ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel

ABB Combustion Engineering Nuclear Operations

ABB

ABB Seismic/LOCA Evaluation Methodology
for Boiling Water Fuel

May 1994

ABB Combustion Engineering Nuclear Operations

ABB

TABLE OF CONTENTS

ABB

TABLE OF CONTENTS (Continued)

ABB

TABLE OF CONTENTS (Continued)

ABB

1   INTRODUCTION

This Topical Report presents the design bases, acceptance criteria, and methodology used to evaluate the ABB BWR fuel assembly when subjected to postulated seismic and Loss of Coolant Accident (LOCA) events. This report, together with References 1 and 2 which present the ABB BWR fuel assembly and fuel rod mechanical design methodologies, demonstrate compliance with the requirements of the U.S. NRC Standard Review Plan, Section 4.2 (Reference 3).

Evaluation of the structural response of fuel assemblies under the most limiting postulated external forces is part of the overall design process. Earthquakes and postulated pipe breaks in the reactor coolant system can result in external forces on the fuel assembly. The postulated most limiting event is a Safe Shutdown Earthquake (SSE) in conjunction with structural and hydraulic loads from the worst LOCA. The fuel assembly design acceptance criteria under these low probability postulated limiting events, are chosen to ensure that there is no fuel rod damage, loss of fuel coolability, or interference with control rod insertion.

The effect of a new fuel assembly design on the mechanical design bases of the reactor internals is also addressed. The evaluation includes the effect of both a mixed core and an equilibrium core on the reactor internals.

2   SUMMARY AND CONCLUSIONS

This topical report describes the general ABB methodology which demonstrates that the ABB BWR reload fuel assembly satisfies the following design bases under a postulated Seismic/LOCA event:

(1) Fuel rod fragmentation will not occur as a result of combined normal operation, seismic, and LOCA loads.

(2) Control rod insertability will not be impaired.

(3) Spacer grid distortion will not be sufficiently great that fuel rod coolability would be prevented.

Evaluation of the fuel assembly design subjected to external forces consists of three parts. First, the postulated maximum forces which could be exerted on the fuel assembly are determined. Then the fuel assembly structural response to such forces is evaluated. Finally, fuel assembly response is evaluated against specific acceptance criteria that ensure compliance with the required design bases:

(1) Fuel assembly channel, fuel rod, and other components integrity is maintained.

(2) The fuel assembly is not dislodged from the lower support structure and fuel channel deflections do not impair control rod insertion.

(3) Spacer grid integrity is maintained.

Figure 2.1 summarizes the fuel assembly evaluation process.

In addition, it is demonstrated that if the reload fuel assembly is similar in weight, dimension, and dynamic properties to the resident fuel assemblies, introducing a new fuel assembly design will not significantly affect the mechanical design bases of the other reactor internals components.

Three examples apply the general evaluation methodology to the SVEA-96 fuel assembly in General Electric built BWRs.

Application of the generic methodology described in this report ensures that the Seismic/LOCA event fuel assembly design bases summarized above are not violated for current and future BWR fuel designs.

A Seismic/LOCA evaluation is performed for each plant application of ABB BWR fuel. The methodology is defined in a clear and generalized format that can be applied:

- To both ABB and non-ABB designed BWR fuel,

- In all BWR reactor designs (e.g. BWR/2 through BWR/6), and

- Accommodating a variety of plant licensing bases and available seismic and LOCA data.

Figure 2.1 has been deleted

Combustion Engineering, Inc. Proprietary Information

Figure 2.1 Fuel Assembly Seismic/LOCA Evaluation Process

ABB

3   DESIGN BASES

3.1   Design Base Event

The nuclear fuel assembly is classified as a Seismic Category I component. To ensure compliance with the requirements of U.S. NRC Standard Review Plan, Section 4.2 (Reference 3), the fuel assembly is designed to withstand a Safe Shutdown Earthquake (SSE) in conjunction with structural and hydraulic loads from the worst LOCA. The postulated design base SSE and LOCA events are described in the following subsections.

3.1.1   Seismic Event

A Safe Shutdown Earthquake (SSE) is that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material (Reference 4). It is that earthquake which produces the maximum vibratory ground motion for which Seismic Category I structures, systems, and components are designed to remain functional. Seismic Category I structures, systems, and components are those necessary to ensure:

(1) The integrity of the reactor coolant pressure boundary, or

(2) The capability to shutdown the reactor and maintain it in a safe shutdown condition, or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposure of 10 CFR Part 100,

All Seismic Category I structures, systems, and components are designed to remain functional during a safe shutdown earthquake. Stress limits in excess of yield are allowed provided safety functions are maintained.

3.1.2   LOCA Event

The Loss of Coolant Accident (LOCA) is a postulated accident, prescribed in the Code of Federal Regulations Title 10 Part 50.46, to determine the design acceptance criteria for the plant Emergency Core Cooling System (ECCS). The postulated LOCA consists of a pipe break in the reactor coolant pressure boundary which exceeds the capability of the reactor coolant makeup system. The pipe breaks to be considered encompasses all sizes and locations up to and including a double-ended rupture of the largest reactor coolant system pipe.

The LOCA is also a limiting postulated event for hydraulic loading on reactor internals. This is because of the rapid internal pressure

ABB

differences that are generated in a LOCA. Typically a large steam space break creates the largest hydraulic loads on the fuel assembly. The depressurization forces from the LOCA event also contribute to the structural forces exerted on the fuel assembly. The LOCA event that produces the maximum hydraulic loads on the fuel is generally considered the Seismic/LOCA design base LOCA event.

The LOCA event also produces core support plate motion in addition to the seismic motion. Other postulated events that do not produce limiting hydraulic loads, but produce more severe core plate motions may be conservatively used for structural LOCA loads on the fuel assembly. Examples of these events are Safety/Relief valve opening, feedwater line break, or recirculation line break. Using limiting LOCA loads from different events is conservative and simplifies the evaluation.

3.2   Fuel Assembly Design Bases

Design bases are chosen such that compliance with the bases will assure that the mechanical acceptance criteria and guidelines in Appendix A of Section 4.2 of the U.S. NRC Standard Review Plan are satisfied. The following design criteria apply with respect to combined seismic plus LOCA loads as defined in Section 3.1:

(1) Fuel rod fragmentation will not occur as a result of combined seismic and LOCA loads.

This design base ensures that there is no radioactive release from the fuel rod. It is satisfied by the acceptance criteria that fuel channel, fuel rod, and other fuel assembly component stresses are within acceptable limits.

(2) Control rod insertability must not be impaired.

These design bases ensure that the reactor can be safely shutdown following the event. It is satisfied by the acceptance criteria demonstrating that the fuel assembly will not move or deform to a position that blocks control rod motion.

(3) Fuel rod coolability must be maintained.

These design bases ensure that the fuel rod remains intact after the event and no subsequent radioactive release from the fuel rod occurs. It is satisfied by demonstrating that large distortion or failure of the spacer grids does not occur.

ABB

4 ACCEPTANCE CRITERIA

A set of specific acceptance criteria are established to demonstrate that the design bases given in Section 3 are satisfied. Results from the evaluations performed using the methodology described in Section 5 are compared to the acceptance criteria given herein.

4.1 Material and Component Stresses

To demonstrate compliance with the first and third fuel assembly design bases, fuel assembly component (i.e., channel, rod, tie plates, spacer grids) stresses and forces resulting from combined SSE and LOCA loads are evaluated against a set of material and component acceptance criteria. Either the component maximum stresses are analytically calculated and evaluated against the material limiting allowable stress intensities, or the component maximum forces are calculated and evaluated against experimentally based acceptable external forces.

The fuel assembly is classified as a Seismic Category I component. Since the Level D Service Limits apply to the SSE and LOCA events, the material stress criteria are chosen in accordance with ASME Section III, Appendix F (Reference 5). They are:

Stress Category	Allowable Stress Intensity	
P_m	min of:	$2.4\,S_m$ $0.7\,S_u$
P_L	min of:	$3.6\,S_m$ $1.05\,S_u$
$P_L + P_b$	min of:	$3.6\,S_m$ $1.05\,S_u$

where

P_m = general primary membrane stress intensity

P_L = local primary membrane stress intensity

P_b = primary bending stress intensity

S_m = allowable stress intensity according to ASME Section II (Reference 6)

S_u = ultimate strength at temperature

Stress intensities, S, to be compared with the allowable stress intensity are calculated according to:

S = Maximum($|\sigma_1 - \sigma_2|, |\sigma_1 - \sigma_3|, |\sigma_2 - \sigma_3|$) where the σ_i are the principal stresses

For parts of the fuel assembly where maximum stresses are not calculated analytically, the maximum loads exerted on the part are compared to experimentally determined acceptable loads. The limiting loads are generally expressed as an acceleration, displacement, or force exerted on the part.

4.2   Control Blade Insertability

The second fuel assembly design basis requires control blade insertability to ensure safe shutdown of the plant following a combined Seismic/LOCA event. Control blade insertability is ensured provided:

(1) The fuel assembly is not displaced from the fuel support piece in a manner which would prevent control rod insertion, and

(2) The channel does not deform to the point of preventing control rod insertion.

4.2.1   Assembly Lift

The BWR fuel assembly rests on the lower core support piece by its own weight, with no mechanical "hold-down" mechanisms. The fuel assembly must remain engaged in the lower support structure following a postulated Seismic/LOCA event, to ensure unimpeded insertion of the control rods.

Hence the assembly lift design acceptance criterion is:

(1) The fuel assembly does not rise off the lower support structure,

or, if the assembly does lift

(2) The assembly distance of travel is not large enough to dislodge the fuel from its normal position on the lower support structure.

4.2.2   Channel Deformation

It must be demonstrated that fuel channel deflection into the path of the control blade motion does not restrict control rod insertion.

The channel deformation design acceptance criteria are:

(1) The maximum stresses calculated analytically do not exceed the allowable stress intensities and therefore large channel deformation and buckling are precluded.

ABB

(2) The maximum deflection of the channel will be compared to values for which safe insertion of control rods can be demonstrated.

4.3 Effect of Reload Fuel on the Reactor Internals

The minor changes in core characteristics due to the introduction of a new reload fuel assembly design should not invalidate the design base analyses for other reactor internal components.

The acceptance criteria for the effect of the reload fuel on the reactor internals is that core configuration(s) with the reload fuel have sufficiently similar weight, dimensions, and dynamic characteristics to that in the plant licensing base analysis. Confirming similar characteristics demonstrates that the conclusions of the design basis analyses for the reactor internal components are not invalidated.

ABB

5   GENERAL EVALUATION METHODOLOGY

The general methodology used to evaluate a BWR fuel assembly mechanical integrity and its effect on the reactor internals including control rods, when subjected to a postulated seismic and LOCA event, is described in this section. Where appropriate a general discussion of the methodology is included. Specific applications which illustrate this general methodology are presented in Section 6.

The fuel assembly evaluation process consists of the following steps:

(1) A fuel assembly structural model is developed for use in the response analyses. The model development process is validated by comparison to relevant fuel assembly test data.

(2) The structural response of the fuel assembly to the seismic event is determined in both the horizontal and vertical directions.

(3) The structural response of the fuel assembly to the LOCA event is determined.

(4) Each component of the fuel assembly is evaluated for the combined normal operating, SSE and LOCA loads against the acceptance criteria given in Section 4.1.

(5) Fuel assembly lift and channel deflections are evaluated against the control rod insertability acceptance criteria in Section 4.2.

(6) The effect of the reload fuel assemblies on the other reactor internals is evaluated against the acceptance criteria in Section 4.3.

The complete fuel assembly evaluation process is shown in detail in Figure 5.1. [Proprietary Information Deleted]

5.1   Fuel Assembly Integrity Evaluation

A fuel assembly is evaluated for both seismic and LOCA transient loads. Seismic loads are applied to the fuel by motion of the core supports, while LOCA loads are applied by both motion of the core supports and hydraulic pressure forces.

[Proprietary Information Deleted]

Section 5.1.1 provides the methodology used to evaluate a fuel assembly for core plate motion, which includes motion from both seismic and LOCA. Evaluation of the assembly for the hydraulic forces due to LOCA is presented in Section 5.1.2.

ABB

5.1.1   Fuel Assembly Response to Core Support Motion

Fuel assemblies are subjected to core support motion in both the horizontal and vertical directions. Fuel assembly response in these two directions is evaluated separately, because the vertical frequencies of the pressure vessel, internals, and fuel are well above the significant horizontal frequencies. In addition, due to symmetry of the pressure vessel and internals, coupling between horizontal and vertical motion is negligible. The methodology for fuel assembly response to horizontal core support motion is discussed in Sections 5.1.1.1 through 5.1.1.3, while response to vertical motion is discussed in Section 5.1.1.4.

5.1.1.1   Fuel Assembly Model Development and Testing

Methodology

The dynamic analysis model of the BWR fuel assembly will contain the detail necessary to accurately predict response to seismic and LOCA core plate motion. Fuel assembly model development is supported by a series of fuel bundle and fuel assembly tests. These tests confirm that the static and dynamic characteristics of the assembly and its components are accurately modeled.

Discussion

Typical models used for fuel analysis are shown in Figures 5.2 and 5.3. [Proprietary Information Deleted]

[Proprietary Information Deleted] The model/test correlation confirms that the models shown in Figures 5.2 and 5.3 can accurately predict fuel assembly response. These models are discussed in more detail in examples of Section 6.

5.1.1.2   Vessel, Internals, and Fuel Horizontal Seismic Response

Methodology

[Proprietary Information Deleted]

Discussion

[Proprietary Information Deleted]

5.1.1.3   Fuel Assembly Horizontal Seismic Response

Methodology

[Proprietary Information Deleted]

ABB

Discussion

|[Proprietary Information Deleted]

5.1.1.4   **Fuel Assembly Vertical Seismic Response**

Methodology

|[Proprietary Information Deleted]

|If the fuel assembly lift evaluation shows that the fuel may lift off the core support plate, then impacting of the fuel will occur and the acceleration levels of the fuel will be significantly higher than when the fuel doesn't lift. In this case, the plant specific maximum vertical acceleration of the fuel or a bounding value is used in the evaluation.

Discussion

|[Proprietary Information Deleted]

5.1.1.5   **Fuel Assembly Response to LOCA Core Support Motion**

Methodology

|Fuel assembly response to LOCA core support motion is evaluated in one of several ways.

|[Proprietary Information Deleted]

Discussion

|As discussed in Section 5.1 above, fuel assembly loads due to a SSE are |[Proprietary Information Deleted]

5.1.2   **Hydraulic Forces from LOCA Event**

During a postulated Loss of Coolant Accident (LOCA) event, defined in Section 3.1.2, the fuel assembly is subjected to variations in hydraulic loads, caused by pressure differentials across the channel wall and along the assembly length.

|[Proprietary Information Deleted]

Fuel assembly pressure differentials also determine the vertical hydraulic lift force, contributing to the potential for assembly lift.

ABB

5.1.2.1 Thermal Hydraulic Analysis Model

Methodology

Loss of Coolant Accident analyses are performed with NRC approved ABB Emergency Core Cooling System evaluation computer codes. [Proprietary Information Deleted]

Discussion

The Loss of Coolant Accident ECCS analyses are performed, in part, with the NRC approved GOBLIN/DRAGON computer codes (Reference 11 and 12). The LOCA system response is calculated using the GOBLIN code. The fuel assembly hydraulic response is calculated using the DRAGON code. This DRAGON code uses the system response calculated by GOBLIN as boundary conditions for the individual fuel assembly response. A description of GOBLIN and DRAGON are given in Appendix A.1.

[Proprietary Information Deleted]

Example 1 and 2 presented in Section 6, each used the approved ECCS LOCA licensing model for that particular plant. For Example 1 the LOCA model included three fuel channel (hot , average, and peripheral regions) representing the core. Example 2 use the single core channel model approved by the NRC for use in the U.S. Sensitivity studies of the channel noding used in Example 1 and comparison of results for the two examples demonstrate a very large conservative margin resulting from simulating the reactor system response by a single average fuel channel. The conservatism arised from restricting upper and lower plenum communication through the core to the pressure response of a single average assembly. Simulating several channels more accurately captures the pressure communication between plenums.

5.1.2.2 Thermal Hydraulic Model Qualification

Methodology

The qualification base for the thermal-hydraulic methods and plant model shall be [Proprietary Information Deleted]

Discussion

Extensive code and modeling qualification of GOBLIN/DRAGON has been performed as part of the NRC approved ABB ECCS LOCA methodology (Reference 11 and 12). [Proprietary Information Deleted]

ABB

5.1.2.3   LOCA Event Conditions

Methodology

The design base LOCA event analyzed is the break location and initial conditions that maximize the net pressure load on the fuel assembly.

The design base LOCA is confirmed to be the main steam line break from previous plant specific design base Seismic/LOCA analyses. [Proprietary Information Deleted]

Discussion

The GOBLIN computer code is used to determine the transient hydraulic conditions within the reactor vessel following the postulated design base break. [Proprietary Information Deleted]

Break Location

A full recirculation line break is the most rapid liquid space depressurization event. A full guillotine steam line break upstream of the flow restrictor is the most rapid steam space depressurization event. Both events are considered in determining the design basis accident for the engineered safeguard features. However, the steam line break yields significantly larger differential pressures across the core than a recirculation line break. Hence for hydraulic forces exerted on the fuel assembly, the design base LOCA event is a full guillotine steam line break.

[Proprietary Information Deleted.]

Initial Plant Conditions

The maximum internal pressure loads can be considered to be composed of two parts: steady-state and transient pressure differentials (Reference 9). For a given plant, the core flow and power are the two major factors which influence the reactor internal pressure differentials. The core flow essentially affects only the steady-state part. For a fixed power, the greater the core flow, the larger will be the steady-state pressure differentials across the core.

The core power affects both the steady-state and the transient parts. As the power is decreased, there is less voiding in the core, and consequently, a lower steady-state core pressure differential. Less core voiding means a smaller steam space contributing to a faster depressurization. However, less voiding in the core also means more liquid in the core available to flash, retarding vessel depressurization. These competing effects can cause the maximum pressure loads at low powers to be limiting at certain locations within the fuel assembly and

ABB

for higher powers to be limiting at others locations. [Proprietary Information Deleted]

[Proprietary Information Deleted]

The next two sections describe the typical steam line break reactor system and hot assembly responses, respectively. Also presented are sensitivity study results of the key reactor and hot assembly initial and boundary conditions.

5.1.2.4 Reactor System Response to LOCA

Methodology

A plant or plant class specific reactor system response to the LOCA is calculated for the design base LOCA event at limiting plant initial conditions and assumed boundary conditions. [Proprietary Information Deleted]

Discussion

The GOBLIN code reactor system response is performed for a specific plant or plant class. The reload fuel design and core configuration changes do not significantly affect the initial rapid vessel blowdown. Initial plant conditions are set to bound variations from cycle to cycle.

A typical reactor pressure vessel response to a design base steam line break is shown in Figures 5.4 and 5.5. [Proprietary Information Deleted]

Several sensitivity studies were performed of the GOBLIN reactor system and average fuel assembly response. The sensitivity studies examined are summarized in Table 5.1.

The system response sensitivity studies [Proprietary Information Deleted]

5.1.2.5 Fuel Assembly Response to LOCA Pressure Loads

Methodology

The limiting fuel assembly response to the plant specific LOCA system response is calculated for the design base LOCA event. [Proprietary Information Deleted]

Discussion

The average fuel assembly response to the main steam line break transient described in Section 5.1.2.4, is shown in Figures 5.6 through 5.8. These representative results were calculated with DRAGON using lower and upper plenum boundary conditions from the GOBLIN simulation. [Proprietary Information Deleted]

5.1.3 Fuel Assembly Stress Limits

The fuel assembly integrity is maintained if the mechanical stress limits of the fuel components are not exceeded. Two types of stress limits are defined:

(1) *Material stress limits* are used in component analytical calculations (e.g., FEM analysis) that provide the maximum material stresses and stress location.

(2) *Component stress limits* are experimentally determined component loading limits which have been demonstrated not to exceed the component material stress limits.

5.1.3.1 Material Stress Limits

Methodology

The material properties, yield strength (S_y) and ultimate strength (S_u), used to evaluate the allowable stress intensity (S_m) and stress limits (given in Section 4.1), are obtained from material data bases.

Discussion

Currently ABB BWR fuel component materials are [Proprietary Information Deleted]

5.1.3.2 Component Stress Limits

Methodology

Acceptable component stress limits are determined based on measured maximum external loads that still demonstrate component integrity and functionality.

Discussion

[Proprietary Information Deleted]

ABB

5.1.4 Fuel Assembly Evaluation

Each component is evaluated for the combination of normal operation, SSE, and LOCA loads.

5.1.4.1 Channel

Methodology

[Proprietary Information Deleted]

Discussion

[Proprietary Information Deleted]

5.1.4.2 Spacer Grids

Methodology

[Proprietary Information Deleted]

Discussion

[Proprietary Information Deleted]

5.1.4.3 Fuel Rods

Methodology

[Proprietary Information Deleted]

Discussion

[Proprietary Information Deleted]

5.1.4.4 Other Components

Methodology

[Proprietary Information Deleted]

Discussion

[Proprietary Information Deleted]

5.2 Control Blade Insertability Evaluation

The ABB methodology for evaluating control blade insertability consists of demonstrating compliance with the design acceptance

ABB

criteria of Section 4.2. Specifically, the potential for fuel assembly lift and channel deformation are evaluated.

5.2.1 Fuel Assembly Lift

[Proprietary Information Deleted]

Methodology

[Proprietary Information Deleted]

Discussion

[Proprietary Information Deleted]

[Proprietary Information Deleted]

5.2.2 Channel Deformation

Methodology

Large channel deformations can have the potential of interfering with the motion of the control rod during insertion. [Proprietary Information Deleted]

Discussion

[Proprietary Information Deleted]

5.3 Reactor Internals Evaluation

Methodology

[Proprietary Information Deleted]

Discussion

[Proprietary Information Deleted]

ABB

Table 5.1 and 5.2 have been deleted

Combustion Engineering, Inc. Proprietary Information

ABB

TABLE 5.3

TYPICAL BWR REACTOR OUTER FUEL SHROUD CLEARANCE

Plant Type	All BWR/6	BWR/6 238" dia./648 ass.	BWR/5 251" dia./764 ass.
Vessel Diameter mm (inch)	5540 - 6380 (218 - 251)	6040 (238)	6380 (251)
Number Fuel Assemblies	624 - 800	648	764
Shroud Inside Diameter mm (inch)	–	5020 (197.5)	5160 (203.2)
Assembly Pitch mm (inch)	152 (6.0)	152 (6.0)	152 (6.0)
Minimum Clearance between assembly and shroud mm (inch)	> 49 (> 1.94[1])	185 (7.3)	45 (1.76)

Note 1: From Reference 13.

ABB

Table 5.4 and Figures 5.1 through 5.10 have been deleted

Combustion Engineering, Inc. Proprietary Information

ABB

6   EXAMPLE APPLICATIONS FOR SVEA-96 FUEL

This section presents some examples of application of the general methodology described in Section 5. The examples presented here are for ABB SVEA-96 fuel in General Electric built BWRs. These examples represent typical future specific applications of the methodology described in this report.

Descriptions of the SVEA-96 fuel assembly design and of fuel assembly stress limits are provided in Section 6.1. In Section 6.2, fuel assembly testing and model development is described. Then three example applications are presented. [Proprietary Information Deleted]

6.1   SVEA-96 Fuel Assembly

6.1.1   Fuel Assembly Description

The SVEA-96 fuel design discussed in these applications is shown in Figure 6.1. A cross sectional view of the past and present ABB BWR fuel assembly designs are shown in Figure 6.2.

The SVEA-96 fuel assembly analyzed in these examples consists of three basic components: The fuel bundle, fuel channel, and handle. The fuel bundle consists of 96 fuel rods arranged in four 5x5 minus 1 (5x5-1) subbundles. The channel has a cruciform internal structure with a square center channel that forms gaps for non-boiling water during normal operation. The subbundles are inserted into the channel from the top and are supported in the bottom end by a stainless steel bottom support and transition piece (bottom nozzle) bolted to the channel. This design principle has been used in various ABB BWR fuel assembly designs for many years, and eliminates the leakage flow path at the bottom end of the channel. This design feature also avoids stresses in the tie rods during normal fuel handling operations. The fuel assemble is lifted with a handle connected to the top of the channel.

The subbundles are freestanding inside the channel. There is sufficient space for subbundles growth at the top of the assembly to eliminate any burnup restrictions due to differential growth between the fuel bundles and the channel. The bottom of the transition piece, or "nose piece," seats in the fuel support piece. The top ends of the fuel assemblies are supported laterally against the adjacent assemblies through the interaction of leaf springs on two sides, and the upper core grid on the other two sides. More details of the SVEA-96 fuel assembly design are provided in Reference 2.

6.1.2   Fuel Assembly Stress Limits

The example material and component stress limits provided below are for current SVEA-96 fuel design.

6.1.2.1 Material Stress Limits

The SVEA-96 fuel assembly is comprised of Zircaloy-4 for the channels, Zircaloy-2 for fuel rods, and stainless steel for bundle and channel end pieces. The properties for these materials are based on applicable material specifications and measured data (see Table 6.1A). Table 6.1B gives the current material stress limits, at 300°C (572° for SVEA-96 fuel. [Proprietary Information Deleted]

Evaluating the stress limits of Table 6.1B for the material properties using the design acceptance criteria given in Section 4.1, yields the acceptance stress intensities shown in Table 6.1C.

6.1.2.2 Component Stress Limits

Some important fuel components have maximum stresses that are not readily determined by analytical evaluations against material stress limits. The design stress limits for these components are alternatively defined by prototypic component testing. The primary tests performed to address potential seismic loads are the lateral load cycling tests. These tests were performed in support of the SVEA-96/100 fuel design.

Lateral Load Cycling Test, Channel and Spacer Grid

Lateral load cycling tests have been performed [Proprietary Information Deleted] The tests were performed as low-cycle fatigue tests with the purpose of qualifying spacer grids and channel welds for seismic type loads. Tests have been performed for a range of different channel and bundle (spacer) designs.

[Proprietary Information Deleted]

Channel Buckling Test

Channel buckling tests with a SVEA channel have been performed to demonstrate that local buckling does not occur [Proprietary Information Deleted]

6.2 Fuel Assembly Model Development and Testing

[Proprietary Information Deleted]

6.2.1 Fuel Assembly Testing

The test programs included a series of subbundle and fuel assembly tests performed to determine static and dynamic characteristics of the fuel assembly and its components. The tests are described below:

ABB

(1)   Subbundle Lateral Stiffness Test

In this test a single subbundle was supported at the top and bottom. A lateral force was applied to a central spacer. The lateral force was increased while measuring the lateral deflection. [Proprietary Information Deleted]

(2)   Subbundle Lateral Vibration and Damping Test

Again, the single subbundle was supported at the top and bottom. The lower end of the bundle was excited laterally by a hydraulic shaker. [Proprietary Information Deleted] The natural frequencies for the first four modes of vibration are summarized in Table 6.2.

A pluck vibration test was conducted to determine the structural damping of the subbundle. The critical damping ratio was found to [Proprietary Information Deleted]

(3)   Fuel Assembly Lateral Stiffness Test

In this test, a lateral force was applied at the mid span of the channel and the applied force and deflection were monitored. [Proprietary Information Deleted]

(4)   Fuel Assembly Lateral Vibration Test

Fuel assembly natural frequencies were obtained using the frequency sweep and pluck vibration methods. [Proprietary Information Deleted]

Lateral damping values for the assemblies were determined from the pluck vibration test using the logarithmic decrement method. [Proprietary Information Deleted]

6.2.2   Fuel Assembly Model Development

Most of the lateral stiffness of a BWR assembly is provided by the channel, while the fuel bundles provide most of the mass. A simple fuel model can be developed based on the geometry of the channel and bundles, weight of the assembly, and weight of water within the channel. Fuel assembly testing is performed to confirm the structural characteristics of the actual assembly and ensure that the model accurately reflects those characteristics.

The [Proprietary Information Deleted] were used in a model/test correlation program. Finite element models were developed for the subbundle and for the fuel assembly. These models were benchmarked by comparison to the actual test data.

ABB

[Proprietary Information Deleted]

(1) Subbundle Model Development

The finite element model of the [Proprietary Information Deleted]

(2) Fuel Assembly Model Development

The fuel assembly model used to simulate the [Proprietary Information Deleted]

(3) Conclusions

[Proprietary Information Deleted]

6.3 Example 1: SVEA-96 Integrity Evaluation - Full Vessel Analysis

The structural dynamic response and integrity evaluation of the SVEA-96 fuel during a Seismic/LOCA event is presented here. The horizontal dynamic response of the SVEA-96 fuel to a Safe Shutdown Earthquake event is analyzed using a finite element model of the reactor pressure vessel (RPV) with internals. The seismic excitation is applied as an acceleration time history at the support of the RPV. [Proprietary Information Deleted] Results are demonstrated to be in compliance with the fuel assembly integrity design acceptance criteria (Section 4.1).

6.3.1 Response to Seismic Event

The horizontal excitation load is the acceleration time history of the reactor pressure vessel support. [Proprietary Information Deleted] The acceleration response spectrum for this time history is shown in Figure 6.8. The acceleration time history is input to the vessel and internals dynamic response analysis.

The vertical acceleration time history is shown in Figure 6.9 and the vertical acceleration response spectrum is shown Figure 6.10. [Proprietary Information Deleted]

6.3.1.1 Vessel, Internals, and Fuel Model

A dynamic analysis model was set up for the specific plant of interest. In this case the plant was a General Electric built BWR/6 plant. The dynamic analysis model represents the reactor pressure vessel with all internal components, including a detailed model of the fuel assemblies.

ABB

The model consists of weightless beam and spring elements, of nodal masses, and of dynamic fluid coupling elements. [Proprietary Information Deleted]

6.3.1.2 **Model Qualification**

Time Step

[Proprietary Information Deleted]

[Modal Analysis] Benchmark

[Proprietary Information Deleted] The results are shown in Table 6.4A.

[Proprietary Information Deleted]

6.3.1.3 **Vessel, Internals, and SVEA-96 Fuel Response to Seismic Event**

Results from a modal analysis with SVEA-96 fuel, a mixed core and a transient dynamic response analysis to the SSE are presented in this section. This analysis includes the fuel assembly response since a detailed model of the fuel is included with the vessel and internals dynamic model. The analysis is performed to understand the response of the fuel, hence the presentation of results focuses on nodes that impact on the fuel.

A modal analysis was performed to obtain natural frequencies and mode shapes for a full core of SVEA-96 fuel. The 10 lowest frequencies and their associated parts of the structure are listed in Table 6.4B. Mode shapes for the first 10 frequencies are shown in Figures 6.11 - 6.15.

[Proprietary Information Deleted]

The narrow water gaps between the channels in a BWR core create a very strong hydraulic coupling between the fuel assemblies, and the core can be considered to move as one unit with the stiffness determined by all assemblies in the core. When control rods are inserted they also contribute to the core stiffness. [Proprietary Information Deleted]

The acceleration response spectrum for the excitation of the RPV support is shown in Figure 6.8. [Proprietary Information Deleted]

ABB

SVEA-96 Fuel Assembly Horizontal Accelerations

Absolute accelerations on the fuel are calculated by superimposing the acceleration of the support on the calculated accelerations relative to the support. Time histories for absolute accelerations of the full SVEA core are presented in Figures 6.16 through 6.18. Top end or core grid (node 1), mid-level (node 4), and bottom end or fuel support (node 7) are shown. Maximum accelerations are given in Table 6.5.

SVEA-96 Fuel Assembly Horizontal Displacements

Displacement time histories for the full SVEA core at the top, middle, and bottom of fuel are shown in Figures 6.19 through 6.21 (nodes 1, 4, 7). The displacements are relative to the RPV support. Maximum displacements of the fuel in any direction are given in Table 6.6. [Proprietary Information Deleted]

SVEA-96 Fuel Assembly Vertical Accelerations

[Proprietary Information Deleted]

6.3.2 Response to LOCA Event

The postulated LOCA event causes fuel response due to both the motion of the core support from blowdown forces and hydraulic pressure loads directly on the fuel.

[Proprietary Information Deleted]

The hydraulic pressure load due to LOCA on the channel could exceed the normal operation load, and therefore has to be considered in combination with loads from seismic plus LOCA core plate motion. The channel wall pressure differential has a maximum at the bottom end of the assembly and decreases to zero at the top end. The pressure differential loads of main interest are at mid core elevation, where the maximum stresses from structural dynamic loads appear, and at the bottom end, where hydraulic loads are maximum.

The differential pressure along the length of the fuel assembly determines the hydraulic lift force. This force contributes to the potential of fuel assembly lift, evaluated in Section 6.3.4.1 below.

6.3.2.1 GOBLIN/DRAGON Model

The GOBLIN/DRAGON plant models developed, qualified, and used for ECCS licensing analysis for the plant in question, were used to determine the hydraulic loads on the fuel assembly. The reactor vessel GOBLIN nodalization used for the steam line break is shown in Figure 6.24. [Proprietary Information Deleted]

6.3.2.2   Vessel and Internals LOCA Response

The limiting break from the standpoint of hydraulic forces was determined to be a main steam line break. For this specific plant the limiting initial conditions are maximum power and flow (100% rate power and 106% rate flow). [Proprietary Information Deleted]

The reactor system response to the main steam line break is shown in Figures 6.25 and 6.26. [Proprietary Information Deleted]

6.3.2.3   SVEA-96 Fuel Assembly LOCA Response

The reactor system response is used as boundary conditions to a hot channel transient calculation. Calculated pressure differences across the channel wall are shown in Figure 6.27, for the bottom and at mid-elevation of the channel. A positive delta pressure means internal overpressure in the channel. [Proprietary Information Deleted]

Channel wall pressure differentials are summarized in Table 6.7. [Proprietary Information Deleted]

6.3.3   Fuel Assembly Integrity Evaluation

6.3.3.1   Channel

The maximum stresses from the seismic analysis and the LOCA pressure loading are evaluated separately and then combined with the normal operating stresses to yield the total maximum stress intensities. Stresses are evaluated at two locations, the bottom end where pressure loading is limiting and at the mid-core elevation where bending stresses due to seismic loading are limiting. These maximum stress intensities are confirmed less than the channel material stress limits.

Stresses from Seismic Displacement Loads

The fuel assembly channel bending stresses were calculated by the ANSYS code, as a function of time. The maximum bending stress occurs [Proprietary Information Deleted]

The material constants used in determining the stresses from calculated displacements are shown in Table 6.1A.

Table 6.9 summarizes the seismic horizontal and vertical stresses determined from the fuel assembly seismic response of the channel mid core and bottom elevations.

ABB

Stresses from LOCA Pressure Loads

Stresses from the LOCA pressure forces are evaluated by scaling a generic finite element analysis of the SVEA-96 channel to the pressure loads calculated in Section 6.3.2.3.

A finite element model was used for stress analysis of the SVEA-96 channel subjected to a differential pressure loading. The ANSYS code (see Appendix A.2) was used for this analysis. [Proprietary Information Deleted]

The relevant results of the general channel overpressure analysis are summarized in Table 6.8. [Proprietary Information Deleted]

The generic finite element analysis of the SVEA-96 fuel channel described above is used to determine the stresses on the channel in specific applications. Since the model is linear, there is a direct proportional relationship between the exerted pressure load and the resultant component stresses. The overpressure loads, calculated for this example in Section 6.3.2.2, are used to scale the general analysis stress results.

For the Example 1 LOCA pressures given in Table 6.7, the maximum membrane and membrane plus bending stress intensities (P_L+P_b) are determined by scaling the general result from the FEM analysis, (given in Table 6.8). [Proprietary Information Deleted] The corresponding axial principal stresses due to the Poisson effect are also shown in Table 6.7. These stresses are combined below with the stresses due to normal operation (static pressure and weight) and seismic stress contributions to yield the total channel stress intensity.

Total Channel Stress Intensity

The principal stresses summarized in Tables 6.7, 6.9, and 6.10 are combined and resulting stress intensities for this example, are shown in Tables 6.11 and 6.12. [Proprietary Information Deleted]

Principal stresses due to horizontal seismic, vertical seismic, and LOCA core plate motion are combined by the SRSS method. These are added to the stresses due to gravity, system pressure, and LOCA hydraulic pressure loads.

Comparison with Stress Limits

The Zircaloy-4 channel material stress limit [Proprietary Information Deleted]

ABB

The requirements on structural integrity for the channel is thus fulfilled.

6.3.3.2 Spacers Grids

Spacer grids are subjected to lateral dynamic loads from the rods during a seismic event. The spacers are required to withstand these loads without failure or significant distortion, as stated in the design bases (Section 3.2).

[Proprietary Information Deleted]

[Proprietary Information Deleted]   The design criterion regarding spacer stability and coolability is thus fulfilled.

6.3.3.3 Fuel Rods

[Proprietary Information Deleted]   These values are much less than the allowables, hence the fuel rod stress requirement is met by a large margin.

6.3.3.4 Other Components

Additional components of the fuel assembly are also loaded during a seismic plus LOCA event, namely top and bottom die plates and the transition piece in the bottom end of the channel. The material in these components is stainless steel and they are relatively solidly designed, resulting in large strength margins.

[Proprietary Information Deleted]

The discussion above shows that the load requirements on the stainless steel components are met with very large margins.

6.3.4 Control Blade Insertability Evaluation

6.3.4.1 Fuel Assembly Lift

The BWR fuel assembly rests on the lower core support by its own weight, with no mechanical "hold-down" mechanisms. The fuel assembly will lift off the lower support structure only if the external vertical forces acting on the assembly are greater than the assembly weight.

[Proprietary Information Deleted]

6.3.4.2 Channel Deformation

Maximum fuel channel deflection [Proprietary Information Deleted]

6.3.5   Reactor Internals Evaluation

[Proprietary Information Deleted]

6.3.6   Summary

[Proprietary Information Deleted]

6.4   Example 2: SVEA-96 Integrity Evaluation - Given Core Support Motion as an Input

This second example is a case where the acceleration response spectra of the fuel assembly support structures are known and can be applied directly to the fuel assembly without an application specific vessel analysis. The horizontal dynamic response of the SVEA-96 fuel to a combined Safe Shutdown Earthquake and LOCA event is analyzed using a finite element model of the fuel assembly. [Proprietary Information Deleted]

6.4.1   Response to Core Support Motion

[Proprietary Information Deleted]

6.4.1.1   Core Support Response Spectra

[Proprietary Information Deleted]

6.4.1.2   SVEA-96 Fuel Assembly Response

[Proprietary Information Deleted]

6.4.2   Fuel Assembly Hydraulic Forces

During a postulated LOCA event the fuel assembly is subjected to variations in hydraulic loads, caused by pressure differentials across the channel wall and over the assembly length.

The peak internal overpressure load on the channel is higher than the normal operation load, and therefore has to be considered in combination with structural loads. Channel pressure differential has its maximum in the bottom end of the assembly and decreases to zero at the top end. The pressure differential loads of interest are at the bottom end (maximum value) and at the mid-core elevation, where the maximum stresses from structural dynamic loads occur.

[Proprietary Information Deleted]

6.4.2.1   Vessel and SVEA-96 Fuel Assembly LOCA Response

System Response

|[Proprietary Information Deleted]

SVEA-96 Fuel Pressure Loads

|[Proprietary Information Deleted]

6.4.3   Fuel Assembly Evaluation

6.4.3.2   Channel

Channel stresses are evaluated at two locations, the bottom end where
the LOCA pressure loads are a maximum and at the mid-core elevation
where bending stresses due to seismic and LOCA core support motion
are a maximum.

Stresses from Seismic Displacement Loads

|[Proprietary Information Deleted]

Stresses from LOCA Pressure Loads

A finite element model was used for evaluation of SVEA-96 channel
stresses under differential pressure loading. The model and analysis
results are discussed in Section 6.3.3.1.

The maximum membrane and membrane plus bending stress
intensities were determined by scaling the results from the FEM
analysis. Table 6.20 provides a summary of the resultant stresses at
the mid-core elevation and bottom end of the channel due to LOCA
pressure loads.

Total Channel Stress Intensity

|[Proprietary Information Deleted]

Comparison with Stress Limits

|[Proprietary Information Deleted]

The requirement on structural integrity for the channel is thus
fulfilled.

6.4.3.2   Spacer Grids

|[Proprietary Information Deleted]

ABB

6.4.3.3   Fuel Rods

|[Proprietary Information Deleted]

6.4.3.4   Other Components

The top and bottom tie plates and the transition piece in the bottom end of the channel were also evaluated. These components have large stress margins.

|[Proprietary Information Deleted]

6.4.4   Control Blade Insertability Evaluation

6.4.4.1   Fuel Assembly Lift

|[Proprietary Information Deleted]

6.4.4.2   Channel Deformation

|[Proprietary Information Deleted]

6.4.5   Summary

This example has demonstrated a typical approach to seismic and LOCA qualification of the SVEA-96 fuel assembly. The plant licensing bases provided acceleration response spectra for the horizontal seismic and LOCA core plate motion. These were combined into an enveloping spectrum which was used as input to a response spectrum analysis. The analysis provided fuel assembly response deflections, accelerations, and stresses. Fuel assembly response due to LOCA hydraulic pressure loads was determined for the specific plant due to a full main steam line break. Fuel assembly stresses due to seismic and LOCA loads were combined with the stresses due to system pressure and weight.

Each fuel assembly component was considered and shown to satisfy the stress and load criteria discussed in Section 4 (see Table 6.26). Furthermore, it was shown that the reload fuel assemblies is bounded by the resident fuel lift analysis, and that channel deflection will not cause buckling of the channel or interference with control blade insertion. Thus control blade insertion is assured.

6.5   EXAMPLE 3: SVEA-96 INTEGRITY EVALUATION - GIVEN RESIDENT FUEL ASSEMBLY RESPONSE PARAMETERS

Seismic acceleration response spectra for the core support plate and the core grid provide the most convenient and direct input for seismic evaluation of the fuel. However, plant specific spectra at the fuel support locations are often not available. [Proprietary Information Deleted]

6.5.1   Response to Seismic Event

6.5.1.1   Typical Core Support Response Spectra

[Proprietary Information Deleted]

6.5.1.2   Resident Fuel Assembly Seismic Response

[Proprietary Information Deleted]

6.5.1.3   SVEA-96 Fuel Assembly Seismic Response

[Proprietary Information Deleted]

6.5.2   Summary

[Proprietary Information Deleted]

TABLE 6.1A

TYPICAL SVEA-96 MATERIAL PROPERTIES

	Zircaloy-4 (channel)	Zircaloy-2 (fuel rods)	Stainless Steel (casting)
Young's modulus	[		]
Poisson's Ratio	[		]

at 300°C (572°F)

TABLE 6.1B

SVEA-96 FUEL ASSEMBLY MATERIAL STRESS LIMITS

	Zircaloy-4 (channel)	Zircaloy-2 (fuel rods)	Stainless Steel (casting)
Yield strength S_y	[		]
Ultimate strength S_u	[		]
S_m	[		]

at 300°C (572°F)

TABLE 6.1C

SVEA-96 MATERIAL STRESS ACCEPTANCE STRESS INTENSITY

	Zircaloy-4 (channel)	Zircaloy-2 (fuel rods)	Stainless Steel (casting)
$P_m <$	[		]
$P_L + P_b <$	[		]

at 300°C (572°F)

ABB

TABLE 6.2

SUBBUNDLE NATURAL FREQUENCIES

Mode	Frequency (Hz)	
	[	]
1	[	]
2	[	]
3	[	]
4	[	]

TABLE 6.3

FUEL ASSEMBLY MODEL/TEST CORRELATION

	Test Data	Model
Stiffness	[	]
Deflection		
Spacer 2	[	]
Spacer 3	[	]
Spacer 4	[	]
Spacer 5	[	]

ABB

TABLE 6.4A

RPV, INTERNALS, AND FUEL NATURAL FREQUENCIES
WITH 8X8 FUEL
(EXAMPLE 1)

TABLE 6.4B

RPV, INTERNALS, AND FUEL NATURAL FREQUENCIES
WITH SVEA-96 FUEL
(EXAMPLE 1)

ABB

TABLE 6.5

ABSOLUTE FUEL HORIZONTAL ACCELERATIONS
(EXAMPLE 1)

Node No.	Position on fuel	Absolute acceleration	Time seconds
SVEA-96 core			
[			]
[			]
[			]
Mixed core			
[			]
[			]
[			]

TABLE 6.6

FUEL DISPLACEMENTS (EXAMPLE 1)

Node No.	Position on fuel	Displacement mm (in)	Time seconds
SVEA-96 core			
[			]
[			]
[			]
Mixed core			
[			]
[			]
[			]

ABB

TABLE 6.7

CALCULATED CHANNEL WALL LOCA DIFFERENTIAL PRESSURE
LOADS AND RESULTANT STRESSES
(EXAMPLE 1)

	Mid-core elevation		Bottom end	
Steady state Δp [[			]]	
Transient Δp [[			]]	
	P_m	P_L+P_b	P_m	P_L+P_b
Maximum Tangential Principal Stress [ ]	[			]
Maximum Axial Principal Stress [ ]	[			]

Note 1: Linearly scaled from Generic FEM Analysis Result in Table 6.8

ABB

TABLE 6.8

SVEA-96 CHANNEL WALL OVERPRESSURE STRESSES
FROM FINITE ELEMENT ANALYSIS

	Location	Value
Pressure Load, Δp	[	]
Maximum Tangential Membrane Stress, P_m	[	]
Maximum Tangential Membrane plus Bending Stress, $P_L + P_b$	[	]
Maximum Channel Deflection	[	]

ABB

TABLE 6.9

SVEA-96 CHANNEL WALL PRINCIPAL STRESSES
DUE TO SEISMIC AND LOCA EXCITATION
(EXAMPLE 1)

	Mid-core elevation	Bottom end
Seismic, Horizontal (axial principal stress)	[	]
LOCA Support Motion, Horizontal (axial principal stress) [1]	[	]
Seismic plus LOCA Support Motion, Horizontal (axial principal stress)	[	]
Seismic, Vertical (axial principal stress)	[	]

[   ]

TABLE 6.10

SVEA-96 CHANNEL STATIC PRINCIPAL STRESSES
(EXAMPLE 1)

	Mid-core elevation	Bottom end
System pressure (radial principal stress)	[	]
Weight (axial principal stress)	[	]

ABB

TABLE 6.11

SUMMARY OF MID-CORE ELEVATION CHANNEL
PRINCIPAL STRESSES AND RESULTANT STRESS INTENSITY
(EXAMPLE 1)

Load Contribution	Principal stress s_j, MPa					
	Tangential		Axial		Radial	
	P_m	P_L+P_b	P_m	P_L+P_b	P_m	P_L+P_b
System pressure	[					]
Static, vertical	[					]
Seismic plus LOCA Support Motion, horizontal	[					]
Seismic, vertical	[					]
Normal operation plus LOCA Δp Load	[					]
Total Stress	[					]

Primary Membrane Stress Intensity (max $|s_i - s_j|$) = [   ]

Primary Membrane plus Bending Stress Intensity (max $|s_i - s_j|$) = [   ]

ABB

TABLE 6.12

SUMMARY OF BOTTOM END CHANNEL PRINCIPAL STRESSES
AND RESULTANT STRESS INTENSITY
(EXAMPLE 1)

Load Contribution	Principal stress s_i, MPa					
	Tangential		Axial		Radial	
	P_m	P_L+P_b	P_m	P_L+P_b	P_m	P_L+P_b
System pressure	[					]
Static, vertical	[					]
Seismic plus LOCA Support Motion, horizontal	[					]
Seismic, vertical	[					]
Normal operation plus LOCA Δp Load	[					]
Total Stress (SRSS)	[					]

Primary Membrane Stress Intensity (max $|s_i - s_j|$) = [   ]

Primary Membrane plus Bending Stress Intensity (max $|s_i - s_j|$) = [   ]

ABB

TABLE 6.13

**SUMMARY OF FUEL ROD CLADDING PRINCIPAL STRESSES
AND RESULTANT STRESS INTENSITY
(EXAMPLE 1)**

Load Contribution	Principal stress s_i, MPa					
	Tangential		Axial		Radial	
	P_m	P_L+P_b	P_m	P_L+P_b	P_m	P_L+P_b
Fuel rod pressure loads[1]	[					]
Seismic plus LOCA Support Motion, horizontal [[[						]]]
Seismic and Static, vertical[ ]	[					]
Total Stress	[					]

Primary Membrane Stress Intensity (max $|s_i - s_j|$) = [   ]

Primary Membrane plus Bending Stress Intensity (max $|s_i - s_j|$) =[   ]

Note 1: Conservatively assumes core overpressure of [   ]

TABLE 6.14

SUMMARY OF EXAMPLE SVEA-96 FUEL ASSEMBLY COMPONENT MAXIMUM STRESSES AND ALLOWABLE LIMITS
(EXAMPLE 1)

Component		Load or stress intensity		
		Calculated	Allowable	Margin[1] (%)
Spacer grid		[		]
Fuel rod	P_m P_L+P_b	[		]
Top tie plate		[		]
Bottom tie plate		[		]
Channel	P_m P_L+P_b	[		]
Welds		[		]
Transition piece		[		]

Note 1: Margin $\equiv \left(\dfrac{\text{Allowable - Calculated}}{\text{Allowable}}\right) \times 100$

ABB

TABLE 6.15

VERTICAL FORCES ON THE SVEA-96 FUEL
DURING A COMBINED SSE & LOCA EVENT
(EXAMPLE 1)

Load	Fuel Assembly Force	
	N	(lbs)
[		]
[		]
[		]
[		]
[		]

[   ]

[   ]

ABB

TABLE 6.16

REACTOR INTERNALS MAXIMUM HORIZONTAL
NODAL FORCES
(EXAMPLE 1)

Node No.	Position	Nodal Force SVEA-96/8x8
[		]
[		]
[		]
[		]

[

]

ABB

TABLE 6.17

FUEL MODE 1 THROUGH 5 FREQUENCIES
(EXAMPLE 2)

Mode Number	Frequency (Hz)
1	[ ]
2	[ ]
3	[ ]
4	[ ]
5	[ ]

TABLE 6.18

FUEL DISPLACEMENTS AND ACCELERATIONS
(SRSS, HORIZONTAL)
(EXAMPLE 2)

Nodes		Level	Displacement	Acceleration	
Channel	Bundle	mm (inch)	mm (inch)	Relative (g)	Absolute (g)
[					]
[					]
[					]
[					]
[					]
[					]
[					]
[					]
[					]
[					]
[					]

ABB

TABLE 6.19

BENDING STRESSES IN FUEL ASSEMBLY (HORIZONTAL DIRECTION) (EXAMPLE 2)

Level	Channel		Fuel rods	
mm (inch)	node number	stress MPa (ksi)	node number	stress MPa (ksi)
[				]
[				]
[				]
[				]
[				]
[				]
[				]
[				]
[				]
[				]
[				]

ABB

TABLE 6.20

CALCULATED CHANNEL WALL LOCA PRESSURE DIFFERENTIAL LOADS AND RESULTANT STRESSES
(EXAMPLE 2)

	Mid-core elevation		Bottom end	
Steady state Δp [[			]]	
Transient Δp [[			]]	
	P_m	P_L+P_b	P_m	P_L+P_b
Maximum Tangential Principal Stress [ ]	[			]
Maximum Axial Principal Stress [ ]	[			]

Note 1: Linearly scaled from Generic FEM Analysis Result in Table 6.8

ABB

TABLE 6.21

CHANNEL WALL PRINCIPAL STRESSES
DUE TO SEISMIC AND LOCA REACTION EXCITATION
(EXAMPLE 2)

	Mid-core elevation	Bottom end
Seismic plus LOCA Support Motion, Horizontal (axial principal stress)	[	]
Seismic plus LOCA Support Motion, Horizontal, SRSS - two Directions (axial principal stress)	[	]
Seismic plus LOCA, Vertical (axial principal stress)	[	]

TABLE 6.22

CHANNEL STATIC PRINCIPAL STRESSES
(EXAMPLE 2)

	Mid-core elevation	Bottom end
System pressure (radial principal stress)	[	]
Weight (axial principal stress)	[	]

ABB

TABLE 6.23

SUMMARY OF MID-LEVEL CHANNEL PRINCIPAL STRESSES
AND RESULTANT STRESS INTENSITY
(EXAMPLE 2)

Load Contribution	Principal stress s_i, MPa					
	Tangential		Axial		Radial	
	P_m	P_L+P_b	P_m	P_L+P_b	P_m	P_L+P_b
System pressure	[					]
Static, vertical	[					]
Seismic plus LOCA Support Motion, horizontal	[					]
Seismic, vertical	[					]
Normal operation plus LOCA Δp Load	[					]
Total Stress	[					]

Primary Membrane Stress Intensity (max $|s_i - s_j|$) = [   ]

Primary Membrane plus Bending Stress Intensity (max $|s_i - s_j|$) = [   ]

ABB

TABLE 6.24

SUMMARY OF BOTTOM END CHANNEL PRINCIPAL STRESSES AND RESULTANT STRESS INTENSITY
(EXAMPLE 2)

Load Contribution	Principal stress s_i, MPa					
	Tangential		Axial		Radial	
	P_m	P_L+P_b	P_m	P_L+P_b	P_m	P_L+P_b
System pressure	[					]
Static, vertical	[					]
Seismic plus LOCA Support Motion, horizontal	[					]
Seismic plus LOCA, vertical	[					]
Normal operation plus LOCA Δp Load	[					]
Total Stress	[					]

Primary Membrane Stress Intensity (max $|s_i - s_j|$) = [   ]

Primary Membrane plus Bending Stress Intensity (max $|s_i - s_j|$) = [   ]

TABLE 6.25

SUMMARY OF FUEL ROD CLADDING PRIMARY STRESSES
AND RESULTANT STRESS INTENSITY
(EXAMPLE 2)

Load Contribution	Principal stress s_j, MPa					
	Tangential		Axial		Radial	
	P_m	P_L+P_b	P_m	P_L+P_b	P_m	P_L+P_b
Fuel rod pressure loads[1]	[					]
Seismic plus LOCA Support Motion, horizontal	[					]
Static, vertical	[					]
Seismic plus LOCA, vertical	[					]
Total Stress	[					]

Primary Membrane Stress Intensity (max $|s_i - s_j|$) = [   ]

Primary Membrane plus Bending Stress Intensity (max $|s_i - s_j|$) =[   ]

Note 1: Conservatively assumes core overpressure of [   ]

ABB

TABLE 6.26

SUMMARY OF EXAMPLE SVEA-96 FUEL ASSEMBLY COMPONENT MAXIMUM STRESSES AND ALLOWABLE LIMITS (EXAMPLE 2)

Component		Load or stress intensity		
		Calculated	Allowable	Margin[1] (%)
Spacer grid		[		]
Fuel rod	P_m $P_L + P_b$	[		]
Top tie plate		[		]
Bottom tie plate		[		]
Channel	P_m $P_L + P_b$	[		]
Welds		[		]
Transition piece		[		]

Note 1: Margin $\equiv \left(\dfrac{\text{Allowable - Calculated}}{\text{Allowable}} \right) \times 100$

ABB

TABLE 6.27

**VERTICAL FORCES ON THE SVEA-96 FUEL
DURING A COMBINED SSE & LOCA EVENT
(EXAMPLE 2)**

Load	Fuel Assembly Force	
	N	(lbs)
[		]
[		]
[		]
[		]
[		]

[   ]

[   ]

ABB

TABLE 6.28

FUEL ASSEMBLY CHANNEL DISPLACEMENTS
AND BENDING MOMENTS
(EXAMPLE 3)

Channel Nodes	Displacement mm (inch)	Bending Moment N-m (in-lbs)
[		]
[		]
[		]
[		]
[		]
[		]
[		]
[		]
[		]
[		]

[		]

ABB

TABLE 6.29

[   ]

[		]
[		]
[		]

ABB

Figure 6.1 SVEA-96 Fuel Design

ABB

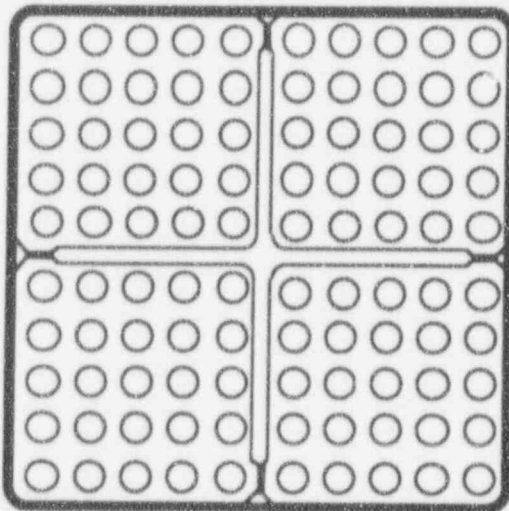**8x8**
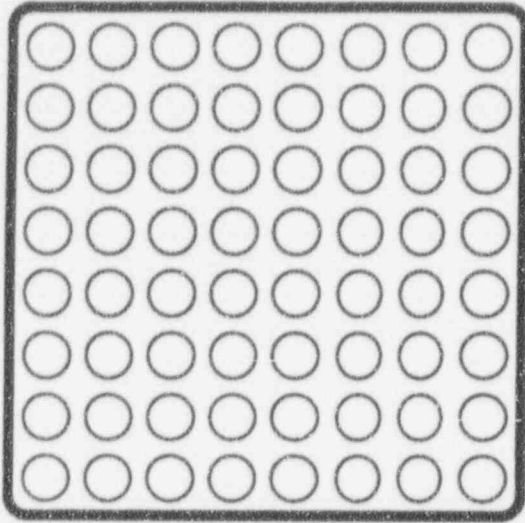
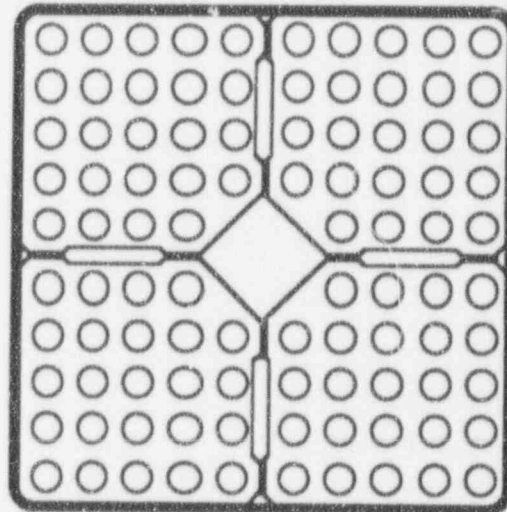**SVEA-64**
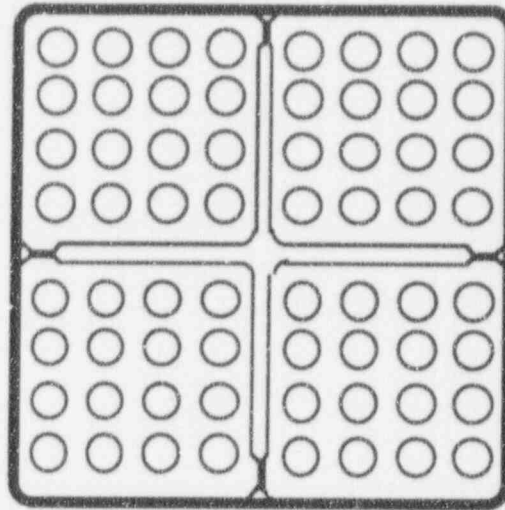
SVEA-100

SVEA-96

Figure 6.2 Cross Sectional View of Fuel Designs

ABB

Figures 6.3 through 6.35 have been deleted

Combustion Engineering, Inc. Proprietary Information

ABB

7   REFERENCES

1. "Fuel Rod Design Methods for Boiling Water Reactors," ABB Report CENPD-285-P (proprietary), May, 1994.

2. "Fuel Assembly Mechanical Design Methodology for BWR Fuel," ABB Report CENPD-287-P (proprietary), May, 1994.

3. NUREG-0800, U.S. NRC Standard Review Plan, Section 4.2 Appendix A, June, 1987.

4. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

5. ASME Boiler and Pressure Vessel Code, Section III, Appendix F, 1992 Edition.

6. ASME Boiler and Pressure Vessel Code, Section II, Part D, Appendix 2, 1992 Edition.

7. "Combining Modal Responses and Spatial Components in Seismic Response Analysis," U.S. NRC Regulatory Guide 1.92, Rev. 1, February, 1976.

8. D. L. Hagrman and G. A. Reymann, "MATPRO II - A handbook of materials properties for use in the analysis of light water reactor fuel behavior," NUREG/CR-0497, 1979.

9. "Damping Values for Seismic Design of Nuclear Power Plants," U.S. NRC Regulatory Guide 1.61, October, 1973.

10. ASME Boiler and Pressure Vessel Code, Section III, Appendix II-1520, 1992 Edition.

11. "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," RPB 90-93-P-A (proprietary), October, 1991.

12. "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity," RPB 90-94-P-A (proprietary), October, 1991.

13. BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown (SSE) and Loss-of-Coolant Accident (LOCA) Loadings (Amendment No. 1), GE Report NEDO-21175-1-A, April 1977.

ABB

APPENDIX A: ANALYSIS TOOLS

A.1 Hydraulic Analysis Codes

A.1.1 GOBLIN-EM

The GOBLIN-EM code performs the detailed thermal-hydraulic calculations for the entire reactor system following a postulated loss-of-coolant accident.

The reactor primary system under study is divided into a number of principal volumes which are further divided into any number of subvolumes. The subvolumes are the computational cells of the hydraulic model. All types of BWRs have been analyzed with the code.

Four main sections of the code can be defined:

(1) The hydraulic model which performs the solution of the basic mass, energy and momentum balances together with the equation of state for each sub-volume. This model includes empirical correlations for the calculation of pressure drops, critical flow rate, steam-water separator efficiency and steam dryer efficiency. A drift flux correlation is used to calculate the flow rates of steam and water which can predict accurately counter-current flow limiting (CCFL) phenomena.

(2) The system models of the code contain models of the various safety systems that are activated after a LOCA such as high/low pressure core spray and coolant injection systems and the automatic depressurization system (ADS). A model for the level measurement system is included. Main steam flow and feedwater flow are modeled as time-dependent sinks and sources.

(3) The fuel thermal model calculates the heat transferred from the fuel rods to the coolant. This model includes the solution of the heat conduction equation for the fuel rods, and calculation of the appropriate heat transfer coefficients at the fuel cladding outside surface.

(4) The pressure vessel and internals thermal model calculates the heat transferred from the pressure vessel and the internal surfaces to the coolant. The model includes the solution of the heat conduction equation for the components and calculation of the appropriate heat transfer coefficients.

Figure A.1 shows the interaction of the models in GOBLIN-EM.

ABB

The hydraulic model solves the set of basic equations for the coolant flow, basically, the mass, energy, and momentum balance and the equation of state.

The fluid conservation equations include approximations of all terms in the theoretical derivations for one-dimensional, drift-flux, thermal equilibrium flow (except for kinetic and potential energy terms in the energy balance which have been excluded due to their very small importance in this type of calculation).

Several empirical correlations are necessary to formulate the basic fluid equations. The most important correlations in the hydraulic part of the GOBLIN code are: the friction and local pressure drop, the drift-flux (slip), and the critical flow rate correlations.

In the pressure drop calculation, correlations are provided for single-phase friction factors and two-phase friction and local pressure drop multipliers. These correlations are based on the extensive experimental program carried out by ABB Atom in the FRIGG loop. The data base for the two-phase friction multiplier has been further extended using experimental results published in the literature. The basic formulation of the two-phase multiplier correlations is based on work done by D. Chisholm.

The two-phase energy transport between subvolumes is calculated using a steam-water drift-flux correlation which has been developed from a large data base including the FRIGG experiments. The correlation is applicable to different flow geometries and it accounts for countercurrent flow limiting (CCFL) effects in different geometries.

The choked flow model includes the Moody model with a subcooled extension, the Henry-Fauske model and the homogeneous equilibrium model (HEM). The flow rate is calculated as a user specified fraction of any of the models. The fractions can be steam quality dependent.

The fuel rod heat conduction equation is solved in its one- dimensional (radial) form (axial conduction neglected) using an implicit finite-difference technique and the appropriate heat transfer coefficients as boundary conditions.

Fuel rods as well as the different types of rods used in experiments can be modeled.

The heat transfer coefficients are calculated using the coolant state data as calculated by the hydraulic model and the surface temperature resulting from the solution of the heat conduction equation.

Heat transfer coefficients or correlations to be used also can be supplied by the user as a function of time and of axial position, steam quality or void fraction.

ABB

Detailed models for heat transfer from the pressure vessel and the internals are included. The user can specify any number of heat transferring plates which can be in contact with coolant on both sides or isolated on either side. The one-dimensional heat conduction equation is solved using a finite difference technique and a user-specified nodal subdivision of each plate. Each plates can be composed of several different materials.

Radiation heat transfer from rod to rod and from rod to channel wall can be included through user supplied gray body factors.

A.1.2 DRAGON

The DRAGON code performs the thermal-hydraulic calculation for a specified fuel assembly in the reactor core. The boundary conditions needed, i.e. pressures and enthalpies at the fuel assembly inlet and outlet, are supplied by the GOBLIN code.

The hydraulic models included in DRAGON are identical to those used in GOBLIN for the core region.

The fuel thermal model in DRAGON also is identical to the GOBLIN fuel thermal model.

The DRAGON code is used to calculate the effect of bundle power and bundle power axial distribution on the cladding temperature distribution.

A DRAGON model typically uses more detail than used in GOBLIN. For example, a SVEA-96 fuel assembly can be modeled in detail with up to seven parallel channels representing the four subchannels, the watercross wings and center channel, and the outer bypass between fuel assemblies.

A.2 Structural Analysis Codes

A.2.1 ANSYS

ANSYS is a large-scale, general purpose code recognized world-wide for its many capabilities. It is used extensively in power generation and nuclear industries. The code is developed and supported by the Swanson Analysis System, Inc., Houston, Pennsylvania. The code's capabilities include:
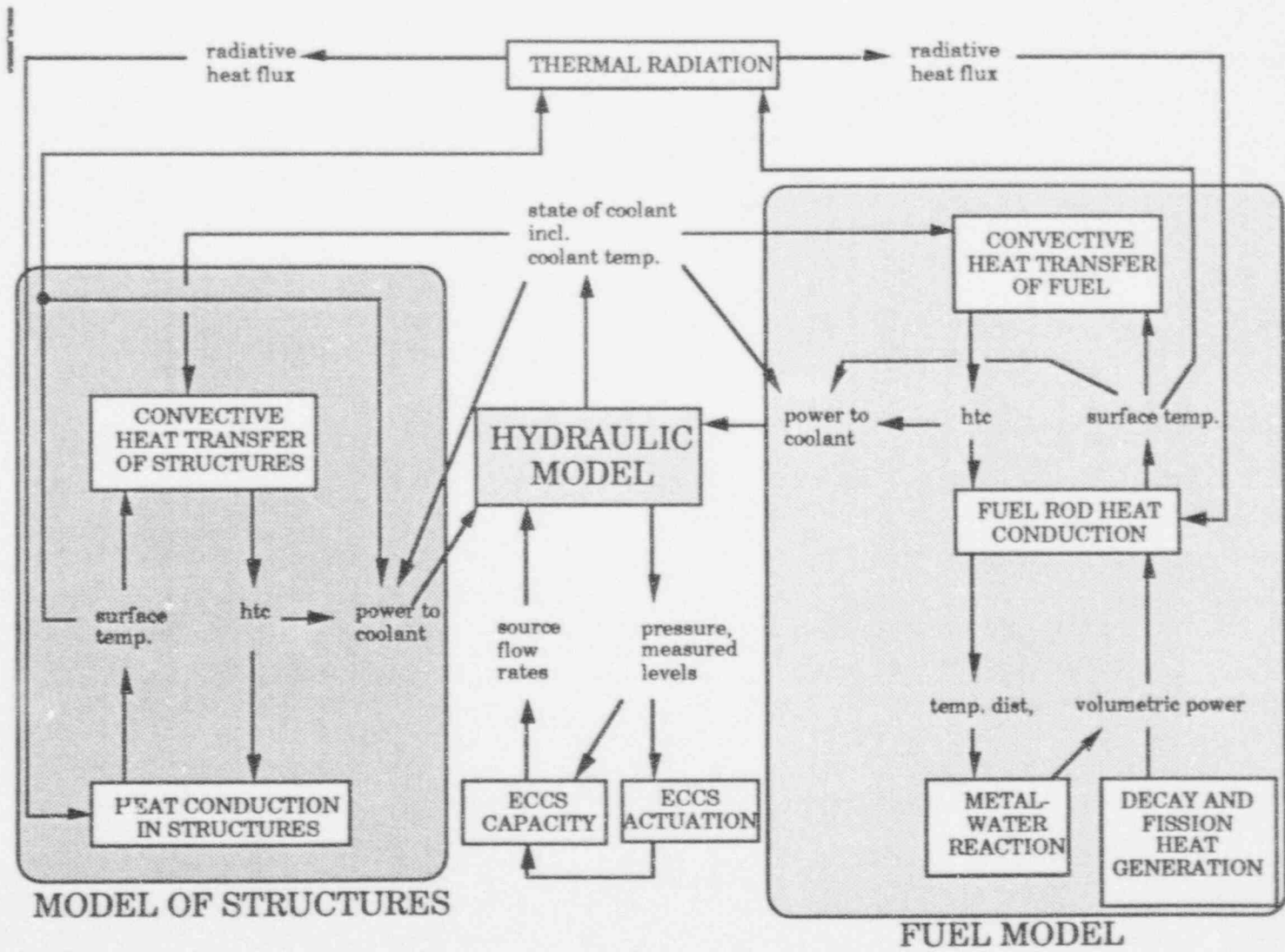
- Static and dynamic structural analysis, with linear and nonlinear transient methods, harmonic response methods, mode-frequency method, modal seismic method, and vibration analysis.

ABB

- Buckling and stability analysis with linear and nonlinear buckling.

- Heat transfer analysis with transient capability and coupled thermal structural capabilities.

- Ability to model material nonlinearities such as, plastic deformation, creep, and swelling.

- Fracture mechanics analysis.

The ANSYS element library consists of 78 distinct element types. However, many have option keys which allow further specialization of element formulation in some manner, effectively increasing the size of the element library.

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

ABB

Figure A.1 Interaction of Models in GOBLIN-EM

THERMAL RADIATION

radiative heat flux

radiative heat flux

state of coolant incl. coolant temp.

CONVECTIVE HEAT TRANSFER OF FUEL

CONVECTIVE HEAT TRANSFER OF STRUCTURES

HYDRAULIC MODEL

power to coolant

htc

surface temp.

FUEL ROD HEAT CONDUCTION

surface temp.

htc

power to coolant

source flow rates

pressure, measured levels

temp. dist.

volumetric power

HEAT CONDUCTION IN STRUCTURES

ECCS CAPACITY

ECCS ACTUATION

METAL-WATER REACTION

DECAY AND FISSION HEAT GENERATION

MODEL OF STRUCTURES

FUEL MODEL

ABB

7

ABB