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Reference Fuel Design SVEA-96 Optima3

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Acronym	Definition
AISI	American Iron and Steel Institute
AMS	Aerospace Material Specification
AOO	Anticipated Operational Occurrence
APLHGR	Average Planar Linear Heat Generation Rate
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BA	Burnable Absorber
BE	Best Estimate
BOC	Beginning of Cycle
BOL	Beginning of Life
BWR	Boiling Water Reactor
CCFL	Counter-Current Flow Limit
CF	Continuous anneal Furnace
CHF	Critical Heat Flux
CILC	Crud Induced Localized Corrosion
CPR	Critical Power Ratio
CRDA	Control Rod Drop Accident
DM	Design Multiplier
ECCS	Emergency Core Cooling System
EFPH	Effective Full Power Hours
EOC	End of Cycle
EOL	End of Life
FE	Finite Element
FEA	Finite Element Analysis
FEM	Finite Element Methodology
FGR	Fission Gas Release
FLR	Full-length Rod
FSAR	Final Safety Analysis Report
GDC	General Design Criteria

ID	Inner Diameter
LHGR	Linear Heat Generation Rate
LTR	Licensing Topical Report
LOCA	Loss-of-Coolant Accident
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
MF	Maneuvering Factor
NCLO	No Clad Lift Off
NFIR-V	Nuclear Fuels Industry Research phase V
NRC	Nuclear Regulatory Commission
OD	Outer Diameter
OLMCPR	Operating Limit Minimum Critical Power Ratio
PCI	Pellet-Cladding Interaction
PIE	Post-Irradiation Examination
PLR	Part-Length Rod
PPM	Parts-per-million
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RIP	Rod Internal Pressure
RMS	Root Mean Square
SAFDL	Specified Acceptable Fuel Design Limit
SPH	Segment Power History
SRP	Standard Review Plan
TD	Theoretical Density
TIG	Tungsten Inert Gas
TMOL	Thermal Mechanical Operating Limit
Zry-2	Zircaloy-2
Zry-4	Zircaloy-4
3-D	3-dimensional

1 SUMMARY AND CONCLUSIONS

1.1 SUMMARY

WCAP-15942-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors Supplement 1 to CENP-287" (Reference 1.0) and WCAP-15942-P Supplement 1, "Material Changes for SVEA-96 Optima2 Fuel Assemblies" Reference 4.3 describe the Westinghouse methodology for conducting fuel assembly and fuel rod mechanical evaluations that are identified in Section 4.2 of the Standard Review Plan (SRP), NUREG-0800 (Reference 1.4) and WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1" (Reference 1.2) describes the fuel rod design methods. This Licensing Topical Report (LTR) describes the application of these methods and methodologies to the Westinghouse SVEA-96 Optima3 fuel assembly and it also includes some minor improvements to the methodology.

In conjunction with an expanded fuel rod and assembly inspection database and test basis, this sample application demonstrates that the SVEA-96 Optima3 assembly satisfies the Westinghouse design criteria to a rod-average burnup of [ ][a,c] for the sample plant application. As discussed in Reference 1.0 and Reference 4.3, 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 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants" (Reference 1.5). Similar information, supporting the thermal-hydraulic, nuclear, and safety analyses evaluations, is provided in CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel" (Reference 1.1).

The SVEA-96 Optima3 fuel assembly contains, as does its predecessor SVEA-96 Optima2, part-length rods in addition to full-length rods.

Design criteria and methods in References 1.0, 1.1 and 4.3 have not been changed and will continue to be used for the reference fuel design for SVEA-96 Optima3. Reference 4.3 is currently under NRC review. Any SER required changes will be implemented for SVEA-96 Optima3 with **Low Tin ZIRLO™** channels. Therefore, this LTR relies on References 1.0, 1.1 and 4.3 for design criteria and methodology. In Sections 2 through 10 of this report, in order to clearly discriminate text already presented in References 1.0, 1.1, 4.3 regarding such subjects as design criteria, methodology, and sample analyses, these texts are given in italics.

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

The contents of this report can be summarized as follows:

1. A description of the Westinghouse SVEA-96 Optima3 boiling water reactor (BWR) fuel assembly design,

2. A modified stress analysis using the ANSYS code,

3. A sample application of the Westinghouse design evaluation methodology demonstrating compliance of the SVEA-96 Optima3 assembly with the design criteria for normal operations and anticipated operational occurrences (AOOs) to a fuel rod burnup of [ ]^{a,c},

4. An applicability demonstration of the Westinghouse LOCA methodology for SVEA-96 Optima3,

5. A summary of the computer codes used in the Westinghouse methodology described in References 1.0 and 1.1,

6. A description of the manufacturing inspection measures which assure that the assembly is constructed as required by the design specifications described in References 1.0 and 1.1,

7. A summary of the operating experience with the SVEA-96 Optima3 design and similar Westinghouse designs,

8. A summary of the ex-core prototype test programs relative to the methodology described in References 1.0 and 1.1,

9. An updated summary discussion of ongoing testing, inspection, and surveillance plans relative to the methodology described in References 1.0 and 1.1.

Therefore, in conjunction with References 1.0 and 1.1, 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 Optima3 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 Reference 1.4.

1.2 CONCLUSION

The information contained in this report in conjunction with References 1.0, 1.1, and 4.3 supports the following conclusions:

1. The design bases identified in Reference 1.0 are sufficient to assure that the requirements and guidelines identified in Section 4.2 of NUREG-0800 (Reference 1.4), 10 CFR 50, Appendix A (Reference 1.5), and Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, "Rules for Construction of Nuclear Facility Components" (Reference 1.3) will be satisfied.

2. The methodology for evaluating fuel assembly and fuel rod mechanical behavior relative to the design basis remains acceptable for design and licensing applications to a rod-average burnup of [ ]^{a,c} (References 1.0, 1.1, and 4.3).

3. The evaluation of the SVEA-96 Optima3 fuel assembly applying Westinghouse methodology demonstrates the capability of this fuel assembly to satisfy the fuel performance, mechanical, and

materials design bases under normal operation and AOOs to a peak rod-average burnup of
[ ]a,c.

4. Westinghouse LOCA methodology is applicable for the SVEA-96 Optima3 fuel design.

2 GENERAL DESCRIPTION

The SVEA-96 Optima3 fuel with its 4x(5x5-1) lattice with three part-length rods in each subbundle is an evolution of the SVEA-96 Optima2 fuel and a new generation of Westinghouse 10x10 fuel.

The SVEA-96 Optima3 fuel has, compared to SVEA-96 Optima2:

- Simplified mechanical design
- Increased fuel rod plenum volume
- Reduced pressure drop
- Reduced parasitic neutron absorption
- Improved fuel reliability

These improvements have been achieved by:

- The subbundle top tie plate is replaced with a top spacer and the bottom tie plate is simplified. All fuel rods, except the tie fuel rods, which also have the spacer capture function, rest freely on the bottom tie plate,

- The modified design in the subbundle ends allows shorter end plugs and correspondingly longer cladding tube, where the starting point of the pellet stack is lowered by [ ]^{a,c},

- A new spacer design (sleeve type), where a simplified version without mixing vanes is used at [ ]^{a,c}. This, in combination with the simplified bottom tie plate, leads to reduced pressure drop,

- The new spacer design leads to a reduced amount of parasitic neutron absorption material in the active fuel region,

- The spacer is designed to minimize the risk of debris getting caught in the spacer and thus reduces the risk of debris fretting damage to the fuel,

- The **TripleWave+™** debris filter is introduced with improved debris catching efficiency.

2.1 ASSEMBLY DESCRIPTION

The SVEA-96 Optima3 fuel is based on the same general principles as SVEA-96 Optima2 and previous SVEA-96 fuel types which have been delivered to several plants in the U.S. including Columbia Generating Station, Hope Creek, Dresden 2&3, Quad Cities 1&2, and Hatch-1. This section contains a general description of the SVEA-96 Optima3 fuel and a discussion of the main differences between SVEA-96 Optima3 and SVEA-96 Optima2 (Reference 1.0).

As with previous SVEA-96 fuel designs, the SVEA-96 Optima3 fuel assembly consists of one fuel bundle, arranged in four subbundles, one fuel channel and one handle with spring. The subbundles are separated by a cruciform internal structure (water cross) in the channel. The subbundles are inserted into the channel from the top and are supported at the bottom end by a stainless steel inlet piece (transition piece and bottom support), which is bolted to the channel. This design principle eliminates any leakage flow uncertainties at the bottom end of the channel. Since the handling load is carried by the channel via the bottom support and the four screws, stresses in the tie rods are also avoided during normal fuel handling. The bottom support is equipped with a **TripleWave+**™ debris filter, which is designed to prevent potentially harmful debris from entering the fuel bundle. The fuel assembly is lifted by the handle, which is connected to the top end of the channel, and supported against adjacent assemblies in the core module by a double leaf spring. A schematic overview of the SVEA-96 Optima3 assembly with its main components is shown in Figure 2-1a and the fuel assembly design is shown in Figure 2-1b.

As with previous SVEA-96 fuel designs, the fuel bundle consists of 96 fuel rods arranged in four 5x5-1 subbundles. The outer channel forms, together with the water cross structure, four subchannels for the subbundles. 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 Optima2, there are eight part-length fuel rods with about 2/3rd active length (two in each subbundle) and four part-length fuel rods with about 1/3rd active length (one in each subbundle) in the fuel bundle. Consequently, the lower part of the fuel bundle (zone 1) consists of 96 fuel rods, the middle part (zone 2) consists of 92 fuel rods and the upper part (zone 3) consists of 84 fuel rods. The cross section of the fuel assembly is shown in Figure 2-2.

The fuel assembly outer dimensions are maintained from previous SVEA-96 fuels, including SVEA-96 Optima2. 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 Optima3 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 pads/rollers. The SVEA-96 Optima3 assemblies also provide adequate clearances to instrument guide tubes. The excellent 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 previous SVEA-96 fuel designs, the SVEA-96 Optima3 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 2.5.

The number of spacers is increased and the SVEA-96 Optima3 fuel is equipped with 40 (10x4) sleeve type spacers. [

]^{a,c}

The SVEA-96 Optima3 subbundle has no top tie plate and has a simplified bottom tie plate. The normal rods and the part-length rods rest freely on the bottom tie plate. [

]^{a,c}

The lengths of the part-length rods are increased compared to the SVEA-96 Optima2 design.

As with SVEA-96 Optima2, all rods in the SVEA-96 Optima3 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 Optima3 assembly for the U.S. contains about [ ]^{a,c} than SVEA-96 Optima2.

The M8 nuts and the external compression springs in the subbundle top end of previous SVEA-96 fuel designs have been excluded as a consequence of the replacement of the top tie plate by a spacer.

As with SVEA-96 Optima2, the top end plugs are equipped with a notch for a more safe lift of a single rod as well as a subbundle. The top end plugs for SVEA-96 Optima3 are shortened and simplified due to the replacement of the top tie plate by a spacer.

Since only the tie rods are fastened to the simplified bottom tie plate, the number of M6 nuts in the subbundle is reduced to two.

The bottom end plug design for normal and part-length rods is shortened and simplified since these rods rest freely on the simplified bottom tie plate upper surface. The threaded bottom end plug for the tie rods is shortened due to the simplified bottom tie plate with reduced height.

The starting point of the pellet stack is [ ]^{a,c} due to the simplified bottom tie plate. This allows the possibility to increase the pellet stack and thereof increasing the amount of Uranium in the fuel.

Both tie rods in the subbundle also carry the spacer capture function, which hence is doubled for redundancy. The previous separate spacer capture rod is then replaced by a normal fuel rod.

The fuel assembly individual identification number and subbundle identification previously engraved on the top tie plate is moved to the bottom tie plate.

The bottom tie plate is resting on the flat surface of the fuel channel bottom support. The lateral guiding of the subbundle bottom end in the fuel channel is provided by the lowermost spacer, positioned adjacent to the bottom tie plate.

2.1.1 Lattice and Fuel Rod Types

The fuel bundle has four mechanically different types of fuel rods, namely:

- 76 normal fuel rods
- 8 tie fuel rods (with spacer capture function)
- 8 part-length fuel rods with 2/3rd active length
- 4 part-length fuel rods with 1/3rd active length

The pitch between the central rods is [ ]^{a,c} The lattice configuration is shown in Figure 2-4.

2.2 FUEL SUB-BUNDLE DESCRIPTION

The subbundle consists of 24 fuel rods arranged in a 5x5-1 lattice. Each subbundle is assembled as a separate unit with free standing rods, resting on a bottom tie plate. Two tie fuel rods are guided by their end plugs in the bottom tie plates, fastened by nuts, and keep, together with the ten spacers, the subbundle together as one unit. The tie fuel rods also secure the axial positions of the spacers.

The fuel bundle is shown in Figure 2-5.

The fuel rods consist of a Zircaloy cladding tube sealed by end plugs and containing a stack of fuel pellets. In the upper end of the fuel rod there is a space for axial expansion of the fuel pellet stack and for fission gases released from the fuel pellets. The fuel pellet stack is prevented from moving up into this space during transport by means of a plenum spring. The cladding and the end plugs are made of low corrosion material and the cladding tubes are provided with liner on the inside. The fuel pellet stack consists of sintered and ground uranium dioxide pellets. A few rods also contain pellets with burnable absorber in the form of gadolinium oxide mixed with uranium dioxide. The fuel rods are prepressurized with helium and sealed by welding.

The tie fuel rods are connected to the bottom tie plate by threaded end plugs, extending through the plate, and nuts. They are placed next to the central position and are locked against rotation by slits in the bottom tie plate, which engage the bottom end plugs. Above each spacer position, the tie fuel rods have small heads welded to the cladding tube. These heads provide the spacer capture function. The tie fuel rods permit small movements of the spacer to avoid strong axial forces on the spacer. A typical tie fuel rod is shown in Figure 2-6.

Twelve part-length rods are included in the fuel bundle. Eight part-length rods (two in each subbundle) are placed adjacent to the central water channel. They end about [ ]^{a,c} above the seventh spacer and are about 2/3rd in active length. Four more part-length rods (one in each subbundle) are placed in the outer corners. They end about [ ]^{a,c} above the fourth spacer and are about 1/3rd in active length. The part-length rods have in principle the same design as full-length rods, with plenum spring in the upper end. The part-length rods have the same type of end plugs as the normal fuel rods. The positions of the part-length rods are chosen for shut down margin improvement and, at the same time, to optimize the critical power performance. These part length rods are in the same location as for SVEA-96 Optima2. Typical normal full-length and part-length rods are shown in Figure 2-7 and typical plenum springs for full-length and part-length rods are shown in Figure 2-8.

The fuel pellet design of SVEA-96 Optima3 is identical to that of SVEA-96 Optima2. The fuel pellet design is also the same for uranium oxide pellets and pellets with burnable absorber (BA) containing Gd_2O_3. The fuel pellet design is shown in Figure 2-9.

2.2.1 Bottom Tie Plate

The bottom tie plate is machined from stainless steel bar, type American Iron and Steel Institute (AISI) 304 L, and has the fuel assembly individual identification number and subbundle identification engraved in the side. The number is used for administrative control of the correct positioning of the subbundle, with respect to subchannel position. The bottom tie plate rests on the flat surface of the fuel channel bottom support.

[ ]^{a,c} The bottom tie plate is shown in Figure 2-10.

2.2.2 Spacers

The SVEA-96 Optima3 fuel is equipped with a new, sleeve type spacer design. The material, Aerospace Material Specification (AMS) 5542 (Inconel® X-750), is the same as in previous SVEA and 8x8 spacers. Each spacer consists of 24 cells (sleeves) welded to each other and to 3 frames. The cells are fabricated from Nickel Base Alloy strip, punched, stamped and coiled to form octagonal cells. Every cell has four lines for contact with the fuel rod. The sides of the cell provide the deflection necessary to maintain contact with the rod. [

]^{a,c} The wall thickness is [ ]^{a,c} in the cells and [ ]^{a,c} in the frames. Spacer levels [ ]^{a,c} have a simplified spacer design [ ]^{a,c}. The simplified spacer is shown in Figure 2-11a and the spacer [ ]^{a,c} is shown in Figure 2-11b.

The basic element in the spacer is the sleeve type cell. [

]^{a,c} has shown superior properties with a minimum risk for grid-to-rod fretting as well as a reduced risk for debris hang-up and ensuing fretting. The cells are welded together to form a structure resembling honeycomb.

The vanes are almost identical with the vane in SVEA-96 Optima2 but slightly larger. This vane design is well protected at handling, both at assemblage and during service and inspections.

The cells have "waists" at the attachment to the neighbors. The cell height is [ ]^{a,c} and the height at the waist is [ ]^{a,c}. The waists have several purposes:
[

]^{a,c}

The frame has been designed with the same basic criteria as for SVEA-96 Optima2.

2.3 FUEL CHANNEL

The fuel channel design is basically identical with the SVEA-96 Optima2 fuel channel (Reference 1.0). Only minor changes have been introduced:

- Introduction of a further optimized **TripleWave™** debris filter, the **TripleWave+™** debris filter.
- Adaptation of the bottom support for the new bottom tie plate and the **TripleWave+™** debris filter.
- Slight modification of outer channel-water cross connection dimensions (embossing) for improved manufacturability.
- Including the possibility of using **Low Tin ZIRLO™** channel material as alternative material instead of Zircaloy-2 channel material.

The bottom support, which is machined from stainless steel bar material, type AISI 304 L, is on its inlet side equipped with four **TripleWave+™** debris filter units. The bottom support is designed with grooves for fitting the **TripleWave+™** debris filter units below each subbundle. The **TripleWave+™** debris filter is an evolution of the previous **TripleWave™** debris filter. The legacy from the previous model is obvious; the same design with the three waves, the same material, and the same manufacturing processes. The filter units are secured to the bottom support by lock welds. The filter is built from corrugated or wavy plates, formed from stainless steel sheet metal, type AISI 316 L, which is a well-known and proven material. The filter plates are joined at a large number of weld points, which gives a robust and redundant design.

The **TripleWave+™** debris filter is aimed at catching all debris with length [ ]^{a,c} as those are regarded to pose the largest risk for fretting on the fuel rods.

The plates have a wavy shape across the flow in the inlet, a wavy shape along the flow in the center, and then again a wavy shape across the flow in the outlet, hence the name **TripleWave™**. The wavy shape of the inlet and the outlet edge serves several purposes. It functions as support points where the plates can be welded together. It also forms a grid that provides a first filter for large objects. In the inlet it reorients medium size objects parallel to the flow that are subsequently trapped at their head-on entrance into the filter. Finally, the design of the plates forms a flow path with a smooth and vertical outward flow from the filter outlet.

The channel dimple design is slightly modified (reduced width and depth) compared to SVEA-96 Optima2 for improved manufacturing margins. The cross edge is also modified to maintain the hydraulic communication area between subchannels when dimple depth is reduced.

The fuel channel is shown in Figure 2-13 and the bottom support with **TripleWave+™** debris filter is shown in Figure 2-14.

2.4 HANDLE WITH SPRING

The handle, which is identical to the SVEA-96/-96+/-96 Optima2 handle, is made from stainless steel bar material, type AISI 304 L, and has a double leaf spring of AMS 5542 (Inconel X-750).

After insertion of the four subbundles into the channel, the handle is fastened to the central lifting screw with a nut. The nut is locked in position by deforming an integral washer. For redundancy the handle is also connected to the channel in two corners. Two spring-loaded plungers that extend through openings in the channel wall achieve this connection.

The handle with spring and the mounting to the fuel channel is shown in Figure 2-15.

The handle is designed for lifting with the ordinary handling equipment of the reactor. An individual identification number for the fuel assembly is engraved in the handle.

The top end of the channel and the handle are designed in such a way that the handle can be mounted in only one way, thus assuring correct orientation of the handle relative to the fuel assembly.

The leaf spring has the same function as the leaf springs in other fuel assembly designs, i.e., to interact with the corresponding springs on adjacent fuel assemblies and press the fuel assembly into the corner of the core grid module.

2.5 PLANT DEPENDENT FEATURES

Sections 2.1 to 2.4 provide, in combination with Reference 1.0, a complete specification of the mechanical design features of the fuel product and most of these will be common for each plant specific application. However, some of the mechanical design parameters are determined to accommodate the reactor internals dimensions and co-resident fuel dimensions and may vary with each plant specific implementation. These are identified as compatibility features and are listed below:
[

]^{a,c}

2.5.1 Channel Length & Mechanical Compatibility with Co-resident Fuel

Different generations of BWR plant designs have adopted different active fuel lengths. The range of active fuel lengths for SVEA-96 Optima3 in approved U.S. BWR plant designs is typically between
[ ]^{a,c}.

The length of the channel may be adjusted to accommodate the active fuel length and the co-resident fuel length to ensure worst case differential channel growth between different fuel types. Minimum engagement of the channel springs along with locations and size of channel dimples ensure that proper lateral spacing at the top of the fuel is maintained throughout the life of the fuel. The evaluation is performed as defined in Section 4.2.1.

2.5.2 Fuel Rod/Bundle Length

The active fuel rod length and fuel bundle length varies according to the plant design as described above. The overall length of the fuel rod may be changed for plant specific applications according to the plant specific active fuel length to a length that ensures the fuel rod design criteria described in Section 3.3 are met. The methodology that will be used to assess the design criteria is presented in Section 4.

2.5.3 Channel Bypass Flow Hole Size

Hydraulic compatibility of the fuel bundles is required per the design criterion identified in Section 3.2.4 of CENPD-287-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors" (Reference 2.0) for assembly lift and implicitly in critical power ratio (CPR) assessment of the fuel
[

]^{a,c}

2.5.4 Channel Alignment and Offset

For some plant applications, [

]^{a,c}

2.5.5 Adaptations of Handle Dimensions

Handle dimensions such as lifting beam height and control rod gap fixed support/spacer button dimensions may be adjusted to ensure geometrical compatibility at all conditions with other fuel types as well as core internals, handling equipment, and storage facilities. Handle leaf spring dimensions may also be adjusted to ensure minimum engagement with channel springs as well as matching the spring force of adjacent fuel.

2.5.6 Bottom Tie Plate Flow Hole Size

[

]^{a,c}

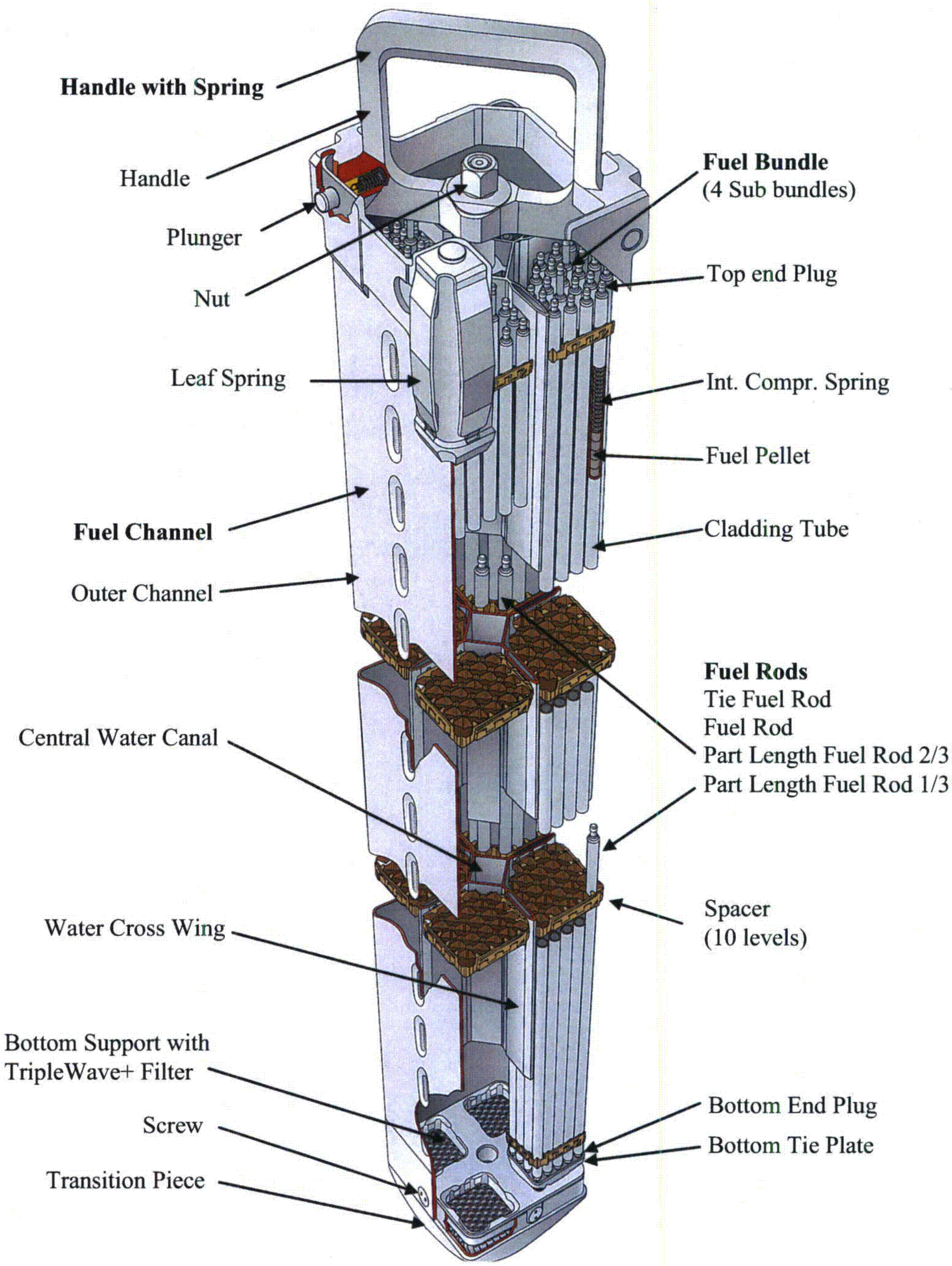


Figure 2-1a SVEA-96 Optima3 Fuel Assembly Overview

a,c

Figure 2-1b SVEA-96 Optima3 Fuel Assembly

a,c

Figure 2-2 SVEA-96 Optima3 Fuel Assembly Cross Section

a,c

Figure 2-3a SVEA-96 Optima3 Assembly and Control Rod Orientation in a C-lattice Plant

a,c

Figure 2-3b Typical Control Rod Gap Dimensions with SVEA-96 Optima3 Fuel in a C-lattice Plant

a,c

Figure 2-4 SVEA-96 Optima3 Fuel Assembly Lattice

a,c

Figure 2-5 SVEA-96 Optima3 Fuel Bundle

a,c

Figure 2-6 SVEA-96 Optima3 Tie Fuel Rod

a,c

Figure 2-7 SVEA-96 Optima3 Normal- and Part-length Rods

a,c

Figure 2-8 Typical Internal Compression Springs Used for the Various Rod Lengths

a,c

Figure 2-9 UO_2 and UO_2+Gd_2O_3 Pellet Dimensions

a,c

Figure 2-10 SVEA-96 Optima3 Bottom Tie Plate

a,c

Figure 2-11a SVEA-96 Optima3 Spacer [ ]a,c

a,c

Figure 2-11b SVEA-96 Optima3 Spacer [ ]a,c

a,c

Figure 2-12 SVEA-96 Optima3 Spacer Cell [ ]a,c

a,c

Figure 2-13 SVEA-96 Optima3 Fuel Channel

a,c

Figure 2-14 Bottom Support with TripleWave+™ Debris Filter

a,c

Figure 2-15 Mounting of Handle with Spring

3 DESIGN CRITERIA

The design criteria described in WCAP-15942-P-A/CENPD-287-P-A (References 1.0 and 2.0) were applied to the reference fuel SVEA-96 Optima3 design without change, but are repeated below to facilitate the reading of this topical report.

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 insertability 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 SRP, Section 4.2 (Reference 1.4), relative to:

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

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

The mechanical integrity design criteria below are provided in three categories:

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

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

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

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

3.1 DESIGN CRITERIA, GENERAL

3.1.1 Normal Operations and AOOs

Criterion

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

Discussion

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

3.1.2 Accident Conditions

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

3.1.2.1 Fuel Rod Mechanical Failure

Criterion

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

Discussion

The methodology for evaluating fuel assembly performance and in illustration the performance of the fuel assembly *for mechanical fracture under seismic/loss-of-coolant accident (LOCA) external loads are described in CENPD-288-P-A, "ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel" (Reference 3.0).*

3.1.2.2 Fuel Coolability

Criterion

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

1. *Cladding embrittlement is limited by requiring that the peak clad temperature (PCT) during a postulated LOCA be less than 1204 °C (2200 °F), 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 U.S. Nuclear Regulatory Commission (NRC) requirements.*

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

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

Discussion

During normal operation and 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 References 3.1 through 3.6.

The Westinghouse methodology for evaluating the consequences of a BWR CRDA and an illustrative application for a core loaded with Westinghouse fuel *during a CRDA is described in CENPD-284-P-A, "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification" (Reference 3.7).*

The Westinghouse methodology for evaluation of the consequences during a seismic plus LOCA event is given in Reference 3.0.

3.1.2.3 Clad Bursting

Criterion

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

Discussion

The Westinghouse methodology for evaluating fuel rupture during postulated LOCAs is described in References 3.1 through 3.6.

3.1.2.4 Excessive Fuel Enthalpy

Criterion

The number of fuel rods predicted to reach assumed fuel failure thresholds during a CRDA shall be input to a radiological evaluation. The assumed failure threshold(s) must be in compliance with NRC requirements.

Discussion

The Westinghouse methodology for evaluating the consequences of a BWR CRDA and an illustration of the application methodology are described in Reference 3.7.

3.1.3 Evaluation Methodology

Criterion

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

Discussion

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

3.1.4 New Design Features

Criterion

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

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

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

Discussion

New design features will be tested with out-of-reactor prototype testing, with Lead Fuel Assemblies, or with a combination of both approaches. As illustrated in Section 7, 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

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

Discussion

The post-irradiation surveillance program described in Section 9 has been fashioned to meet the guidance provided in the SRP. As illustrated by the extensive inspections of the 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

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

Criterion

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

Discussion

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

3.1.8 Burnup

Criterion

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

Discussion

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

3.2.1 Compatibility with Other Fuel Types and Reactor Internals

Criterion

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

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

3.2.2 Geometric Changes in the Assembly during Operation

Criterion

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

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

3.2.3 Transport and Handling Loads

Criterion

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

3.2.4 Hydraulic Lifting Loads during Normal Operation and AOOs

Criterion

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

Discussion

Assembly lifting loads resulting from accident conditions are addressed in Reference 3.0.

3.2.5 Stress and Strain during Normal Operation and AOOs

Criterion

Mechanical failure of assembly components shall not occur. Assembly component dimensions must be maintained within operational tolerances, and functional capabilities shall not be reduced below those assumed in the safety analysis. This criterion is implemented by establishing design limits for stresses, alternatively using collapse load analysis criteria, *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 and collapse load analysis criteria *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

Criterion

The design criterion on assembly component fatigue is that fatigue failure of assembly components shall not occur during normal operation and AOOs.

3.2.7 Fretting Wear of Assembly Components

Criterion

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

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

Discussion

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

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

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

3.2.8 Corrosion of Assembly Components

Criterion

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

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

Discussion

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

3.2.9 Hydriding of Zircaloy Assembly Components other than Fuel Rods

Criterion

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

3.3 DESIGN CRITERIA, FUEL RODS

3.3.1 Rod Internal Pressure

Criterion

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

Discussion

This criterion is based on the recognition that the physical phenomenon to be avoided is an increase in the pellet-to-cladding gap at high burnups which could cause a rapid fuel pellet temperature increase and fission gas release resulting from the thermal feedback mechanism associated with an increasing gap. This criterion is believed to meet the intent of the SRP guidance. The fuel rod internal pressure must be limited to avoid an increase in gap size which could cause positive thermal feedback and rapidly increasing pellet temperatures. The 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

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 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

These criteria result from the requirements that the fuel rods shall not be damaged due to excessive fuel cladding strains. The 1% limit on cladding strain is in compliance with Reference 1.4, SRP Section 4.2.

3.3.4 Hydriding

Criterion

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

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

Criterion

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

3.3.6 Cladding Collapse (Elastic and Plastic Instability)

Criterion

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

3.3.7 Cladding Fatigue

Criterion

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

3.3.8 Cladding Temperature

Criterion

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

3.3.9 Fuel Temperature

Criterion

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

3.3.10 Fuel Rod Bow

Criterion

Excessive fuel rod bowing shall be precluded for the design life of the fuel assembly. Fuel rod bowing shall be evaluated, and 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.

4 DESIGN METHODOLOGY AND SVEA-96 OPTIMA3 EVALUATION

The design methodology described in WCAP-15942-P-A/CENPD-287-P-A/WCAP-15942-P Supplement 1 (References 1.0, 2.0, and 4.3) was applied to the reference fuel SVEA-96 Optima3 design without change with the exception of a few improvements identified in Section 4.3.

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 3.0 through 3.7 and 4.0 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 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 Optima3 *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* Optima3 *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 Optima3 *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 is consistent. Supporting information in Section 4.3 which does not directly correspond to any criteria in Section 3.3 has been provided in Section 4.3.1.*

Mechanical Properties

The materials used in the SVEA-96 Optima3 *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 NUREG/CR-0497, "A Handbook *of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior" (Reference 4.2), 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, ***Low Tin ZIRLO™*** *channel material, and Stainless Steel components currently used for the fuel assembly design evaluations are provided in Table 4-1.*

[

]a,c

The dependence of irradiation on cladding yield and tensile strength are specifically treated in STAV7.2 and are based on Reference 4.2 as described in Reference 1.2 Appendix A.

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 water cross 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, S_m, [

]^{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.

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 of 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}

These limits need not be satisfied at a specific location if it can be shown that the design loadings do not exceed 2/3rd of the test collapse load determined in compliance with Section III of Reference 1.3.

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

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

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

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

Design Loads

Design loads are established 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

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 FUEL ASSEMBLY COMPONENTS EVALUATION

4.2.1 Compatibility with Other Fuel Types and Reactor Internals

Methodology

For each plant application of a Westinghouse fuel assembly type (e.g. SVEA-96 Optima3*) 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.

Control rod interference due to the combined effects of channel bow and creep is evaluated to be sufficiently low during the life of the fuel bundle to ensure that maximum SCRAM insertion times for operable rods given in the technical specifications are not exceeded. This criterion is confirmed on a plant specific basis for SVEA fuel by [

]a,c.

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

[

]a,c

Sample Application

This section contains an example of the methodology for evaluating compatibility in a mixed core by evaluating the SVEA-96 Optima3 *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* Optima3 *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}

Therefore, the SVEA-96 Optima3 *assembly is concluded to be compatible with the resident non-SVEA assembly with regard to axial growth.*

[

]^{a,c}

[

]^{a,c}

The SVEA-96 Optima3 *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}

This example demonstrates the compatibility of the SVEA-96 Optima3 *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 Optima3 *assembly and control rod orientation for a full core of SVEA-96* Optima3 *fuel in a BWR/6 C-Lattice plant is shown in Figures 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 27 mm. The available minimum space for the detector is* [ ]^{a,c} *when surrounded by SVEA-96* Optima3 *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 Optima3 *channel dimension on a side at BOL is* [ ]^{a,c}. *From Figure 2.3a, this maximum dimension provides at least* [ ]^{a,c}

[ ]^{a,c}. *Therefore, adequate clearances are available at BOL to avoid interference.*

The effects of irradiation on the SVEA-96 Optima3 *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}

Due to the Nb presence in ***Low Tin ZIRLO***™ *material,* [

]^{a,c}

Consequently, the calculations and assumptions described below are still valid.

Since the correlation has [

]^{a,c}

[

]a,c

Application of the creep model described above demonstrates that for the SVEA-96 Optima3 *channel the combination of the axial variations of* [

]a,c

Percent of Period	*Percent Core Flow*

a,c

[

]a,c

Location	*Deformation (mm)*	*Direction*

a,c

[

]a,c

Channel Bow

[

]a,c

Measurements on irradiated SVEA channels have shown a good dimensional stability. SVEA channel bow in a symmetrical lattice is shown in Figure 4.2-6. [

]a,c

Figure 4.2-7 shows the SVEA Zircaloy-2 channel bow database for a symmetrical core lattice, including a statistical evaluation. 2 x standard deviation have been calculated for intervals of 5 MWd/kgU (first interval 0-5 MWd/kgU, second interval 5-10 MWd/kgU etc.) and each interval is represented by a midpoint of the interval. The average bow is, as expected in symmetrical lattice, about zero.

[

]a,c

[

]a,c

The methodology for evaluation of compatibility with control rod, introduced in the response to request for additional information (RAI)-15 of Reference 1.0, *included an extensive statistical evaluation of the Westinghouse SVEA-10x10 channel bow database, including previously used Zry-4 channel material. Control rod* [

]a,c *as input in the analysis for SVEA-96* Optima3 with the current Zry-2 and **Low Tin ZIRLO**™ channel materials.

[

]a,c

[

]a,c

Sample Application to **a BWR/3-4** ***plant***

Applying the methodology described in the response to RAI-15 of Reference 1.0 *to SVEA-96* Optima3 *in* a BWR/3-4 *asymmetric lattice plant – results:*

Maximum bow toward control rod:

[

]a,c

Manufacturing tolerance, creep and elastic deflection (one channel side):

[

]a,c

Nominal control rod roller/pad - channel clearance (average top and bottom):

[ ]a,c

Nominal control blade - channel clearance (average top and bottom):

[ ]a,c

Maximum channel - control rod interference – roller/pad:

[

]a,c

Maximum channel - control rod interference – blade:

[

]a,c

The calculated interference is, [

]a,c

[

]a,c

Discussion

As mentioned above, *the increased dimensional stability of the SVEA-96* Optima3 *channel and its greater flexibility substantially reduces the risk of unacceptable control rod interference relative to open lattice designs.* The SVEA-channel is also more flexible than known [

]a,c

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 has been indicated or reported. Therefore, it is concluded that SVEA-96 Optima3 *fuel in C- and D-lattice BWR reactors will not pose a risk of jeopardizing control rod maneuverability.*

The SVEA-96 Optima3 *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, this example demonstrates the compatibility of the SVEA-96 Optima3 *assembly with control rods and detectors. Similar compatibility evaluations as the one presented in the sample application 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 Optima3 *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 design (e.g. SVEA-96 Optima3*), 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* Optima3 *design for the design lifetime of the assembly:*

[

]^{a,c}

4. *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. *Bottom Tie Plate*

      [ ]^{a,c}

   b. *Assembly Handle Configuration*

      [ ]^{a,c}

   c. *Spacer Capture Function*

      [ ]^{a,c}

   d. *Spacer*

      [ ]^{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 Optima3 *assembly components as a function of burnup.* [ ]^{a,c}

[

]^{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.

1. Sub-bundle Growth

The differential growth between the SVEA-96 Optima3 *channel and subbundles based on the most current data base can be summarized as follows:*

[

c

]^{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 4.2-12:

[

]^{a,c}

[

]^{a,c}

a. Normal Fuel Rods

Minimum margin to maintain a satisfactory guidance for normal fuel rods is calculated below:

Maximum differential rod growth between the shortest tie fuel rod and the normal fuel rods within a subbundle is expected to be [ ]^{a,c} according to above.

The axial distance between the spacer and the heads of a tie fuel rod is [ ]^{a,c} BOL, see Figure 4.2-12.

Maximum differential rod growth in combination with spacer movement up to the tie rod heads reduces the margin for minimum guidance of a full-length fuel rod with [ ]^{a,c}, which is less than the minimum margin for satisfactory guidance, [ ]^{a,c}.

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

The maximum rod length above the spacer for a full-length rod is calculated below:

Maximum rod length above the spacer is [ ]^{a,c} BOL.

The maximum differential growth of a normal fuel rod is [ ]^{a,c} compared to the shortest tie fuel rod.

The maximum rod length above the spacer is [ ]^{a,c} mm. This is acceptable, since this rod length above the spacer is covered by performed fretting tests, with no signs of fretting.

Also in the worst case according to above, the requirement on freedom from fretting wear is fulfilled.

b. Part-length Fuel Rods

[ ]^{a,c} Therefore, the requirements on rod guidance in the spacer and freedom from fretting wear are fulfilled also for part-length fuel rods.

3. Fuel Rod Guiding in Bottom Spacer

A known mechanism that can lift a fuel rod during operation is high friction between top tie plate hole and top end plug extension, combined with differential rod growth within the subbundle. This mechanism is eliminated in SVEA-96 Optima3, since the top tie plate is replaced by a spacer. [

]^{a,c} and the margin for normal rods are larger than for the subbundle, due to the resulting lifting force by the spacers acting on the subbundle. Subbundle and fuel assembly lift forces are evaluated in Section 4.2.4.

The full guidance below the bottom spacer is for normal and part-length rods [

]^{a,c}

4. Performance of bottom tie plates, assembly handle configuration, spacer capture function, and spacer:

a. Bottom Tie Plates

[

]^{a,c}

b. *Assembly Handle Configuration*

[

]^{a,c}

[

]a,c

c. Spacer Capture Function

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

[

]a,c

d. Spacer

The SVEA-96 Optima3 spacer is a sleeve type design with linear contact to the fuel rod rather than point contact used in previous SVEA and 8x8 spacers. There are four lines in each spacer cell supporting the fuel rod, while previous SVEA spacers were equipped with six contact points (two springs and four fixed supports) per spacer cell. The material, Nickel Base Alloy type AMS 5542 (Inconel X-750), is the same as in previous SVEA and 8x8 spacers and the basic manufacturing techniques for spacer and spacer components are the same [

]a,c

[

]^{a,c}

Spacers with the same material as the SVEA-96 Optima3 spacer have been used in 8x8, SVEA-64, SVEA-100, SVEA-96/96+, SVEA-96 Optima, and SVEA-96 Optima2 assemblies. Extensive reactor experience has not shown any indication of stress corrosion cracking or fatigue failure. [

]^{a,c} Furthermore, laboratory tests described in Section 8 demonstrate that the SVEA-96 Optima3 spacer can withstand repeated seismic-type loads. [

]^{a,c}

Therefore, reactor experience with the SVEA-96 Optima3 spacer, as well as other spacer designs of the same material, manufacturing techniques and stress level, 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.

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}

[

]^{a,c}

Sample Application

The current design loads for shipping and handling of SVEA-96 Optima3 *fuel for U.S. applications can be summarized as follows:*

Load Description	***Design Load***

a,c

Sample Evaluation of Response to Shipping Loads - SVEA-96 Optima3

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 tests 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. [ ]^{a,c} *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.*

To date, [ ]^{a,c} fully assembled (channeled) SVEA-96 Optima2 fuel assemblies have been shipped in the U.S. from the Westinghouse Columbia fuel fabrication facility.

Similar transport test as described above has also successfully been performed in Europe with fully assembled (channeled) SVEA-96 Optima3 fuel in EMBRACE (former RA-2/3) shipping container. Also here the transport route was chosen to conservatively represent the road quality experienced when performing fuel transports between the Westinghouse facility and reactor sites. The total route distance was [

]^{a,c}.

Similar handling tests as described above were also performed as part of the qualification. Shock loads in excess of [ ]^{a,c} were applied and the dimensional changes found were within what is considered as normal due to assembly and disassembly. All requirements were fulfilled.

The results from the performed tests have verified that SVEA-96 Optima3 can be transported and handled with the tested equipment without damages and SVEA-96 Optima3 fuel is currently being shipped in reload quantities in EMBRACE (RA-2/3) shipping containers in Europe.

Sample Evaluation to Response to Handling Loads – SVEA-96 Optima3

The evaluation of the SVEA-96 Optima3 *assembly for design handling loads addresses the stresses in the channel assembly, the lifting handle, the bottom tie plate, and the tie rods.*

Channel

[

]^{a,c}

[

]^{a,c}

Bolt Stresses	*Calculated (N/mm^2)*	*Allowable (N/mm^2)*

a,c

[

]^{a,c}

Handle

The sample evaluation of the handle is performed on a SVEA-96 Optima3 *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/SVEA-96 Optima2/SVEA-96 Optima3 *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 is thus met for the handle.

Bottom Tie Plate

[

]a,c

Tie Rods

[

]a,c

[

]a,c

	Maximum Calculated stress N/mm²	*Maximum Allowable Stress (S_m) N/mm²*

a,c

[

]a,c

Therefore, margins to very conservative stress limits for the tie rods during handling operations are substantial and the stresses would be within the limits also if only one of the tie fuel rods would take the total load.

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 is 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 3.0.

[

]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 Optima3 assemblies in a BWR/6 plant.

Stresses in SVEA-96 Optima3 fuel assembly components have been evaluated for loads during normal operation and AOOs for several BWR plants. [

]a,c

Spacer

The SVEA-96 Optima3 spacer material, Nickel Base Alloy type AMS 5542 (Inconel X-750), is the same as in previous SVEA and 8x8 spacers and the basic manufacturing techniques for spacer and spacer components are the same [

]a,c

As discussed in Section 8, SVEA-96 Optima3 spacers have been demonstrated to be capable of withstanding lateral seismic-type loads. [

]a,c

Channel

[

]a,c

[

]a,c

[

]a,c

a,c

Location	Displacement [mm]

[

]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 $\frac{ni}{Ni}$ 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 Nuclear Science and Engineering, Vol. 20, "Fatigue Design Basis for Zircaloy Components" (Reference 4.4) 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 [

]^{a,c}.

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

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 Optima3 *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* Optima3 *channel.*

[

]^{a,c}

Relative Bundle Power	*Relative Core Flow, %*	*Percent of Channel Life-time*	*Number of Cycles*	*Maximum Channel Pressure Load kPa*

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 Optima3 design has been minimized by retaining materials from previous designs, and designing the fuel to maintain or improve margins compared to previous designs for which the effectiveness in minimizing wear has been demonstrated. In addition, both SVEA-96 Optima3 prototype loop tests and post irradiation examinations of SVEA-96 Optima3 fuel and fuel components have demonstrated that wear of SVEA-96 Optima3 components is minimal and does not impair the function of the assemblies.

[

]a,c

[

]a,c

The conclusion from the tests is that the mechanical behavior of the SVEA-96 Optima3 fuel is satisfactory and that reactor operation without unacceptable wear for the design life of the fuel caused by fretting can be expected.

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

[

]^{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 Optima3 design (Section 5.2.2), the SVEA-96 Optima3 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. The end plugs are made of Zircaloy-2 material, [ ]^{a,c}. By replacing the top tie plate with a spacer in SVEA-96 Optima3 the risk of fuel rod bow caused by differential rod growth combined with high friction in the top tie plate due to excessive top end plug corrosion is eliminated. Extensive operating experience and post irradiation examinations have verified satisfactory corrosion behavior of the end plugs.

As discussed in Section 4.2.2, the spacer-capture head weld does not reduce the corrosion resistance of the fuel rod.

Based on the extensive in-reactor experience with Inconel X-750 spacers discussed in Section 7,
[

]^{a,c} *The level of corrosion observed on these* *spacers is much less than that which would impair the function of the spacer or lead to sufficient corrosion to significantly impact the activity level of the coolant.*

Figures 4.2-20a and 4.2-20b show the measured maximum and average oxide thickness as a function of the bundle average burnup for SVEA channels. As shown in these figures, [

]^{a,c}.

Westinghouse has gathered an extensive database pertaining to oxide thickness measurements with Zircaloy-4, Zircaloy-2 super-α (Continuous anneal Furnace, CF) and β-quenched Zircaloy-2 channels.
[

]^{a,c}

During the inspections reflected by the data in Figures 4.2-20a and 4.2-20b, channel welds are examined. These inspections have not revealed any substantial corrosion in the vicinity of the channel welds that could impact channel functionality.

Low Tin ZIRLO™ *channels are continuously being examined with respect to corrosion,* [

]^{a,c}. *Shadow corrosion is believed to drive hydrogen pick-up, and this early-life hydrogen enrichment in combination with fast neutron fluence is believed to drive channel distortion.*
[ ]^{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 for more than *35 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.

SVEA-96 **Optima3** ***Component***	***Westinghouse Maximum Cobalt Specification Requirement (wt %)***

a,c

4.2.9 Hydriding of Zirconium 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 Zirconium based assembly components is provided in this section.

[

]a,c

[

]a,c

The following measures are taken to minimize the impact of 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 Zirconium based materials 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}

	Zircaloy-2	*Low Tin ZIRLO™ material*

a,c

[

]a,c

Table 4-1 Typical Fuel Assembly Material Properties

a,c

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

a,c

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

a,c

a,b,c

Figure 4.2-1 SVEA Channel Growth

a,c

Figure 4.2-2 SVEA-96 Optima3 Assembly (BOL) and non-SVEA Assembly (BOL)

a,c

Figure 4.2-3 SVEA-96 Optima3 Assembly (BOL) and non-SVEA Assembly (EOL)

a,c

Figure 4.2-4 SVEA-96 Optima3 Assembly (EOL) and non-SVEA Assembly (BOL)

a,b,c

Figure 4.2-5 SVEA-64 Channel Creep Deformation

a,b,c

Figure 4.2-6 SVEA-10x10 Channel Bow Measurements in a Symmetric Lattice Plant

a,b,c

Figure 4.2-7 Statistical Evaluation of Zry-2 Channel Bow in a Symmetric Lattice Plant

a,b,c

Figure 4.2-8 SVEA-10x10 Channel Bow Measurements in Asymmetric Lattice Plants

a,b,c

Figure 4.2-9 SVEA-96 Optima2/Optima3 Fuel Rod Growth

a,b,c

Figure 4.2-10a SVEA-96 Optima2/Optima3 Differential Fuel Rod Growth

a,b,c

Figure 4.2-10b SVEA-96 Optima2/Optima3 Differential Growth of Tie Fuel Rods

a,c

Figure 4.2-11 Clearance Between Subbundle and Handle

a,c

Figure 4.2-12 Fuel Rod Growth Allowances

a,b,c

Figure 4.2-13 Spacer Spring Relaxation

a,b,c

Figure 4.2-14 Low Tin ZIRLO™ Material Model and Tensile Test Curves

a,c

Figure 4.2-15 SVEA-96 Optima3 Channel Section for FE-modeling

a,c

Figure 4.2-16 FE-Model of SVEA-96 Optima3 Channel

a,c

Figure 4.2-17 Collapse Load Diagram of SVEA-96 Optima3 Channel

a,b,c

Figure 4.2-18 Equivalent Plastic Strain at SVEA-96 Optima3 Channel Spot Weld

a,b,c

Figure 4.2-19 Stress Range at SVEA-96 Optima3 Channel Spot Weld

a,b,c

[redacted]

Figure 4.2-20a Maximum SVEA Channel Oxide Thickness

a,b,c

Figure 4.2-20b Average SVEA Channel Oxide Thickness

4.3 FUEL RODS EVALUATION

The Westinghouse fuel rod evaluation methodology was originally licensed in Reference 2.0. This methodology was then updated in Reference 1.0 in 2006 for SVEA-96 Optima2 to reflect code improvements and the current industry practices. The methodology in this submittal remains basically unchanged, except for changes to address the SVEA-96 Optima3 hardware changes and updates to current industry practices.

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 4.3.12 describe the methodologies and provide a specific application to SVEA-96 Optima3 for evaluation relative to the design criteria described in Sections 3.3.1 through 3.3.10.

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.
[

]^{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 4.3.1.

4.3.1 Fuel Rod Power Histories

The methodology for fuel rod power history selection is unchanged compared to Reference 1.0 WCAP-15942-P-A.

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} *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

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 AOOs.

Power History Envelopes

Using the base power histories, [

]^{a,c}

Thermal Mechanical Operating Limits (TMOLs)

The 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 LHGR operating limit which should not be exceeded during normal operation. [

]a,c The enveloping LHGR and sample SPH1 and SPH3 for UO_2 fuel rods are shown in Figures 4.3.1-1 and 4.3.1-2, respectively. Similarly, the enveloping LHGR and sample SPH1 and SPH3 for UO_2-Gd_2O_3 fuel rods are shown in Figures 4.3.1-3 and 4.3.1-4, respectively.

Treatment of Part-Length Rods

[

]a,c

Treatment of AOO Power Ramps

[

]a,c

[

]a,c

Axial Power Shape

[

]a,c

For the sample analysis a generic axial power profile will be used.

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 4.3.1-1 UO_2 TMOL and Corresponding SPH 1

a,c

Figure 4.3.1-2 UO_2 TMOL and Corresponding SPH 3

a,c

Figure 4.3.1-3 UO_2-Gd_2O_3 TMOL and Corresponding SPH 1

a,c

Figure 4.3.1-4 UO_2-Gd_2O_3 TMOL and Corresponding SPH 3

4.3.2 Rod Internal Pressure

Methodology

The rod internal pressure methodology is unchanged compared to Reference 1.0.

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 Optima3 *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. [

]^{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. [

]^{a,c}

[

]a,c

Maximum Internal Pressure

[

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

A RIP analysis was then performed to accommodate the potential impact of AOOs [

]a,c

[

]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 4.3.2-1.

Table 4.3.2-1 Fuel Rod Maximum Internal Pressures (MPa)						

a,c

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

]a,c

a,c

[ ]

Figure 4.3.2-1 Irradiation Hardening of BWR Cladding

a,c

[ ]

Figure 4.3.2-2 Critical NCLO Pressure Limit

4.3.3 Cladding Stresses

Methodology

The methodology for cladding stress evaluation has been changed compared to Reference 1.0. The VIK-3 code is replaced with finite element simulations done in the ANSYS program described in Section 6. The methodology follows the directives in Reference 1.3 ASME Boiler and Pressure Vessel Code 2010, Section III, Subsection NB.

[

]a,c was selected for this example.

Cladding Thickness

The measured yield limit and tensile stress is for the base material together with the liner. The liner is however softer than the base component and the mechanical properties of the base components are higher than showed by data from tensile testing of the bi-material tube. The tensile load capacity of the bimaterial cladding tube is expressed as:

$$1.5 \cdot Sm \cdot A_{tot} = R_{base} \cdot A_{base} + R_{liner} \cdot A_{liner}$$

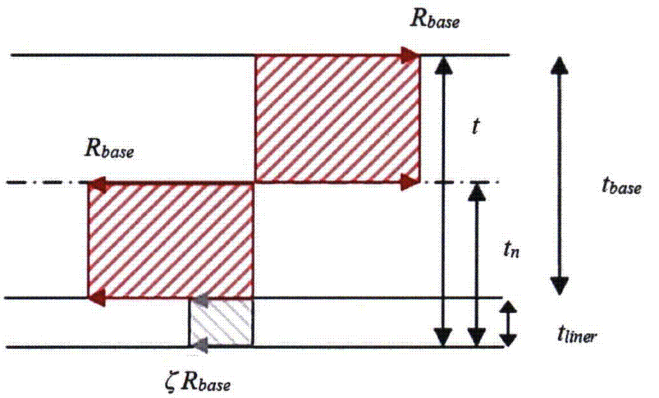
Where A_{tot} is total cross section area, A_{base} is base component cross section area and A_{liner} is liner cross section area. R_{base} is yield limit of base component and R_{liner} yield limit of liner material. The yield limit of the liner material is specified as:

$$R_{liner} = \zeta \cdot R_{base}$$

where ζ is unknown and belongs to the interval $0 \le \zeta \le 1$ because liner is softer than base component. The yield limit of the base component is then expressed as:

$$R_{base} = \frac{1.5 \cdot Sm \cdot A_{tot}}{A_{base} + \zeta \cdot A_{liner}}$$

The stress state at moment collapse is drawn in Figure 4.3.3-1 below, once again assuming that nonlinear material behaviors of the base component and liner are described by ideally plastic material models.



t	total cladding thickness
t_n	neutral line position
t_{liner}	liner thickness
t_{base}	base component wall thickness

Figure 4.3.3-1 Stress Distribution through Cladding at Moment Collapse

Collapse moment (moment/length unit) for the bimaterial tube is:

$$M_{bimaterial} = R_{base} \cdot (t - t_n)^2$$

Where the position of the neutral line is:

$$t_n = \frac{t^2 - t_{liner}^2 (1 - \zeta)}{2 \cdot (t - t_{liner} (1 - \zeta))}$$

The stress calculation of the cladding tube is based on geometry that guarantee that load and moment capacity of the geometry is conservatively calculated and derivation of allowable stress is based on material specification. Thus, find the wall thickness in the stress calculation model that gives the same collapse moment as described by the bimaterial model presented above. The yield limit in the material specification is by definition 1.5*Sm and collapse moment (moment/length unit) is therefore:

$$M_{\text{mod}\,el} = \frac{3 \cdot Sm \cdot t_{\text{mod}\,el}^2}{8}$$

The equivalent wall thickness in the stress calculation model is defined as:

$$M_{model} = M_{bimaterial}$$

giving

$$t_{model} = (t - t_n)\cdot\sqrt{\frac{8\cdot R_{base}}{3\cdot Sm}} = 2\cdot\left(t - \frac{t^2 - t_{liner}^2(1-\zeta)}{2\cdot(t - t_{liner}(1-\zeta))}\right)\cdot\sqrt{\frac{A_{tot}}{(A_{base} + \zeta\cdot A_{liner})}}$$

[

]a,c

Mechanical Data of Liner Material

The measured yield limit and tensile stress is for the base material together with the liner. The liner is however softer than the base component and the mechanical properties of the base components are higher than showed by data from tensile testing of the bi-material tube. That is a conservative approach for the base material.

[

]a,c (tubes delivered from SANDVIK to Westinghouse between years 2000 and 2011).

Liner material has been tensile tested and results are summarized below.

Test no.	Room Temperature			300 °C		
	$R_{p0.2}$ (MPa)	$R_{p1.0}$ (MPa)	R_m (MPa)	$R_{p0.2}$ (MPa)	$R_{p1.0}$ (MPa)	R_m (MPa)

a,c

These data above is used to determine parameter ζ in the calculation of conservative cladding thickness.

As stated above the standard deviation of yield stress and tensile strength is [

]a,c.

Cladding Temperature in Plenum Regions

The cladding temperature, T_{Clad}, in the plenum region in the primary stress evaluation shall be [ ]a,c higher than the coolant temperature:

[ ]a,c

Acceptable Differential Pressure (MPa)

[

]a,c

The input data used in the STAV7.2 calculations are given in Section 5 and below

Parameter	Deviation from Nominal Value	Value

a,c

The results of these calculations are summarized in Table 4.3.3-1.

Table 4.3.3-1 Maximum Differential Pressure Over Cladding				
	Coolant Pressure	Cladding Temperature	Maximum Allowed Differential Pressure	Calculated Differential Pressure Over Cladding

a,c

[ ]

Since the maximum allowed differential pressure exceeds the calculated differential pressure over the cladding [ ]^{a,c} it is concluded that the margin to the stress limits for the SVEA-96 Optima3 will be acceptable for any credible BWR application.

4.3.4 Cladding Strain

The basic methodology identified in Reference 1.0 is unchanged.

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

Parameter	Deviation from Nominal Value	Value

a,c

[

]a,c

[

]^{a,c}

The resultant power history for clad strain calculation is shown in Figure 4.3.4-1.

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 4.3.4-2. [
]^{a,c}

This example demonstrates that ample margins to cladding strain limits are available for peak rod average burnups to [ ]^{a,c}.

a,c

Figure 4.3.4-1 SVEA-96 Optima2 Limiting Strain Power History

a,c

Figure 4.3.4-2 Maximum SVEA-96 Optima2 Transient Cladding Strain

4.3.5 Hydriding

Methodology

The methodology identified in Reference 1.0 is unchanged.

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* Optima3 *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.

<u>*Control of Hydrogen Inside the Fuel Rod*</u>

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 Optima3 *fuel rod manufacturing are currently applied to minimize the hydrogen trapped in a sealed fuel rod:*

Component	***Hydrogen or Equivalent H_2O/UO_2 by Weight***
[	]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 SRP Reference 1.4).

<u>*Hydrogen Pickup in Service for SVEA-96* Optima3 *Cladding*</u>

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.

Figure 4.3.5-1 shows the hydrogen pick-up for the modern **LK3/L**™ cladding as a function of burnup for each sample. The **LK3/L**™ cladding has a low and stable hydrogen pick-up up to the burnup level expected at the assembly end of life. This example demonstrates that ample margins to the hydrogen content limits of [ ]a,c are available for fuel rod average burnups to [ ]a,c.

[

]a,c

a,b,c

Figure 4.3.5-1 Total Hydrogen Concentration versus Burnup

4.3.6 Cladding Corrosion

Methodology

The methodology identified in Reference 1.0 is unchanged.

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

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

]^{a,c}

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

1. [

]^{a,c}

b. [

]a,c

Sample Application

This example is for the **LK3™** Zircaloy-2 cladding currently utilized for the SVEA-96 Optima3 assembly.

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 **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 4.3.6-1 and 4.3.6-2, respectively. The data include measurements on SVEA-96 Optima2 and SVEA-96 Optima3 fuel assemblies. Note the data points labeled as "two-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,b,c

Figure 4.3.6-1 Rod Average Oxide Thickness

a,b,c

Figure 4.3.6-2 Rod Maximum Oxide Thickness

4.3.7 Cladding Collapse (Elastic and Plastic Instability)

Methodology

The basic methodology identified in Reference 1.0 is unchanged.

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. For creep, collapse margin from to the projected irradiation time to the creep collapse time will be maintained. The creep collapse time will be identified as the time when rapid increase in ovality (simulated by the infinite slope of the ovalization curve) as agreed upon in RAI-6 of Reference 1.0.

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 Optima3 *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:

<u>*Instantaneous Collapse at Beginning of life*</u>

[

]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

Rod Power (kW/m)	Time to [ ]a,c (hours)	STAV EOL Oxide Thickness Microns	Collapse Time Hours	Collapse Safety Factor

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 [ ]a,c.

a,c

Figure 4.3.7-1 Calculated Worst-case Ovality as a Function of Time for Constant Power of 25 kW/m

4.3.8 Cladding Fatigue

The methodology identified in Reference 1.0 WCAP-15942-P-A 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'Donnell and Langer in Reference 4.4 is used to calculate the fatigue usage factors. [ ]^{a,c} 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}

Sample Application

[

]^{a,c}

Example of Fatigue Calculation Due to Fuel Rod Power Changes

Load Follow Cycles

[

]^{a,c}

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

[

]^{a,c}

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

]^{a,c}

Start-Up Cycles

[

]^{a,c}

[

]a,c *which are less than 1.0.*

Therefore, the results demonstrate that the SVEA-96 Optima3 *fuel design, has considerable margin to fatigue failure for any credible reactor operation to peak rod average burnups of* [ ]a,c.

4.3.9 Cladding Temperature

The 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 Optima3 fuel is documented in WCAP-17794-P, "10x10 SVEA Fuel Critical Power Experiments and New CPR Correlation: D5 for SVEA-96 Optima3" (Reference 4.5).

4.3.10 Fuel Temperature

The methodology identified in Reference 1.0 is unchanged.

Methodology

The objective of this analysis is to predict the maximum fuel temperature in SVEA-96 Optima3 *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. 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 are shown in Figures 4.3.1-1 through 4.3.1-4. The maximum fuel centerline temperature power history, including AOO transients, is shown in Figure 4.3.10-1. [

]^{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 4.3.10-2 through 4.3.10-4 for the UO_2 rod designs (full-length (FL) and part-length (PL) (2/3rd and 1/3rd length), respectively), and are shown in Table 4.3.10-5 for UO_2-Gd_2O_3 rods also referred to as burnable absorber (BA) rods. *A summary is provided in Table 4.3.10-6.*

Similarly, the results for the "limiting centerline temperature" rod which includes the AOO transients are shown in Table 4.3.10-7

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

Table 4.3.10-1 Parameters and Values used for Fuel Temperature Uncertainties				
Parameter		***Units***	***Nominal***	***Limiting***

a,c

Table 4.3.10-2 Maximum Fuel Temperature in FL-UO_2 Rods				
	Burnup at Max Temp.	***Melting Temperature***	***Max Temp. NOM***	***Max Temp UB***
Power History	***(MWd/kgU)***	***(°C)***	***(°C)***	***(°C)***

a,c

Table 4.3.10-3 Maximum Fuel Temperature in PL-2/3rd UO_2 Rods				
Power History	***Burnup at Max Temp.***	***Melting Temperature***	***Max Temp. NOM***	***Max Temp UB***
	(MWd/kgU)	***(°C)***	***(°C)***	***(°C)***

a,c

Table 4.3.10-4 Maximum Fuel Temperature in PL-1/3rd UO_2 Rods				
	Burnup at Max Temp.	***Melting Temperature***	***Max Temp. NOM***	***Max Temp UB***
Power History	***(MWd/kgU)***	***(°C)***	***(°C)***	***(°C)***

a,c

Table 4.3.10-5 Maximum Fuel Temperature in Gadolinia Rods				
	Burnup at Max Temp.	***Melting Temperature***	***Max Temp. NOM***	***Max Temp UB***
Power History	***(MWd/kgU)***	***(°C)***	***(°C)***	***(°C)***

a,c

Table 4.3.10-6 Summary of Maximum Pellet Centerline Temperatures		
	Melting (°C)	*Max UB Temp (°C)*

a,c

Table 4.3.10-7 Maximum AOO UO_2 Pellet Centerline Temperatures		
Rod Average Burnup	*Melting Temperature*	[ ]a,c *Fuel Centerline Temperature*

a,c

a,c

Figure 4.3.10-1 Transient Power History (AOO) for Maximum Temperatures

4.3.11 Fuel Rod Bow

Methodology

The methodology identified in Reference 1.0 is unchanged and the chapter has been updated with the Optima3 *experience.*

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 Optima3 *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

The SVEA-96 Optima3 *bundle design includes the following design features to address these phenomena and minimize fuel rod bow:*

[

]a,c

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

]a,c *Therefore, based on Westinghouse experience with SVEA fuel,*

fuel rod bow in U.S. plants is not expected to be significant. This conclusion will be confirmed by continuing post-irradiation examination programs as described in Section 9.

[

]a,c

[

]a,c

4.3.12 Pellet-Cladding Interaction

The basic methodology described in Reference 1.0 is unchanged and should be applied for Westinghouse fuel without liner.

Methodology

The most effective measure in the Westinghouse long-tem program for PCI failure mitigation has been the development of the modern Westinghouse fuel rod design with the tin-alloy liner (described in Section 2.5.2 in Reference 2.0). The cladding with liner is a standard feature of Westinghouse current products since the introduction of SVEA-96 Optima2 fuel. The efficiency of the liner has been demonstrated by the extensive operation experience of the different Westinghouse fuel designs, without ever detecting any PCI failures, in either the 8x8 or the 10x10 lined fuel products.

As stated in the SRP, Reference 1.4, here are no specific criteria for fuel failures resulting from 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. In addition to this, no specific design criteria are applied to PCI. However, Westinghouse has implemented the cladding liner and institutes generic/plant specific PCI guidelines and best practices that add additional levels of protection against PCI in addition to the 1% strain and fuel melting criteria.

The PCI best practices/guidelines include components such as ramp rate restrictions, conditioning thresholds, and preconditioning requirements. Both the Westinghouse hardware and best practices have been proven in various power ramp tests, exposing the fuel rods to very challenging conditions. Westinghouse routinely evaluates plant operation, fuel duty, and new data for incorporation into the PCI best practices.

4.4 STEADY-STATE INITIALIZATION OF TRANSIENTS AND ACCIDENTS

The methodology for initializing various dynamic analyses with STAV7.2 results is basically the same as described in Reference 1.0. A few improvements were introduced and are identified below.

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}

Some dynamic analyses are performed on a cycle-specific basis. If it cannot be confirmed that the 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.

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 RPA 90-90-P-A, "BISON – A One Dimensional Dynamic Analysis Code for Boiling Water Reactors" (Reference 4.9) and CENPD-292-P-A, "BISON – One Dimensional Dynamic Analysis Code for Boiling Water Reactors: Supplement 1 to Code Description and Qualification" (Reference 4.10). [

]a,c

For the fast transient analyses, [

]a,c

In addition, the fast transient analysis can be performed [

]a,c

4.4.3 Control Rod Drop Accident (CRDA) Analysis

The control rod drop accident uses gap heat transfer coefficients based on a built-in best-estimate STAV7.2 model in POLCA-T. The STAV7.2 model works dynamically and calculates in each time step the gap heat transfer coefficient based on the actual fuel rod conditions (burnup, burnup history, pressure, power, pellet temperature, etc.).

[

]a,c

[

]a,c

The CRDA methodology, including treatment of uncertainties, are described in Appendix A of WCAP-16747-P-A," POLCA-T: System Analysis Code with Three-Dimensional Core Model" (Reference 4.1).

4.4.4 LOCA Analysis

The Westinghouse BWR Appendix K LOCA analysis methodology is described in References 3.1 through 3.6 and 4.0 (RPB 90-93-P-A, RPB 90-94-P-A, CENPD-283-P-A, CENPD-293-P-A, WCAP-15682-P-A, WCAP-16078-P-A and WCAP-16865-P-A). Gap heat transfer coefficients input to the LOCA calculations are based on STAV7.2 calculations.

Section 4.4.4 of Reference 1.0 (*WCAP-15942-P-A)* describes how the initial fuel rod gap conditions for the LOCA Analysis are generated using the STAV7.2 code. The SVEA-96 Optima2 fuel design was used to describe the methodology. The inputs to STAV7.2 are selected to assure that the gap heat transfer coefficient will be conservatively small to ensure the 10CFR50 Appendix K requirement I.A1 is met:

The methodology described in Section 4.4.4 of Reference 1.0 is used without modification for SVEA-96 Optima3 applications. The calculation to generate initial conditions for the CHACHA-3D fuel rod heat-up calculations uses [

]a,c

4.4.5 Stability Analysis

The nominal [

]a,c in accordance with the Safety Evaluation Reports (SERs) for Reference 4.11.

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,* *"Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors" (Reference 4.13) as required by Section 15.6.5 of Reference 1.4. The calculation of doses due to a hypothetical Fuel Handling Accident incorporates the appropriate conservative assumptions outlined in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (Reference 4.12) as required by Section 15.7.4 of Reference 1.4.*

4.5 APPLICABILITY OF THE LOCA METHODS AND METHODOLOGY

The purpose of this section is to describe the Westinghouse LOCA methodology and the effect of implementing the SVEA-96 Optima3 fuel design on the methodology.

4.5.1 LOCA Methodology

The Westinghouse Emergency Core Cooling System (ECCS) evaluation methodology is implemented using the two computer codes shown below in Figure 4.5.1-1.

The GOBLIN code is used to determine the thermal-hydraulic response of the reactor system to the postulated large- and small-break LOCAs. The calculations include interactions between the reactor system and the various safety systems. The results of the calculation are the thermal-hydraulic response of the hot assembly and the sequence of key events. This calculation may be done in one or two steps. In the one-step process, the hot assembly is modeled as a channel parallel to the average core. In the two-step process, the hot assembly is modeled as a stand-alone channel using the DRAGON option in GOBLIN where boundary conditions are provided by the single-channel average core analysis.

The CHACHA-3D code determines the detailed temperature distribution and cladding oxidation at a selected axial cross section of the hot assembly analyzed by GOBLIN. The results of the calculation are peak cladding temperature, local maximum oxidation, core wide oxidation and maximum average planar linear heat generation rate (MAPLHGR) operating limits for each new fuel design.

The inputs to the computer codes and the flow of information between the two codes are also presented in Figure 4.5.1-1.

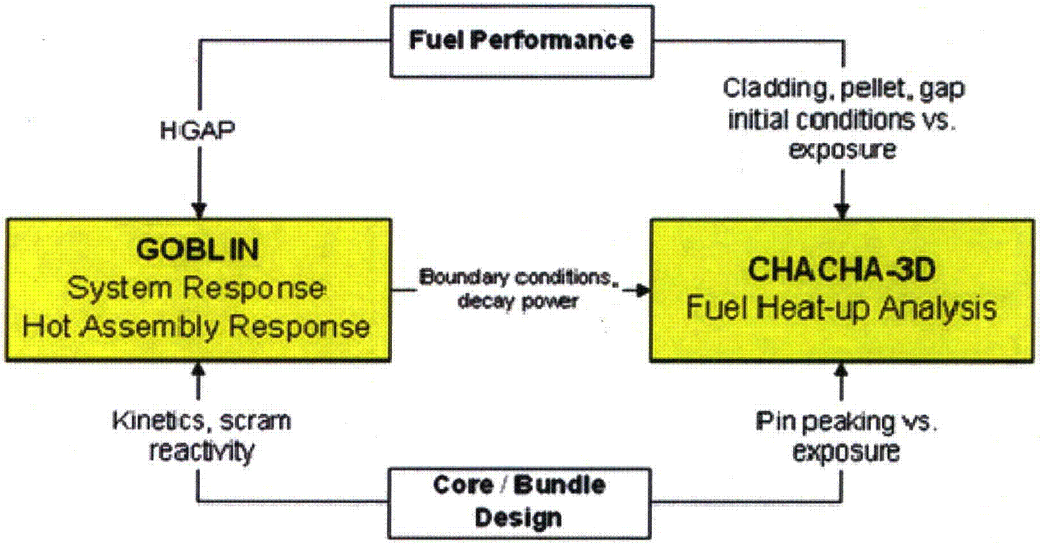


Figure 4.5.1-1 Flow of Information Between Codes

The Westinghouse Appendix K methodology was first approved by the NRC in 1989. Several supplements to the original topical report have been submitted as described in References 1.0-1.1, 3.1-3.6, and 4.0. The application of the LOCA methodology was extended to the SVEA-96 Optima2 fuel design in WCAP-16078-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel" (Reference 3.6).

4.5.2 Comparison of SVEA-96 Optima3 to SVEA-96 Optima2 Fuel

Table 4.5.2-1 compares the design parameters for SVEA-96 Optima2 and Optima3. As shown, the two designs are very similar. The design change with the largest impact on LOCA is the increased plenum volume for SVEA-96 Optima3. The larger plenum volume will reduce the rate of pressure increase as the gas temperature increases during the LOCA transient. Assuming the initial fill gas pressure is the same, this will delay cladding rupture. The other design differences have a negligible impact on the response to the LOCA event. Except for the CPR correlation, these differences are easily accommodated by code input changes. The CPR correlation requires a modification to the GOBLIN code.

Table 4.5.2-1 Comparison of SVEA-96 Optima3 to SVEA-96 Optima2		
Quantity	**Optima2**	**Optima3**

a,c

4.5.3 Evaluation Model Changes

4.5.3.1 Nodalization

Methodology

The number of axial nodes used to represent the active fuel rods in the GOBLIN code for SVEA-96 Optima2 fuel as described in Section 5.1.2 of WCAP-16078-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel" (Reference 3.6) were found to adequately predict the initial boiling transition during a LOCA. Due to the very similar design of the SVEA-96 Optima3, at least as many number of axial nodes will be used for LOCA analysis.

The number of nodes used to represent the lattice cross section in CHACHA-3D is not changed and remains the same as described in Section 4.3.1 of CENPD-283-P-A, "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity" (Reference 3.3).

Discussion

The GOBLIN code is used to determine the system and hot assembly responses to a LOCA transient. The hot assembly responses are provided to the downstream heat-up calculations by CHACHA-3D. The GOBLIN nodalization for the SVEA-96 Optima2 fuel design is described in References 3.6.

The fuel design has 3 zones. The first zone starts at the bottom of the active fuel and ends at the top of active fuel of the 1/3rd length rods. The second zone starts at the top of active fuel of the 1/3rd length rods and ends at the top of active fuel of the 2/3rd length rods. The third zone starts at the top of active fuel of the 2/3rd length rods and ends at the top of active fuel of the full-length rods. Consistent with Reference 3.6, at least [ ]^{a,c}. Since the SVEA-96 Optima3 fuel design is very similar to SVEA-96 Optima2 fuel design as shown in Table 4.5.2-1 above, the GOBLIN noding remains adequate.

The standard CHACHA-3D noding is described in Section 4.3.1 of Reference 3.2. Sensitivity studies performed there and in Section 6.3.1 of Reference 3.3 show little sensitivity to fuel rod noding. Since the SVEA-96 Optima3 fuel design is very similar to other SVEA-96 fuel designs, the standard CHACHA noding remains adequate.

4.5.3.2 CPR Correlation for SVEA-96 Optima3 Fuel

Methodology

The fuel-specific critical power ratio (CPR) correlation for SVEA-96 Optima3 is the D5 correlation, which is described in Reference 4.5. The use of GOBLIN for licensing calculations involving SVEA-96 Optima3 fuel requires that the D5 licensing topical report (LTR) be reviewed and approved by NRC, and that the approved CPR correlation be installed in the GOBLIN code.

Discussion

Fuel-specific CPR correlations are part of the heat transfer model in GOBLIN. The fuel-specific CPR correlation and the pool boiling critical heat flux (CHF) correlation are used to determine the transition between non-dryout and post-dryout heat transfer during a LOCA event. The transition CHF is determined conservatively by [ ]^{a,c}.

The fuel-specific CPR correlation is also used to establish the initial power of the hot assembly by establishing a conservative operating limit minimum CPR (OLMCPR).

4.5.4 Justification of Existing Models

4.5.4.1 Spray Heat Transfer Model

Methodology

The convective spray heat transfer coefficients described in Section 6.1. of WCAP-16078-P-A, "Westinghouse BWR ECCS Evaluation Model Updates: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel" (Reference 3.6) are applied without modification to analyses determining the hot plane heat-up response for a reactor containing SVEA96 Optima3 fuel. These heat transfer coefficients are based on the values prescribed by 10CFR50 Appendix K as described in Section 7.2 of Reference 3.3. As shown in Table 4.5.2-1 the SVEA-96 Optima3 geometry is essentially identical to SVEA-96 Optima2. The spray heat transfer coefficients are given in Table 4.5.4-1.

Table 4.5.4-1 SVEA-96 Optima3 Spray Cooling Heat Transfer Coefficients (W/m^2-°C)

Corner Rods	Side Rods	Inner Rods	Channel
[			]

a,c

Discussion

Section 6.1 of Reference 3.6 and the response to Question 27 of Reference 3.6 present the basis for the spray heat transfer coefficients used for SVEA-96 Optima2 applications. The spray heat transfer coefficients are based on spray cooling tests that simulated a 7x7 array (the BWR FLECHT program). Since the SVEA-96 Optima3 geometrical design is nearly identical to the SVEA-96 Optima2 design (see Table 4.5.2-1), the same spray heat transfer coefficients apply to the SVEA-96 Optima3 fuel design.

4.5.4.2 Radiation Heat Transfer Model

Methodology

Thermal radiation is an important phenomenon for LOCA transients that result in sustained uncovery of the core when core cooling is due only to convective spray heat transfer and radiation. The limiting fuel rod location for LOCA transients is located in the [ ]^{a,c} of the 5x5-1 subbundle when radiation is important. The thermal radiation model in CHACHA-3D, which is described in RPB 90-93-P-A, "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification" (Reference 3.1), is applied to the SVEA-96 Optima3 fuel design without change. However, the model [

]^{a,c}

Discussion

The radiation model in CHACHA-3D determines the gray body factors throughout the transient accounting for the fuel rod dimensional changes due to cladding deformation. As shown in Figure 4.5.4-1 below, [

]a,c

a,c

Figure 4.5.4-1 SVEA-96 Optima3 Sub Assembly Cross Section (see Table 4.5.2-1)

4.5.4.3 Counter-Current Flow Limit (CCFL)

Methodology

The CCFL methodology described in Section 5.4.2 of Reference 3.6 is applied without modification to the SVEA-96 Optima3 fuel design.

Discussion

The CCFL model in GOBLIN, which is described in Section 3.3 of Reference 3.1, was modified slightly in Section 5.4.2 of Reference 3.6 where it was approved for SVEA-96 Optima2 applications. The correlation constants are expressed in terms of basic geometrical parameters, which enable the correlation to be applied to a variety of geometries, provided there isn't a significant departure from the experimental database. Since the SVEA-96 Optima2 and SVEA-96 Optima3 fuel designs are nearly identical geometrically, the CCFL model continues to be applicable to the SVEA-96 Optima3 fuel design.

4.5.4.4 Transition Core Evaluation

Methodology

[

]a,c

If this simplification is not justified, the mixed core model will be used for the system response analysis to determine the MAPLHGR limits for the Westinghouse fuel. In this case, Westinghouse will inform the utility of the results so that the other fuel vendor may assess the impact of the transition core on its MAPLHGR limits.

Discussion

When a utility changes the fuel vendor, the reactor core will be loaded with different types of fuel bundles. These reload cycles are referred to as mixed, or transition cores. The presences of non-Westinghouse fuel challenges the usual LOCA analysis approach [

]a,c

4.5.5 Conclusion

Westinghouse Appendix K methodology was extended to the SVEA-96 Optima2 fuel design in Reference 3.6. As discussed in Section 4.5.3 and 4.5.4 above applicability of the LOCA methodology to the SVEA-96 Optima3 fuel design is justified due to the similarities of the fuel designs. A fuel-specific CPR correlation for SVEA-96 Optima3 (D5) will be implemented after NRC acceptance and at least as many number of axial nodes will be used for LOCA analysis of a SVEA-96 Optima3.

5 TECHNICAL DATA

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, Style 1 provides typical data for BWR/4, *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 and BOL.

5.1 FUEL RODS

5.1.1 Pellets

5.1.1.1 Pellet Dimensions

	UO_2 and Gadolinia Pellets		
	Units	***Nominal Value***	***Note***

a,c

5.1.1.2 Pellet Data

	Units	*Nominal Value*

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

2. [

]a,c

5.1.2 Fuel Rod Cladding

5.1.2.1 Cladding Dimensions

a,c

	Units	*Nominal Value*	*Note*

5.1.2.2 Cladding Chemical and Physical Properties

[

]a,c

5.1.3 Fuel Rod Length

	Units	Nominal Value

a,c

5.1.4 Fuel Rod Miscellaneous Data

	Units	Nominal Value

a,c

(cont.)	Units	Nominal Value

a,c

(cont.)	Units	Nominal Value

a,c

5.1.5 Fuel Rod Materials

a,c

5.1.6 Typical Fuel Rod Weights

	Units	Nominal Value

a,c

(cont.)	Units	Nominal Value

a,c

5.1.7 Spacer Grid

	Units	Nominal Value

a,c

5.2 FUEL ASSEMBLY DATA

5.2.1 Fuel Assembly Miscellaneous Data

	Units	Nominal Value

a,c

a,c

(cont.)	Units	Nominal Value

5.2.2 Fuel Assembly Materials

Component	*Material*

a,c

5.2.3 Typical Fuel Assembly Weights

	Units	Nominal Value

a,c

6 CODE DESCRIPTION

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

STAV7.2 is the primary fuel performance analysis tool. 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 and Optima3 fuel assembly. ANSYS will additionally be used for fuel rod cladding stress analysis for SVEA-96 Optima3 and in future applications. The ANSYS code is well known in Europe and the U.S., and has been broadly used for reactor design and analysis applications within the nuclear industry.

Cladding stress analysis was performed using VIK-3 for the previous fuel designs. The code has become obsolete and difficult to maintain so Westinghouse has decided to replace the code with the commercial ANSYS code as described in section 4.3.3.

6.1.1 STAV7.2

The STAV7.2 code is used by Westinghouse in Europe for both 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 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.

Details of the STAV7.2 code description are presented in Section 2 of Reference 1.2.

6.1.2 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.*

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

6.1.3 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 several 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 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 water cross 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 water cross 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

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

[

]a,c

The next generation fuel is SVEA-96 Optima3 [

]a,c The top tie plate is replaced with a top spacer and all fuel rods except tie rods rest freely on the bottom tie plate. The spacer in SVEA-96 Optima3 is of a sleeve type design with a four line contact compared to the six point contact in SVEA-96 Optima2.

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 Optima3 design.

The experience base is steadily increasing and as of January 2013, [

]a,c. As of January 2013, Westinghouse has contracts for the delivery of [ ]a,c.

7.2 EXPERIENCE

A complete summary of Westinghouse 10x10 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 [ ]a,c 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 Olkiluoto 2. Since 1986, SVEA-64 fuel assemblies have been loaded into the German reactors Krümmel, Philippsburg 1, Brunsbüttel, and the Swiss reactor Leibstadt. In total [ ]a,c 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, [ ]a,c 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. [ ]a,c reloads have currently been delivered to these two European GE reactors.

In 1990 and 1991 [ ]a,c SVEA-96 Lead Fuel Assemblies were installed in four U.S. GE BWR reactors. The first [ ]a,c U.S. LFAs were loaded in Columbia Generating Station in 1990, to be followed by [ ]a,c LFAs in Fermi2, Peach Bottom 2 and Limerick 2 (all BWR/4s) the following year. In addition, Susquehanna received [ ]a,c SVEA-96+ LFAs 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 January 2013, [

]a,c

SVEA-96 Optima2 LFAs were inserted into Forsmark 2 and Leibstadt reactors in 2000 and reload quantities were delivered in 2002. In January 2013, [
]a,c assemblies have been delivered. The leading SVEA-96 Optima2 assembly has reached an assembly average burnup of [ ]a,c. The SVEA-96 Optima2 design has been delivered in reload quantities to Quad Cities and Dresden units and as of January 2013, [ ]a,c reloads have been delivered in the U.S.

7.2.2.4 SVEA-96 Optima3

SVEA-96 Optima3 LFAs were inserted in [ ]a,c.

As of January 2013, [ ]a,c have been delivered. SVEA-96 Optima3 assemblies have been inserted in [ ]a,c.

7.3 FUEL RELIABILITY

7.3.1 General

Primary fuel leakers in SVEA 10x10 fuel is shown in Figure 7-3. Debris fretting has been identified as the only cause of primary failure in all fuel inspected equipped with modern **LK3/L™** cladding material with liner. To mitigate this failure mode, Westinghouse developed the **TripleWave™** and **TripleWave+™** debris filters described in Section 2.3 and the spacer with sleeve type design described in Section 2.2.2. These improvements are 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

[

]^{a,c}. *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:

[

]^{a,c}

Westinghouse 10x10 fuel performance for lined fuel during the period 2000 through 2012 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

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

SVEA-96 assemblies in [ ]a,c

SVEA-96 Optima LFAs have been inspected [ ]a,c

Table 7-1 SVEA 10x10 Fuel Deliveries

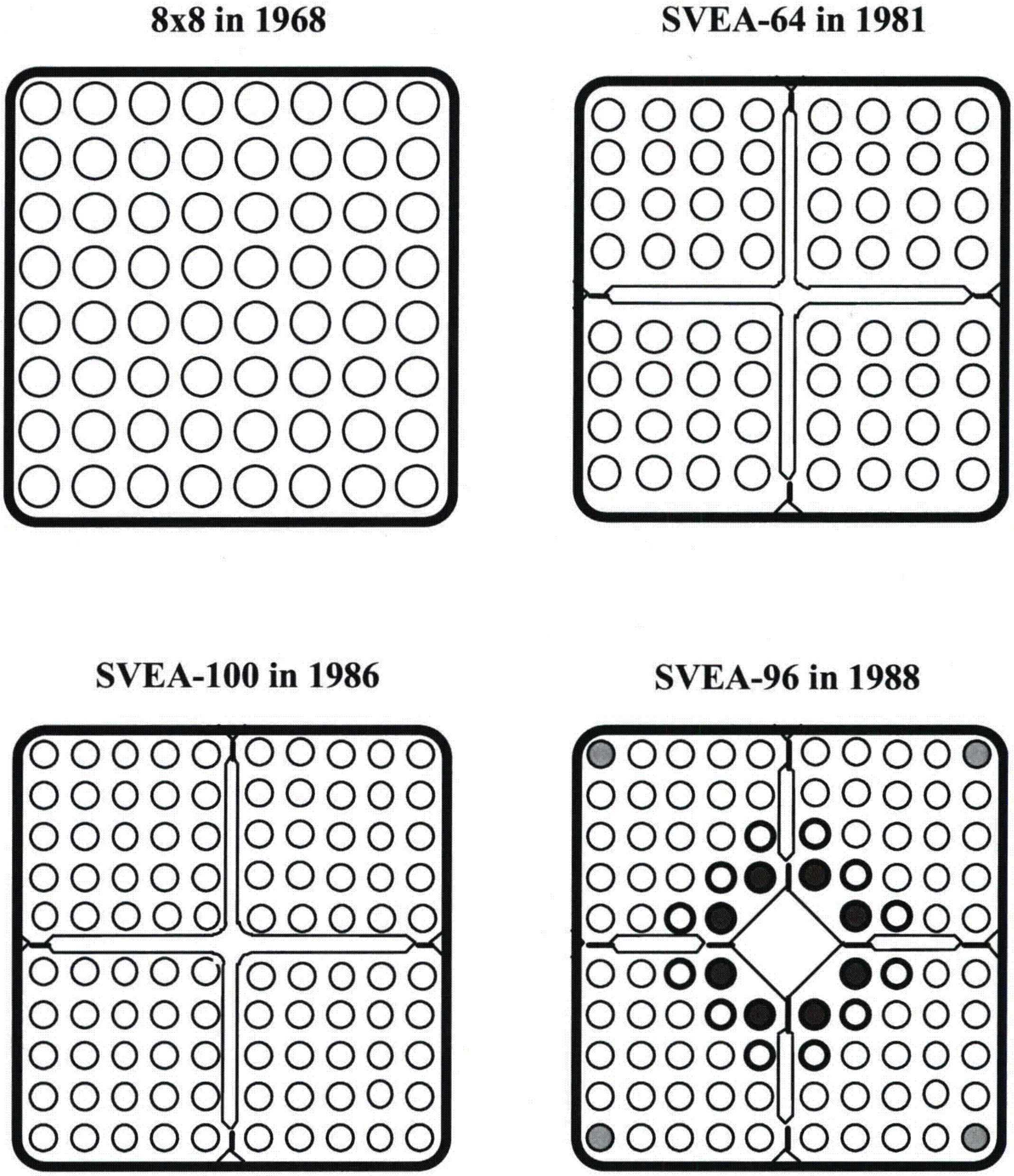
Plant	SVEA-96/96+/100	Optima	Optima2	Optima3	Total

a,c

Table 7-1 SVEA 10x10 Fuel Deliveries (cont.)

Plant	SVEA-96/96+/100	Optima	Optima2	Optima3	Total

a,c



Note: The SVEA-96 design shown above contains 96 full-length rods with the same diameter in SVEA-96 and SVEA-96+.The part-length rods for SVEA-96 Optima, Optima2, and Optima3 are identified as follows.

- 2/3rd Part-length Rods in SVEA-96 Optima, Optima2 and Optima3
- 1/3rd Part-length Rods in SVEA-96 Optima2 and Optima3
- Increased diameter (10.3 mm) in SVEA-96 Optima

Figure 7-1 SVEA Fuel Designs

a,c

Figure 7-2 Burnup Statistics as of December 2012

a,c

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

a,c

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

8 PROTOTYPE TESTING

Westinghouse has a continuing program to perform prototype testing for all of their fuel assembly designs. Tests have been performed on the Westinghouse 8x8 design, the SVEA-64 design, the SVEA-100 design, the SVEA 96/96+ design, the SVEA-96 Optima design, the SVEA 96 Optima2 design and the SVEA-96 Optima3 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 Optima3 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 Optima3 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 Optima3 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 spacers were adjusted to [

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

Nominal SVEA-96 Optima3 Test, Operating Parameters			
		2-Phase Test	1-Phase Test

a,c

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

]a,c

The conclusion from the fretting tests is that the mechanical behavior of the SVEA-96 Optima3 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 to verify its ability to withstand load following during reactor operation. The test is [ ]a,c.

The test was performed [ ]a,c

8.3 LATERAL LOAD CYCLING TEST, CHANNEL AND SPACER

Lateral load cycling tests have been performed with low-cycle fatigue tests with the purpose of qualifying spacers and channel 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

A similar test has also been performed with a [ ]a,c

[

]a,c

8.4 SPACER CAPTURE TEST

The spacer capture function has been tested on SVEA-96 Optima3 tie rods and spacers.

Test of spacer capture force with SVEA-96 Optima3 spacers shows [

]a,c

Shear tests show [

]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/SVEA-96 Optima2/SVEA-96 Optima3 *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

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}

– [

]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 Final Safety Analysis Report, (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

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 Optima3. *Furthermore, Westinghouse has performed examinations of various SVEA-96 fuel types in Westinghouse Nordic plants, Siemens plants in Germany, GE plants in Switzerland,* Spain *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, Finland, Germany and Switzerland.

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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