.

An RIP analysis was then performed to accommodate the potential impact of Anticipated Operational Occurrences (AOOs) [

]a,c The STAV7.2 code described in Reference 1.2 was used for this evaluation.

[

]a,c

[

]a,c

The results of these calculations are summarized in Table 4.3.1-1.

Table 4.3.1-1 Fuel Rod Maximum Internal Pressures (MPa)					

a,c

The internal fuel rod pressure required for lift-off is [

]a,c

a,c

Figure 4.3.1-1 Irradiation Hardening of BWR Cladding

a,c

Figure 4.3.1-2 Critical NCLO Pressure Limit

4.3.2 Cladding Stresses

Methodology

The basic methodology identified in Reference 1.0 is unchanged

For each plant application detailed stress analysis of the fuel rod is performed [ ]^{a,c}. 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 1.3).

[ ]^{a,c}

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.

[ ]^{a,c}

Sample Application

[ ]^{a,c}

[

]a,c

a,c

[

]a,c

The results for the maximum cladding stress indicate that the limiting stresses occur at the beginning of the life. This is due to the fact that mechanical properties of cladding material, Zircaloy, are strongly affected by neutron fluence (>1 MeV), both the yield stress and the ultimate (tensile) stress increases very rapidly with irradiation. Therefore, the calculated stress design ratio decreases rapidly with time. It indicates that the limiting cladding stress occurs at the beginning of life, and that during the irradiation, the 1% limit on clad strain will be more limiting than clad stress intensities.

[

]a,c

The maximum (over all power histories and rods) calculated stress and stress design ratios, defined as maximum equivalent cladding stress divided by allowable stress limits, for FL-UO2 rods are shown in Table 4.3.2-1, Table 4.3.2-2, and Table 4.3.2-3, for the positions at the spacer, between the spacers and close to bottom end plug, respectively. The stress results are from VIK-3 analysis.

Table 4.3.2-1 Maximum Cladding Stress at the Spacer for FL-UO2 Rod

a,c

Table 4.3.2-2 Maximum Cladding Stress Between Spacers for FL-UO2 Rod

a,c

Table 4.3.2-3 Maximum Cladding Stress Close to Bottom End Plug for FL-UO2 Rod

a,c

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 the stress limits for the SVEA-96 Optima2 fuel rod will be acceptable in any credible BWR application.

4.3.3 Cladding Strain

The methodology identified in Reference 1.0 is unchanged. The internal Westinghouse maximum effective strain limit of 2.5 % has been deleted.

Methodology

For each plant application, 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. [

]^{a,c}

The maximum cladding strains calculated in this manner are compared with the 1% limit on elastic and plastic strain excluding the effects of steady-state creep and irradiation growth.

Sample Application

[

]^{a,c}

a,c

[

]^{a,c}

[

]a,c

The resultant power history for clad strain calculation is shown in Figure 4.3.3-1.

The calculations were performed with the STAV7.2 code described in Reference 1.2

The transient hoop strain from this calculation is shown on Figure 4.3.3-2. [
]a,c

This example demonstrates that ample margins to cladding strain limits are available for peak rod average burnups to 62 MWd/kgU.

a,c

Figure 4.3.3-1 SVEA-96 Optima2 Limiting Strain Power History

a,c

Figure 4.3.3-2 Maximum SVEA-96 Optima2 Transient Cladding Strain

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. [

]^{a,c}

Due to the complexity, and resultant uncertainties, involved in incorporating hydride concentration, distribution, size, and shape directly into stress and strain analyses, the impact of hydrides in the fuel cladding is not specifically treated on a microscopic basis in the fuel cladding stress and strain analyses. Instead, a conservative design limit on hydride concentration in the Zircaloy cladding is established based on available industry and Westinghouse experience and testing. The design lifetime of the fuel rod is restricted such that this limit is not exceeded.

The following measures are taken to minimize the impact of Zircaloy hydriding on the cladding and to establish the rate of hydrogen pick-up in the fuel rod:

[

]^{a,c}

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

Sample Application

This example is for the LK3 Zircaloy-2 cladding currently utilized for the SVEA-96 Optima2 assembly.

[

]a,c

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.

Control of Hydrogen Inside the Fuel Rod

Hydrogen in elemental form, or as an unstable chemical compound, may be trapped in the UO_2 pores in the pellet, absorbed on the pellet surface, or dissolved in the pellet material. The following specifications on SVEA-96 Optima2 fuel rod manufacturing are currently applied to minimize the hydrogen trapped in a sealed fuel rod:

a,c

It should be noticed that the 2 ppm requirement on the water in the entire rod is a very conservative application of the ASTM limit of ≤ 2 ppm hydrogen cited in the Standard Review Plan (Reference 1.5).

Hydrogen Pickup in Service for SVEA-96 Optima2 Cladding

Measurements of hydrogen concentrations in fuel rods in Westinghouse BWR assemblies following plant operation are utilized to establish average hydrogen pick-up rates for design and licensing applications. The data base is updated continuously as additional data becomes available. Recent measurements of the hydrogen content in Westinghouse cladding materials show that the hydrogen pickup is generally low.

For example in 2001, measurements of clad hydrogen have been performed at the Paul Scherrer Institute (PSI) and in Studsvik. These rods were irradiated to rod average burnups of 58 MWd/kgU. The results of the examinations showed hydrogen contents of [

]a,c

[

]^{a,c}

a,c

Figure 4.3.4-1 Typical Westinghouse BWR Cladding Hydriding

4.3.5 Cladding Corrosion

Methodology

The methodology identified in Reference 1.0 is unchanged.

Sample Application

This example is for the LK3 Zircaloy-2 cladding currently utilized for the SVEA-96 Optima2 assembly. Section 4.3.5 of Reference 1.0 describes the LK2 (referred to in Reference 1.0 as LK-II) cladding type principally used in Westinghouse BWR fuel from 1987 until the mid-1990's. Section 2.5.5 of Reference 1.0 describes an advanced version of LK2, referred to as LK2+, which provided improved high burnup application performance and reduced hydrogen pickup relative to LK2. The characteristic feature of LK2 and LK2+, relative to the earlier LK1 product, is a [

]^{a,c}

Experience with the LK0, LK1, LK2, and LK2+, a continuing, focused effort to reduce product variability, and the goal of improving corrosion performance at higher burnups has led to the LK3 cladding type. This experience has allowed [

]^{a,c}.

The LK3 cladding is similar to LK1 with respect to [

]^{a,c}

A substantial cladding corrosion database for a wide variety of operating conditions has been obtained for Westinghouse BWR cladding. Routine oxide thickness measurements are currently performed for fuel assemblies manufactured by Westinghouse containing LK2+ and LK3 cladding in Nordic, continental European, and U.S. plants. These measurements and oxide observations have provided a broad database, which encompasses the entire range of conditions expected in BWRs.

Typical in-pile data for Westinghouse 10x10 BWR fuel showing rod-average and maximum oxide layer thicknesses are shown in Figures 4.3.5-1 and 4.3.5-2, respectively. The data include measurements on SVEA-100, SVEA-96, SVEA-96 Optima and SVEA-96 Optima2 fuel assemblies. Note the data points labeled as "2-life rods" refers to fuel rods removed from fuel assemblies which achieved their normal design EOL burnup, and were reinserted in lower burnup assemblies to achieve the high burnup levels shown.

[

]a,c

a,c

Figure 4.3.5-1 Rod Average Oxide Thickness

a,c

Figure 4.3.5-2 Rod Maximum Oxide Thickness

4.3.6 Cladding Collapse (Elastic and Plastic Instability)

Methodology

The basic methodology identified in Reference 1.0 is unchanged. As discussed in Reference 1.2, the COLLAPS code has been improved to treat finite length pellet-to pellet gaps in the rod.

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.

[

]^{a,c}

Conservative design limits are utilized for both instantaneous and creep collapse to establish the margin to collapse. Minimum design requirements are specified at BOL for instantaneous elastic and plastic collapse based on standard, accepted classical expressions. Margin to creep collapse is evaluated in terms of the ovality increase rate or a maximum ovality indicative of cladding collapse.

Westinghouse also implements manufacturing controls on fuel and cladding to minimize the potential for cladding collapse. Specifically, the fuel rod cladding is controlled to ovality, clad thickness, and strength specifications during the manufacturing. In addition, the thermal stability of the pellets is carefully controlled to assure that unacceptable pellet densification and variations in densification do not occur in service.

Fuel rod cladding is examined for ovality during post irradiation examination of high burnup assemblies after service in reactors to confirm that unacceptable flattening of the cladding is not occurring. To the [

]a,c

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. [

]a,c

The current design limits for SVEA-96 Optima2 fuel can be summarized as follows:

[

]a,c

[

]a,c

The results for instantaneous collapse at BOL for a maximum over pressurization transient and instantaneous and creep collapse for the maximum credible steady-state pressure differential after BOL can be summarized as follows:

Instantaneous Collapse at Beginning-of-Life

[

]a,c

[

]a,c

The margin would be even greater if the pellet support were to be credited.

Collapse Calculations after BOL

The COLLAPS-3.3D code described in Reference 1.2 was used to calculate the cladding ovality as a function of burnup for the limiting conditions provided in Reference 1.0 and above.

[

]a,c

These very conservative examples, therefore, demonstrate that ample margins to cladding collapse are available for any realistic operation for peak rod burnups to 62 MWd/kgU.

a,c

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

4.3.7 Cladding Fatigue

The basic methodology identified in Reference 1.0 is unchanged.

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 [

]^{a,c}.

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.

<u>Fatigue Due to Fuel Rod Power Changes</u>

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.

[

]^{a,c}

[

]^{a,c}

Sample Application

[

]^{a,c}

<u>Example of Fatigue Calculation Due to Fuel Rod Power Changes</u>

<u>Load Follow Cycles</u>

[

]^{a,c}

The STAV7.2 code described in Reference 1.2 was used for this evaluation.

[

]^{a,c}

[

]^{a,c}

An example of plant-specific fatigue analysis was also performed. [

]^{a,c}

Start-Up Cycles

[

]^{a,c}

Example of Fatigue Calculation Due to Hydraulic Forces

[

]^{a,c} which are less than 1.0.

Therefore, the results demonstrate that the SVEA-96 Optima2 fuel design has considerable margin to fatigue failure for any credible reactor operation to peak rod average burnups of 62 MWd/kgU.

4.3.8 Cladding Temperature

The basic methodology identified in Reference 1.0 is unchanged.

Methodology

The Westinghouse 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 Optima2 fuel is documented in Reference 3.5.

4.3.9 Fuel Temperature

The basic methodology identified in Reference 1.0 is unchanged.

Methodology

The objective of this analysis is to predict the maximum fuel temperature in SVEA-96 Optima2 fuel rods both during normal plant operation and Anticipated Operational Occurrences (AOOs) and to compare those temperatures to the melting temperatures of the limiting fuel pellets.

Fuel pellet temperatures are calculated from BOL to EOL using a fuel performance code accepted for referencing in licensing applications by the NRC.

[

][a,c]

[

]^{a,c}

Sample Application

[
]^{a,c} typical of BWR/4, BWR/5, and BWR/6 plants. This fuel rod is designated "Style 1" in Sections 2 and 5. The rods are [
]^{a,c}

The STAV7.2 code described in Reference 1.2 was used for this evaluation.

[
]^{a,c}

The SPHs for the UO_2 TMOL and the UO_2-Gd_2O_3 TMOL were shown in Figures 4.3.0-1 through 4.3.0-12. The maximum fuel centerline temperature power history, including AOO transients, is shown in Figure 4.3.9-1. [
]^{a,c}

[

]^{a,c}

The total uncertainty due to the combination of these effects [

]^{a,c}

The results of the TMOL temperature calculations are shown in Tables 4.3.9-2 through 4.3.9-4 for the UO_2 rod designs (full length (FL) and part length PL (2/3 and 1/3 length), respectively), and are shown in Table 4.3.9-5 for UO_2-Gd_2O_3 rods. A summary is provided in Table 4.3.9-6.

Similarly, the results for the "limiting centerline temperature" rod which includes the AOO transients are shown in Table 4.3.9-7

The maximum pellet temperatures remain well below the melting temperature of the fuel, where the melting temperature of the fuel has been calculated from:

[

]^{a,c}

a,c

Table 4.3.9-1 Parameters and Values used for Fuel Temperature Uncertainties

Table 4.3.9-2 Maximum Fuel Temperature in FL-UO2 Rods

a,c

Table 4.3.9-3 Maximum Fuel Temperature in PL-2/3 UO2 Rods

a,c

Table 4.3.9-4 Maximum Fuel Temperature in PL-1/3 UO2 Rods

a,c

Table 4.3.9-5 Maximum Fuel Temperature in FL-(U, Gd)O2 Rods

a,c

Table 4.3.9-6 Summary of Maximum Pellet Centerline Temperatures

a,c

Table 4.3.9-7 Maximum Transient (AOO) Pellet Centerline Temperatures

a,c

a,c

Figure 4.3.9-1 Transient Power History (AOO) for Maximum Temperatures

4.3.10 Fuel Rod Bow

Methodology

The basic methodology identified in Reference 1.0 is unchanged.

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. [

]a,c

Evaluation of the potential for fuel rod bow in non-Westinghouse 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 Optima2 design to preclude fuel rod bow. Based on Westinghouse experience, as well as PWR and BWR industry experience, the following phenomena are believed to be the prime contributors to fuel rod bow:

[

]a,c

As discussed in Section 7, Westinghouse maintains a very aggressive post irradiation examination program. [

]a,c

[

]a,c

4.3.11 Pellet-Cladding Interaction

The basic methodology identified in Reference 1.0 is unchanged.

Methodology

As stated in the Standard Review Plan (Reference 1.4), 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. [

]^{a,c}

[

]a,c

Sample Application

[

]a,c

a,c

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

4.4 STEADY-STATE INITIALIZATION OF TRANSIENT AND ACCIDENT ANALYSES

The methodology for initializing various dynamic analyses with STAV6.2 results was provided in the response to RAI 2 for CENPD-287-P-A. The corresponding methodology for initializing these events with output from STAV7.2 is essentially the same as that described in CENPD-287-P-A. The description of this methodology is being incorporated into the body of WCAP-15942-P as Section 4.4 and clarified as judged necessary in the following text.

[

]^{a,c} and input for the evaluation of these models is provided from STAV7.2.

The methodology for the calculation of gap heat transfer coefficients and the treatment of different dynamic analyses are summarized in Sections 4.4.1 through 4.4.6.

4.4.1 Calculation of Gap Heat Transfer Coefficients

Under certain circumstances, the use of minimum or maximum gap heat transfer coefficients throughout the fuel rod lifetime can be shown to provide a conservative response. In these cases, fuel rod design characteristics, model parameters, and power histories can be selected to achieve minimum or maximum gap heat transfer coefficients which provide the desired level of conservatism in the parameter being calculated.

[

]^{a,c}

Nominal or bounding gap heat transfer coefficients are selected by utilizing nominal or conservative inputs to the STAV7.2 calculation of gap heat transfer coefficients. In either case, [

]^{a,c} An assembly type is defined as an assembly with a specific mechanical and nuclear design. [

]^{a,c}

[

]^{a,c}

Some dynamic analyses are performed on a cycle-specific basis. If it can not be confirmed that gap heat transfer coefficients established for the previous cycle(s) continue to be applicable for these analyses for the current cycle, the full process described above is utilized to calculate gap heat transfer coefficients for the current cycle being evaluated.

Sample Application

The example provided in the section on "Fast Transient Analysis" in the response to Question A2 in Appendix A of Reference 1.0 continues to be typical of the process to establish representative fuel rod power histories for different fuel types.

4.4.2 Fast Transient Analyses

The current Westinghouse fast transient analysis methodology used to evaluate Anticipated Operational Occurrences (AOOs) utilizes the BISON family of codes and methodology described in References 4.1 and 4.2. [

]^{a,c}

For the fast transient analyses, [

]^{a,c}

In addition, the fast transient analysis can be performed [

]a,c

Sample Application

As noted in Section 4.4.1, the example provided in the section on "Fast Transient Analysis" in the response to Question A2 in Appendix A of Reference 1.0 continues to be typical of the process to establish representative fuel rod power histories for different fuel types.

4.4.3 Control Rod Drop Accident (CRDA) Analysis

The control rod drop accident uses [

]a,c

Sample Application

[

]a,c

4.4.4 LOCA Analysis

The Westinghouse BWR Appendix K LOCA analysis methodology is described in References 3.3a-d. Gap heat transfer coefficients input to the LOCA calculations are based on STAV7.2 calculations.

Gap heat transfer coefficients supporting a conservative systems response [

]^{a,c}

Inputs to the STAV7.2 calculation which generates initial conditions for the [

]^{a,c}

Sample Application

As noted above, inputs to the systems response and time to dryout calculations are [

]^{a,c}

Figure 4.4-1 is a comparison of the [

]^{a,c}

[

]a,c

a,c

Using the conservative "LOCA" power history together with the conservative values cited in the above table led to the predicted fuel centerline temperature versus burnup shown in Figure 4.4-2. Also shown in Figure 4.4-2 is the fuel centerline temperature versus burnup for the same case run with nominal fuel geometry and model parameters using the "limiting centerline temperature" power history discussed in Section 4.3.0 and shown in Figure 4.4-1. [

]a,c

4.4.5 Stability Analysis

The nominal [

]a,c in accordance with the Safety Evaluation Reports (SERs) for References 4.9 and 4.10.

Sample Application

[

]a,c

4.4.6 Dose Calculations

The fission product inventory predicted by STAV7.2 [

]a,c incorporates the appropriate conservative assumptions outlined in Regulatory Guide 1.3 (Reference 4.11) as required by Section 15.6.5 of Reference 4.12. The calculation of doses due to a hypothetical Fuel Handling Accident incorporates the appropriate conservative assumptions outlined in Regulatory Guide 1.25 (Reference 4.13) as required by Section 15.7.4 of Reference 4.12.

a,c

Figure 4.4-1 Power Histories

a,c

[ ]

Figure 4.4-2 Pellet Centerline Temperatures

5 TECHNICAL DATA

This section augments Section 5 of Reference 1.0 to provide technical data for SVEA-96 Optima2.

The data in this table are typical for domestic BWRs. Some data, such as assembly and fuel rod length, can differ from plant to plant. For example, as with Section 5 of Reference 1, Style 1 provides typical data for BWR/5 and BWR/6 plants, while Style 2 is typical of a BWR/3 plant. Furthermore, some parameters can be cycle-specific. For example, bundle mass will change as the UO_2-Gd_2O_3 design changes.

All dimensions are at room temperature.

5.1 FUEL RODS

5.1.1 Pellets

5.1.1.1 Pellet Dimensions

UO_2 and Gadolinia Pellets

a,c

5.1.1.2 Pellet Data

a,c

5.1.1.3 Pellet Densification

[

]a,c

5.1.1.4 Burnable Poison Pellet

Westinghouse utilizes gadolinia (Gd_2O_3) as a burnable poison. The pellets are a mixture of Gd_2O_3 and UO_2 [

]^{a,c}

5.1.2 Fuel Rod Cladding

5.1.2.1 Cladding Dimensions

a,c

5.1.2.2 Cladding Chemical and Physical Properties

[

]^{a,c}

5.1.3 Fuel Rod Length

a,c

5.1.4 Fuel Rod Miscellaneous Data

a,c

5.1.5 Fuel Rod Materials

a,c

5.1.6 Typical Fuel Rod Weights

a,c

5.1.7 Spacer Grid

a,c

5.1.8 External Spring

a,c

5.2 FUEL ASSEMBLY DATA

5.2.1 Fuel Assembly Miscellaneous Data

a,c

5.2.2 Fuel Assembly Materials

a,c

5.2.3 Typical Fuel Assembly Weights

a,c

6 CODE DESCRIPTION

This section contains a brief description of the computer codes used by Westinghouse in mechanical design calculations. More detailed descriptions of the fuel rod design codes are contained in Reference 1.2.

The VIK-3 code is a collection of the models and formulae which can be used to calculate clad stresses as a function of burnup from BOL to EOL. STAV7.2 is the principal code for fuel performance analysis. COLLAPS II is used to calculate cladding ovality as a function of irradiation.

Westinghouse utilizes the finite element code ANSYS for stress analysis of the SVEA-96 Optima2 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-3

The computer code VIK-3 calculates stresses in light water reactor (LWR) fuel rod cladding as a function of fuel burnup or irradiation time. Both fully recrystallized and cold work stress-relieved Zircaloy cladding can be evaluated. The VIK-3 models utilize the same standard engineering mechanics formulae described in VIK-2 in Reference 4.2. VIK-3 has been equipped with the option to allow execution in conjunction with STAV to provide cladding stress evaluations as a function fuel rod burnup based on materials properties and STAV7.2 calculated parameters.

The code consists of a number of subroutines, each calculating the stress due to the different sources or load cases. The stresses are calculated at the clad inner and outer radii at three axial locations, namely at a spacer, between spacers and at the bottom end plug. Depending on the origin of the stress and on geometrical and material discontinuities in the design, each stress is classified with the appropriate stress category. The effective stresses are calculated using the Tresca relationship in accordance with Section III of the ASME Code.

Changes relative to the VIK-2 code described in Reference 4.2, which have been incorporated in VIK-3 can be summarized as follows:

1. Stress calculations can be performed as a function of fuel rod burnup using STAV7.2 materials properties, fuel rod parameter inputs, and loads.

2. The process to obtain the maximum effective stresses based on different load combinations has been improved relative to the process in VIK-2 to support burnup-dependent calculations.

Details of these changes relative to the VIK-2 code described in Reference 4.2 are presented in Section 4 of Reference 1.2. 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. [

]^{a,c}

Pressure at End Plug

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

]^{a,c}

Ovality

The initial ellipticity of the cladding under uniform external pressure gives rise to tangential and axial stresses in the fuel cladding. [

]^{a,c} 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. [

]^{a,c}

Azimuthal Temperature

VIK-3 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, [ ]^{a,c}.

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. [ ]^{a,c}

6.2 STAV7.2

The STAV7.2 code is used by Westinghouse for BWR and PWR fuel rod performance analyses. This report addresses the application of STAV7.2 in the United States for BWR applications only. STAV7.2 is used in Europe for both BWR and PWR applications. STAV7.2 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.

STAV7.2 calculates the variation with time of all significant fuel rod performance quantities including fuel and cladding temperatures, fuel densification, fuel swelling, fission product gas release, rod internal pressure, and pellet-cladding gap conductance. Stresses and strains in the cladding due to elastic, thermal, creep and plastic deformations are calculated. Also, cladding oxidation is evaluated and included in the evaluation of fuel rod performance parameters. Other sub-models include burnup-dependent radial power distributions for both UO_2 and (U, Gd)O_2 fuel, fuel grain growth, and helium release.

The STAV7.2 fuel rod performance code is an improved version of the STAV6.2 code described in Reference 4.2. Some of the models in STAV6.2 have been updated, and new models have been introduced in the STAV7.2 code to obtain improved predictions of various fuel properties throughout the design life of the fuel rod to extend the fuel rod burnups beyond those addressed in Reference 4.2. These changes can be summarized as follows:

1. A model describing burnup-induced degradation of fuel pellet conductivity has been introduced.

2. The pellet radial power distribution model has been improved to more accurately take into account power generation by plutonium isotopes and treat the pellet rim region.

3. A new pellet cladding mechanical interaction (PCMI) model including friction and axial segment interaction and associated rod elongation has been introduced.

4. A new athermal fission gas release model has been introduced which is based on an expanded data base utilizing a rim region burnup calculation for modeling the enhanced athermal release at high burnup.

5. An improved thermal fission gas release model based on an expanded data base has been included. Enhancements to the thermal fission gas release model are:

    a. An improved steady-state gas diffusion coefficient treatment.

    b. Grain boundary fission gas sweeping due to grain growth has been introduced in the fission gas release model.

    c. A transient fission gas release model utilizing modifications to the diffusion coefficient and the grain boundary saturation during a ramp have been implemented.

6. A new creep model for fully-annealed cladding has been introduced.

7. Revised boiling water reactor crud build-up, clad corrosion, and hydriding models have been incorporated.

8. The gap heat transfer coefficient model has been improved.

9. Pellet relocation model has been modified to improve temperature predictions.

Details of the STAV7.2 code description are presented in Section 2 of Reference 1.2. 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 STAV7.2 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, 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 it can be provided directly as input. Fast neutron flux can be provided at each burnup step and is based on neutronic code power distribution calculations.

Thermal Hydraulic Parameters

The input subchannel geometry (pitch), coolant inlet temperature, coolant pressure, and coolant mass flow rate are supplied as input to the code for calculating the cladding outer surface temperature.

The heat transfer between the cladding and the coolant is modeled with either single-phase convection, subcooled boiling, or saturated flow boiling. For the boiling water reactor application, the Jens-Lottes correlation for nucleate boiling is used. [
]^{a,c}

Fuel Rod Parameters

The STAV7.2 computational path starts from the coolant with thermal and hydraulic calculations and extends to cladding strain-stress calculations and fission gas release calculations. Cladding corrosion (oxide layer thickness) is calculated as a function of irradiation time. The effect of Zircaloy cladding wall thinning due to oxide layer growth is taken into account in both the thermal and stress analysis.

STAV7.2 uses Reymann type formulae for fuel pellet thermal conductivity that includes both phonon conduction and electronic conduction. The phonon conduction term is both temperature and burnup dependent. The conductivity reduction for fuel rods containing gadolinia is also modeled. The improved fuel conductivity model is discussed in Reference 1.2.

The fuel pellet density model in STAV7.2 includes contributions from several components including densification, swelling accommodation, and solid fission product swelling.

STAV7.2 models the pellet-cladding mechanical interaction (PCMI). The cladding is displaced by a rigid pellet and is subjected to creep, and for sufficiently large stresses, plastic deformation. Axial and radial PCMI are accounted for in the code. Friction between the pellet and cladding, as well as the interaction between the axial segments of the rod, are modeled. The Zircaloy cladding creep model in STAV7.2 consists of a thermal term and an athermal term. Both primary creep and secondary creep behavior are considered. The pellet-cladding gap and the pellet-cladding contact pressure are treated interactively with the pellet cladding mechanical interaction model.

The fission product gas release (FGR) model consists of an athermal (low-temperature) and a thermal (high-temperature) release component. The athermal FGR model accounts for release contributions from the fission product knock-out processes occurring at regions near the pellet periphery surface and at pellet internal crack surfaces. Athermal release is only a function of burnup. In STAV7.2, this model accounts for the enhanced release observed for burnups above about 40 MWd/kgU. For these high burnups, a non-linear term in the athermal FGR model becomes dominant over a linear contribution. In this model,

the release is considered to be a result of the formation of a porous rim at high burnups, which in turn is related to the radial power and burnup distribution in the pellet. A fine geometric mesh to capture pellet power distribution variation and to track appropriate isotopes has been included in the radial power profile model in STAV7.2.

The model for high temperature FGR is [

]^{a,c}

6.3 COLLAPS-II VERSION 3.3D

The computer code COLLAPS-3.3D is used for prediction of cladding ovality in BWR fuel rods as a function of irradiation time.

The COLLAPS-3.3D 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-3.3D code.

COLLAPS-3.3D 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.

Changes relative to the COLLAPS-3.2S code described in Reference 4.2 incorporated into COLLAPS-3.3D can be summarized as follows:

1. The COLLAPS-3.2S code described the cladding as an infinitely long cylinder with no internal support. The COLLAPS-3.3D code includes the option to describe the cladding as a cylinder with a finite gap between supports. This option corresponds to allowing a finite axial gap between two axially adjacent pellets in a fuel rod. The finite length model is benchmarked against the CEPAN finite length model and measured data.

2. The new creep model for fully-annealed cladding incorporated in STAV7.2 has been incorporated into COLLAPS-3.3D.

Details of the COLLAPS-3.3D code description are presented in Section 6 of Reference 1.2.

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 Westinghouse BWR fuel designs is shown in Figure 7-1. Westinghouse 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.

Westinghouse started manufacturing and delivering 8x8 BWR fuel in 1967. First cores and reload quantities of 8x8 fuel have been delivered to all eleven Westinghouse-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 Westinghouse 8x8 fuel has been excellent. The last 8x8 fuel was manufactured in 1987.

The second generation of Westinghouse 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 Westinghouse 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-Westinghouse 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 in the U.S. by Westinghouse as the QUAD+ assembly.

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

]^{a,c}

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. [

]^{a,c}

As described in Section 2.1, this generation also includes the SVEA-96+ design, [
]^{a,c}.

The fourth evolutionary generation involves the introduction of part length rods, and includes the SVEA-96 Optima and SVEA-96 Optima2 designs. The SVEA-96 Optima design contains [

]^{a,c}

The SVEA-96 Optima2 design has a total of [

]^{a,c}

Since many of the basic mechanical design features of the SVEA design have not been changed, the experience gained on earlier designs is also applicable to SVEA-96 Optima2 design.

The experience base is steadily increasing and as of August 2004, 119 reloads of Westinghouse 10x10 SVEA fuel have been delivered to [ ]^{a,c}. As of September 2004, Westinghouse has contracted for the delivery of [ ]^{a,c}.

7.2 EXPERIENCE

A complete summary of Westinghouse fuel assembly burnup experience is shown in Figure 7-2.

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 peak burnup of [ ]^{a,c}, and the other two in 1988, also with a peak burnup of 35 MWd/kgU after their seventh cycle. In Oskarshamn 2, one SVEA-64 assembly reached [ ]^{a,c} and another SVEA-64 assembly reached [ ]^{a,c}. 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. In total 6174 SVEA-64 assemblies have been delivered, with the last assemblies loaded in Oskarshamn 1 in 2000.

7.2.2 SVEA 10x10 fuel

A summary of all SVEA 10x10 fuel deliveries is shown in Table 7-1.

7.2.2.1 SVEA-100

The first SVEA-100 Lead Fuel Assemblies were loaded in 1986: four into the Oskarshamn 3 and two into the Forsmark 3 reactors. In 1990 the first full SVEA-100 reload consisting of 100 assemblies, was loaded into Oskarshamn-3. Since then SVEA-100 assemblies have been loaded into five Swedish and one Finnish reactor on an annual schedule.

More than [ ]^{a,c}. Several of these assemblies have reached an average of [ ]^{a,c}.

7.2.2.2 SVEA-96/SVEA-96+

The initial eight SVEA-96 Lead Fuel Assemblies (LFAs) were loaded into Forsmark 3 in 1988. Since this first delivery, SVEA-96/SVEA-96+ fuel have been delivered to seven Westinghouse and four Siemens built reactors.

In 1990, 116 SVEA-96 fuel assemblies were delivered to the Swiss Leibstadt reactor, which is a General Electric BWR/6 plant. The same fuel design has also been delivered to the Spanish BWR/6, Cofrentes. Twelve reloads have currently been delivered to these two European GE reactors.

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 loaded in Columbia Generating Station in 1990, to be followed by 4 LFAs in Fermi2, Peach Bottom 2 and Limerick 2 (all BWR/4s) the following year. In addition, Susquehanna received four SVEA-96+ LFAs of in 1996.

Reload quantities of SVEA-96 fuel were delivered to the Columbia Generating Station for five consecutive cycles during the period from 1996 to 2001, and three reloads of SVEA-96+ fuel were loaded in the Hope Creek Generating Station in the period from 1999 to 2003.

More than [

]a,c

7.2.2.3 SVEA-96 Optima/SVEA-96 Optima2

SVEA-96 Optima LFAs were inserted into [

]a,c

As of August 2004, [

]a,c

The SVEA-96 Optima2 design [

]a,c

7.3 FUEL RELIABILITY

7.3.1 General

Primary fuel leakers in SVEA 10x10 fuel is shown in Figure 7-3. With the exception of the Enhanced Space Shadow Corrosion event which occurred in the Leibstadt plant in1997, debris fretting has been identified as the cause of the primary failure in all fuel inspected. To mitigate this failure mode,

Westinghouse developed the TripleWave debris filter described in Section 2.3.1. This improvement is expected to significantly reduce the probability of harmful debris reaching the fuel rods.

Note that the data in Figure 7-3 are based on failed fuel rods, not assemblies. It is Westinghouse 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 hot cells for further investigation. The majority of the remaining unidentified cases are believed to be debris failures.

7.3.2 8x8

Fuel performance for Westinghouse 8x8 fuel has been good with the majority [

]a,c

7.3.3 SVEA-64

Fuel performance following the introduction of Westinghouse SVEA-64 fuel has been excellent in an environment which included plant power uprating and initiation of extended operating flexibility including extended flow windows in most of the Nordic plants. The primary cause of SVEA-64 primary leakers has been debris-related. Fuel reliability per cycle for Westinghouse 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 the four dryout-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.

Four of the SVEA-64 failures were caused by the Dryout Event [

]a,c

7.3.4 SVEA 10x10 fuel

One of the driving forces behind Westinghouse's choice of the 10x10 array was increased fuel reliability via a substantial reduction in fuel rod duty. The impact of the 10x10 design on fuel reliability can be summarized as follows:

- Investigations into fuel rod corrosion have found that corrosion rates are strongly heat flux dependent (greater the heat flux increases the corrosion rate up on the rods, thus degrading fuel rod performance).

- The total heat transfer surface area of a 10x10 array is 25 percent greater than in an 8x8 array. Therefore, the heat flux for a 10x10 assembly will be about 75 percent of that for an 8x8 assembly. Computer simulations indicate a 40 percent reduction in corrosion buildup associated 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 lower fission gas release for the same assembly burnup. This allows higher discharge burnups (hence better economics) or inherently more margin to potential failure mechanisms when compared with 8x8 fuel.

- However a lot of these extra margins have been utilized to improve thermal performance to increase maximum assembly power. Also significant reactor power up-rates have been possible as an increased maximum power also have allowed an increased average assembly power.

Westinghouse 10x10 fuel performance for fuel delivered during the period 1986 through 2004 has been excellent, with [

]^{a,c}

Westinghouse has extensive experience with Sn-alloyed Zirconium liner beginning with [

]^{a,c}

The Westinghouse 10x10 fuel experience with secondary degradation is summarized in Figure 7-4. About [

]^{a,c}

[

]^{a,c}

7.3.5 Reliability Improvement

In the interest of pursuing the goal of failure-free fuel, improvements to both avoid primary failures as well as secondary degradation should a failure occur are being introduced on a continuing basis. [

]^{a,c}

7.4 INSPECTIONS

7.4.1 SVEA-64

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

Over [

]^{a,c}. 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.

[

]a,c

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

7.4.2 SVEA 10x10 Fuel

From a [

]a,c Therefore, the experience gained from operation of SVEA-64 supports that for the SVEA 10x10 designs.

Fuel inspections have been carried out on the lead 10x10 fuel [ ]a,c. These inspections have shown that the fuel assemblies are in good general condition with the expected mechanical performance as well as cladding corrosion levels.

The first high burnup SVEA-100 was inspected in August 1993 in Forsmark 3. [ ]a,c

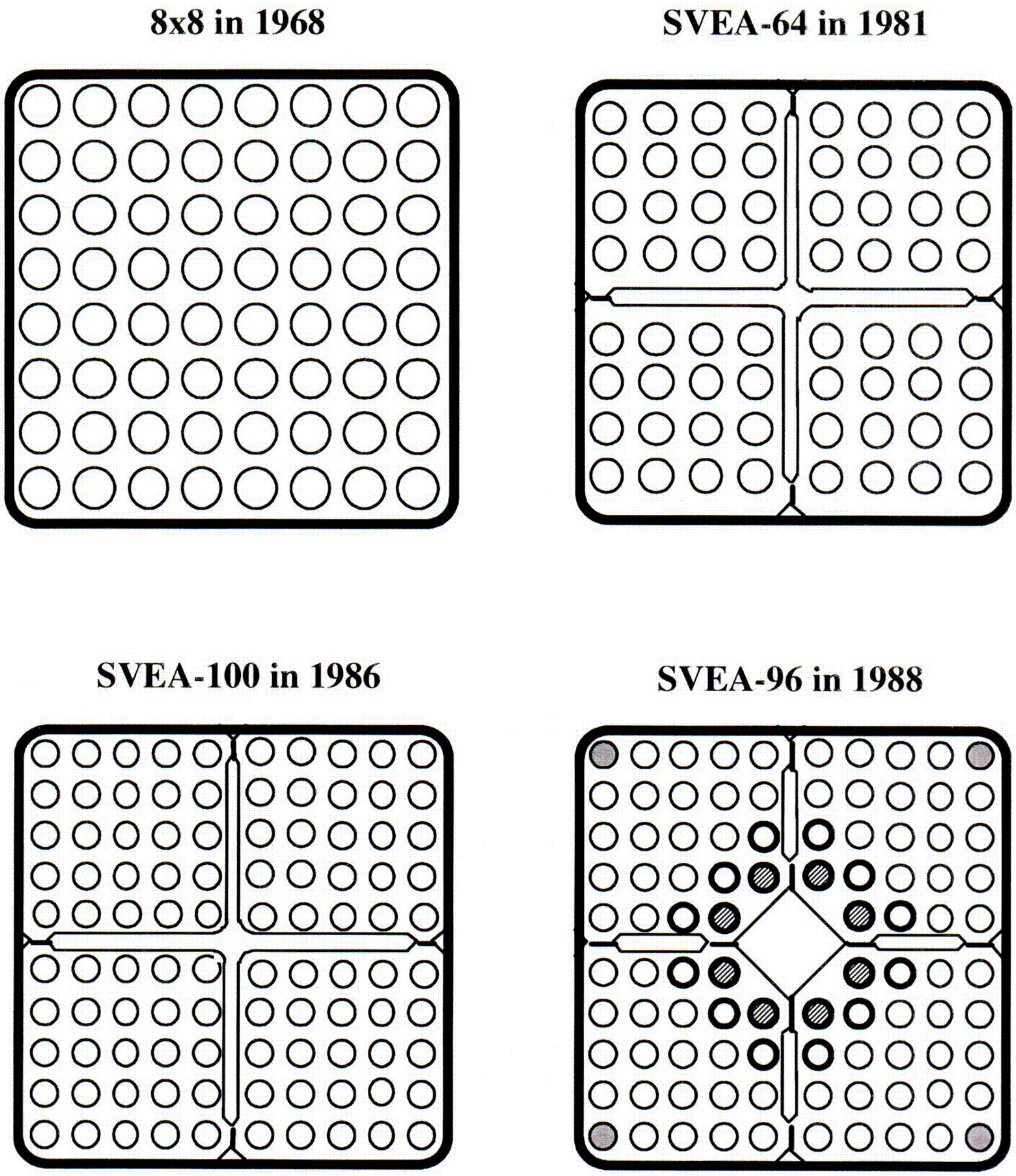
As of the summer of 2004, SVEA-96 assemblies in [

]a,c

SVEA-96 Optima LFAs have been inspected [

]a,c

Table 7-1 SVEA 10x10 Fuel Deliveries				
Plant	**SVEA-96/100**	**Optima**	**Optima2**	**Total number**
Barsebäck 1	288	4		292
Barsebäck 2	478	4	72	554
Oskarshamn 1	110	152		262
Oskarshamn 3	1438	264	384	2086
Ringhals 1	304			304
Forsmark 2	1270		6	1276
Forsmark 3	1222			1222
Olkiluoto 2	468	346		814
Brunsbüttel	480			480
Philippsburg 1	252	4	92	348
Isar 1	952		308	1120
Leibstadt	1056	272	380	1708
Cofrentes	392		136	528
CGS	912			912
Limerick 2	4			4
Susquehanna	4			4
Hope Creek	720			720
Fermi 2	4			4
Peach Bottom 2	4			4
Krümmel	476			476
Total	**10834**	**1046**	**1478**	**13258**



Note: The SVEA-96 design showed above contains 96 full-length rods with the same diameter in SVEA-96 and SVEA-96+. The part-length rods for SVEA-96 OPTIMA and OPITMA2 are identified as follows.

2/3 Part Length Rods in SVEA-96 Optima and Optima2

1/3 Part Length Rods in SVEA-96 Optima2

Increased diameter (10.3 mm) in SVEA-96 Optima

Figure 7-1 SVEA Fuel Designs

C01

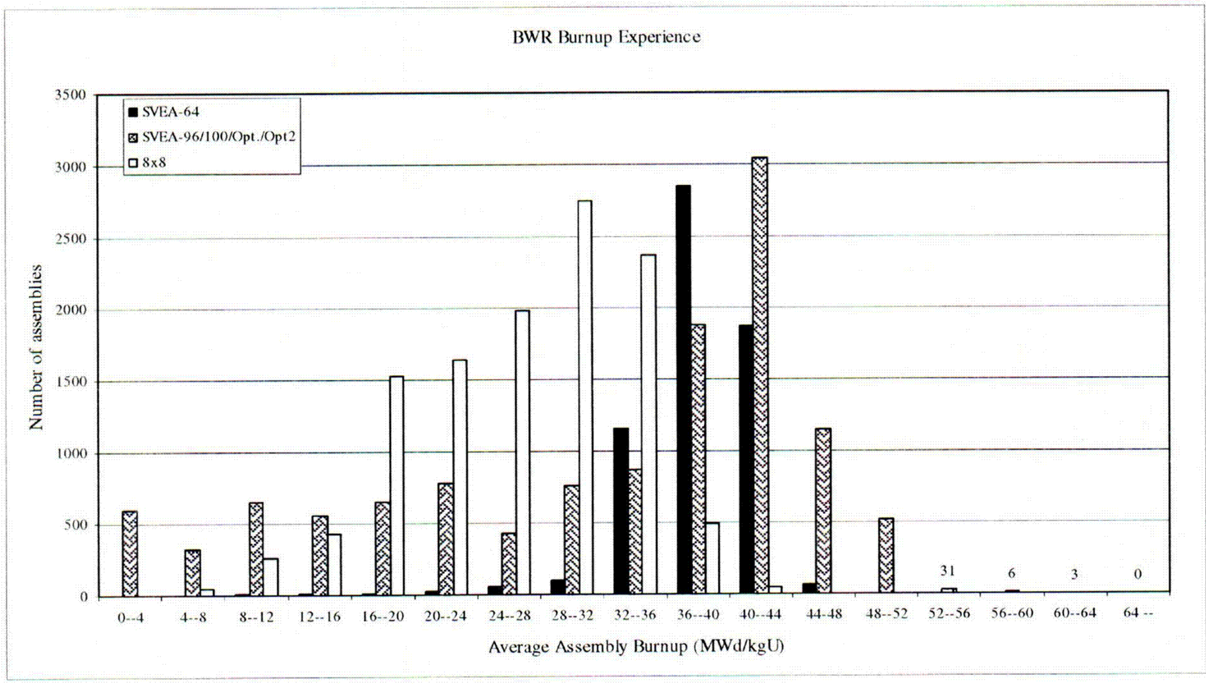


Figure 7-2 Burnup Statistics as of August 2004

C02

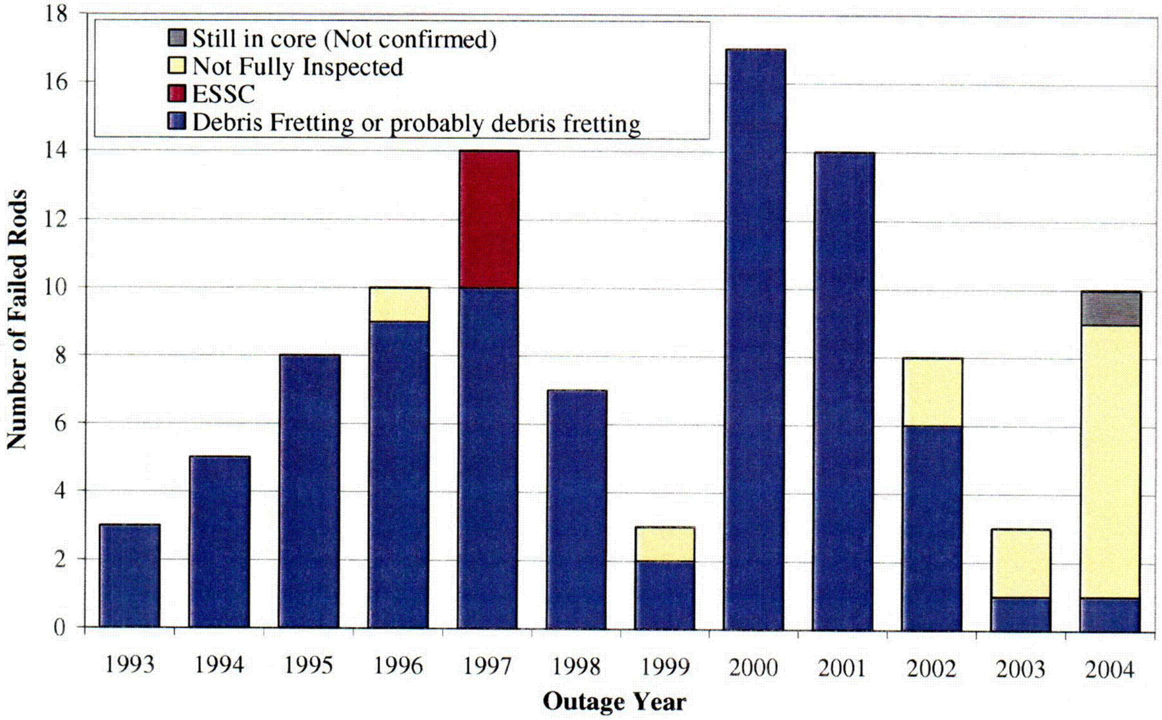


Figure 7-3 Primary Failure Experience in SVEA 10x10 Fuel

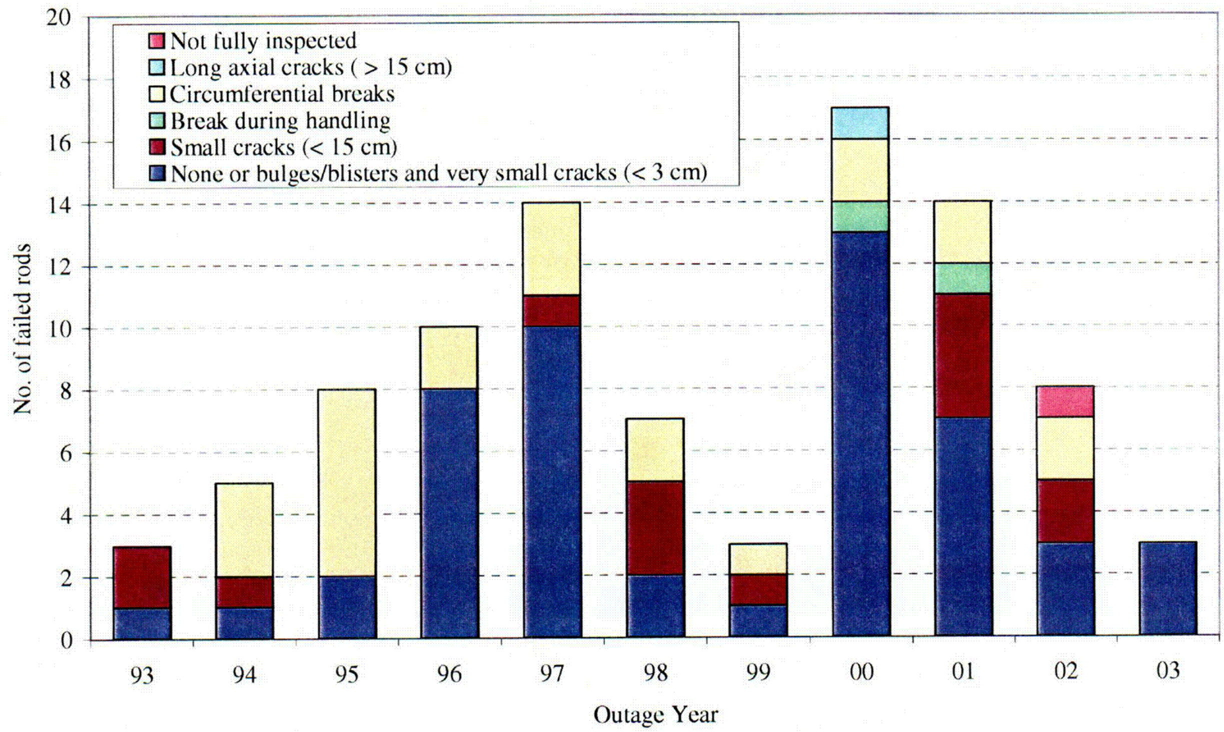


Figure 7-4 Secondary Degradation Experience in SVEA 10x10 Fuel

C04

8 PROTOTYPE TESTING

This section has been updated relative the corresponding section in Reference 1.0 to reflect prototype testing since Reference 1.0 was completed.

Westinghouse has a continuing program to perform prototype testing for all of their fuel assembly designs. Tests have been performed on the Westinghouse 8x8 assembly, the SVEA-64 design, the SVEA-96/96+ design, the SVEA-100 design, the SVEA-96 Optima design and the SVEA-96 Optima2 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 Optima2 design and design evaluation.

This information is provided to supplement the analytical and operating experience bases of the 10x10 SVEA fuel, including the SVEA-96 Optima2 design. A discussion of in-reactor experience, which includes inspection data from Lead Fuel Assemblies at various plants in addition to reload quantities of 10x10 SVEA fuel is provided in Section 7.

8.1 FRETTING TESTS

Full-scale tests using one- and two-phase flow have been carried out on SVEA-96 Optima2 test fuel assemblies in the Westinghouse BURE test loop in Västerås, Sweden. The intent of these tests was to verify that unacceptable fretting wear would not occur under operating conditions. The spacer springs were adjusted to [

]a,c. Conditions for the tests are described in the table below.

Nominal SVEA-96 Optima2 Test, Operating Parameters			
		2-Phase Test	**1-Phase Test**
Temperature	°C	280	270
Pressure	bar	69	72
Total flow	kg/s	19	20
Steam quality	%	15	-
Test time	hour	700	700

The test assemblies and all their components were carefully inspected after the tests. [

]a,c

The results of the inspections revealed that the function of the structural members had been as expected.
[

]a,c

The overall conclusion from the fretting tests are that the mechanical behavior of the SVEA-96 Optima2 fuel is satisfactory and that reactor operation without unacceptable wear for the design life of the fuel caused by fretting can be expected.

8.2 PRESSURE CYCLING TEST

A pressure cyclic test was performed [ ]^{a,c} channel to verify its ability to withstand daily load following during reactor operation. Since the [ ]^{a,c}.

The test was performed [ ]^{a,c}

8.3 LATERAL LOAD CYCLING TEST, CHANNEL AND SPACER GRID

Lateral load cycling tests have been performed with low-cycle fatigue tests with the purpose of qualifying spacers and channel welds for seismic loads. During a seismic event, dynamic forces from the sub-bundle are transmitted by the spacers to the water cross and the outer channel.

The test was performed by [ ]^{a,c} channel. However, since the channel wall [ ]^{a,c}

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 effects of the temperature and irradiation as well as experimental uncertainty.

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:

[

]a,c

8.4 SPACER CAPTURE ROD TEST

The basic SVEA-96 spacer capture tab design and spacer capture tab weld process and qualification process [

]a,c

As discussed in Reference 1.0, a SVEA-96 test was performed to determine the spacer capture force for grid passage. [

]a,c

8.5 HANDLE TENSION TEST

[

]a,c

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. [

]a,c

Furthermore, additional tension tests have been performed on SVEA-96 Optima2 handles. The minimum measured margins against handle rupture in these tests were [

]^{a,c}.

8.6 TENSION TEST ON SCREW MOUNTED IN CHANNEL

[

]^{a,c}

9 TESTING, INSPECTION, AND SURVEILLANCE PLANS

9.1 TESTING AND INSPECTION OF NEW FUEL

Westinghouse Electric Company (headquartered in Pittsburgh, Pennsylvania, U.S.A.) has operations located throughout the world that are responsive to energy industry, utilities, and government needs. Westinghouse operations are made up of organizations that are responsible for specific business areas. These operational organizations are responsible for marketing, design, procurement, manufacture, installation, inspection, testing, servicing, project management, and operation of certain nuclear power plant items, radioactive material packaging and transportation, and non-nuclear items. Westinghouse also offers engineering services such as life-extension studies, diagnostics, service analyses, and item and service testing.

Primary Westinghouse BWR fuel manufacturing facilities are currently located in Västerås, Sweden and Columbia, South Carolina. BWR fuel pellets and assemblies are currently manufactured in either Columbia or Västerås.

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

[

]a,c

[

]a,c

Fuel Subbundles

[

]a,c

Fuel Channel

[

]a,c

Handle

[

]a,c

Fuel Assembly

[

]a,c

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, Westinghouse considers inspection of Westinghouse 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.

[

]a,c

[

]^{a,c}

The data from these examinations, plus historical records are collected, summarized, documented, stored and readily retrievable by Westinghouse in Europe and the U.S. The information is made available to fuel users. Lessons learned are fed back into the design to improve the fuel performance, decrease the risk, and to reduce cost. Westinghouse has performed fuel surveillance on irradiated SVEA-10x10 fuel in Swedish reactors during outages every year since 1987. The experience with SVEA-10x10 fuel is directly applicable to SVEA-96 Optima2. Furthermore, Westinghouse has performed examinations of various SVEA-96 fuel types in Westinghouse 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.

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

Surveillance work may include any or all of the following:

[

]^{a,c}

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

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1.2 "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1," Westinghouse Report WCAP-15836-P (proprietary), WCAP-15836-NP (non-proprietary), June 2002 and RAI responses to this report dated July 2004. (WCAP-15836-P is a supplement to CENPD-285-P-A. As such, it does not repeat the information in CENPD-285-P-A but refers the reader to the appropriate sections in CENPD-285-P-A.)

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