ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel

ABB Combustion Engineering Nuclear Operations

ABB

CENPD-288-NP-A

ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel

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CENPD-288-NP-A

ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel

July 1996

ABB Combustion Engineering Nuclear Operations

CENPD-288-NP-A REPORT CONTENTS

CENPD-288-NP-A REPORT

Part I

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

ABB Combustion Engineering Nuclear Operations

ABB

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

May 8, 1996

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

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT CENPD-288-P, "ABB SEISMIC/LOCA EVALUATION METHODOLOGY FOR BOILING WATER FUEL," (TAC NO. M89663)

Dear Mr. Ebeling-Koning:

We have reviewed the subject topical report of May 1994, and your response of July 17, 1995, to our requests for additional information. On the basis of our review, we conclude that CENPD-288-P provides an acceptable basis for BWR fuel assembly dynamic responses to external loading conditions for licensing applications. Enclosed is our safety evaluation report (SER), which details the basis for and limitations of our approval.

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

Should our acceptance criteria or regulations change, so that our conclusions as to the acceptability of the report are no longer valid, applicants referencing this topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued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

Sincerely,

Eric W. Weiss for

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

Enclosure:
CENPD-288-P Evaluation

ENCLOSURE

SAFETY EVALUATION OF ABB/CE
TOPICAL REPORT CENPD-288-P
"ABB SEISMIC/LOCA EVALUATION METHODOLOGY FOR BOILING WATER FUEL"

1.0 INTRODUCTION

In a letter of May 26, 1994, from D. B. Ebeling-Koning, ABB Combustion Engineering (ABB/CE), to the U.S. Nuclear Regulatory Commission (NRC), ABB/CE submitted a Topical Report CENPD-288-P, "ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel," for NRC review.

CENPD-288-P describes a methodology of analyzing fuel assembly dynamic responses to the seismic and LOCA loading for a BWR. 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. According to the Appendix A to Standard Review Plan (SRP) 4.2, 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. ABB/CE will apply this methodology described in CENPD-288-P to reload licensing applications.

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

2.0 EVALUATION

INEL reviewed the ABB/CE methodology and found that the methodology is conservative and consistent with the intent of SRP 4.2 Appendix A. INEL also

made an independent calculational comparison of stress and displacement between the ABB/CE code and INEL code, ABAQUS. The results showed that the differences between these two codes are very minimal. The independent calculation confirmed that the ABB/CE code is adequate for the seismic and LOCA analysis. Based on our consultant INEL recommendations and the staff review of the TER, we agree with the INEL evaluation and conclude that the TER provides adequate technical basis to approve CENPD-288-P for seismic and LOCA response analysis.

3.0 CONCLUSIONS

The staff has reviewed the ABB/CE BWR fuel dynamic response methodology to the seismic and LOCA external loading conditions described in CENPD-288-P. Based on the conservative methodology and an independent calculation, we conclude that the methodology described in CENPD-288-P is acceptable for reload licensing applications for seismic and LOCA analysis.

INEL-96/002
Revision 1

TECHNICAL EVALUATION REPORT

ON

TOPICAL REPORT ABB SEISMIC/LOCA EVALUATION

METHODOLOGY FOR BOILING WATER FUEL

AUTHOR

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Published April 1996
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and for the
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TAC No. M89663

ABSTRACT

Lockheed Martin Idaho Technologies Company assisted NRC in reviewing, for acceptability, the topical report CENPD-288-P, "ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel" submitted by ABB Combustion Engineering Nuclear Operations. The report discusses the methodology used by the Company to evaluate fuel assembly responses to the seismic and LOCA loading for a boiling water reactor. This methodology will apply to reload applications. The review indicates that the proposed methodology is appropriate for the application. It provides a reasonable assurance that the response-results from its application would be acceptable.

PREFACE

This report is supplied as part of the Technical Assistance in Support of the Reactor Systems Branch. It is being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Systems Safety and Analysis, by Lockheed Martin Idaho Technologies, National Nuclear Operations Analysis Department.

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INTRODUCTION

Structural integrity of fuel assemblies for light water commercial reactors is a very significant factor in its safety. Therefore, safety evaluation of commercial power plants, must include structural integrity evaluation of the fuel assembly. Postulated seismic events and loss-of-coolant accidents (LOCA) cause transient mechanical loadings on the fuel assemblies. Dynamic analyses and tests (if feasible) for these loadings are performed to determine the various component loadings. The loadings are, then, used in establishing the structural integrity of the fuel assembly with respect to function, interference with control rod insertion, and coolable geometry. Based upon studies performed, review of available experimental experience data, analysis on several fuel assemblies, and input from vendors; United States Nuclear Regulatory Commission (USNRC) formulated a uniform acceptance criteria for fuel assemblies. These are given in the Standard Review Plan (SRP), NUREG-0800, Section 4.2.

ABB Combustion Engineering Nuclear Operations (ABB-CE) is one of the suppliers of Boiling Water Reactor (BWR) fuel to commercial nuclear power plants. To facilitate the process of fuel supply to different installations, ABB-CE submitted generic reports to USNRC for review and approval. These reports, among other topics, describe the design bases, acceptance criteria, and methodology used to evaluate the ABB BWR fuel assembly when subjected to postulated seismic and LOCA events. CENPD-288-P: ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel is one of these reports. The report also addresses the situation of a mixed core. It was submitted to USNRC for review of the methodology used to evaluate the fuel assembly for structural integrity including no interference with control rod insertion, and fuel coolability. ABB intends to use the methodology for reload applications in the future.

Lockheed Martin Idaho Technology (LMIT) Company assisted USNRC in the review of the report. This technical evaluation report (TER) contains the results of the review.

Section I describes the review process and includes the request for additional information (RAI) and the responses of ABB-CE. Section II has a summary of the model, computer code used and results of review analysis. The analysis is performed to evaluate the methodology, modeling technique with supporting assumptions, computer program, and reliability of the results. Finally, the conclusion of the technical review and recommendation, if any, is given in Section III.

TABLE I. LIST OF PARTICIPANTS

In discussion of RAI's

1. Shih-Liang Wu	NRC
2. Anders Johansson	ABB-ATOM
3. Art Johnson	ABB-CE
4. Derek Ebeling-Koning	ABB-CE
5. Jag N. Singh	LMIT

TABLE II. LIST OF PARTICIPANTS

In 2nd discussion of RAI's

1. Shih-Liang Wu	NRC
2. Art Johnson	ABB-CE
3. Derek Ebeling-Koning	ABB-CE
4. Jag N. Singh	LMIT

I. The Review

The report: ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel, CENPD-288-P, submitted by ABB Combustion Engineering Nuclear Operations was reviewed. Review concentrated on the methodology of fuel assembly responses to the seismic and LOCA loadings for use in boiling water reactors (BWR). This methodology will apply to reload applications.

Initial review generated sixteen requests for additional information (RAIs). They are given below:

1. It should be noted that the allowable stress intensities given on pg. 11 are for elastic analysis; what allowables are used if the component is analyzed plastically?

2. Why is fatigue evaluation not addressed for the component for the combination of seismic and LOCA loads?

3. On pg. 12, a statement is made that buckling of the channels is precluded when the maximum calculated stresses do not exceed the allowable stress intensities. Meeting these stress limits does not necessarily mean that buckling will not occur. A separate buckling evaluation, such as is prescribed in F-1331.5 Appendix F of the ASME Code Section III, is needed to assure that buckling is precluded. Is a separate buckling evaluation done?

4. The stress limits listed on pg. 11 infer that the analysis performed is elastic. If stresses in the channel structure exceed the yield stress, then the deflections would not be calculated accurately. An accurate determination of the deflection would in this case require plastic analysis. An accurate calculation for the deflection would be needed to assure safe insertion of the control rods. Is it assured?

5. On pg. 18 it is stated that if the fuel assembly lifts it will then impact against the core support plate. It is stated that the resulting fuel vertical load would range from 2 to 5 g. It is not explained how this loading was determined. If the fuel assembly and the core support plate are very stiff structures, it is conceivable that the impact loading could be significantly higher than 2 to 5 g. This would, of course, depend on the energy of the fuel assembly at impact. How was the impact deceleration determined?

6. It is stated on pg. 27 that fast neutron irradiation has little effect on the yield and ultimate strengths of the stainless steel and increased the yield and ultimate strengths of the Zircaloy. Hence, it is concluded that the use of unirradiated properties is conservative. This conclusion, however, does not account for possible effects of irradiation on the material ductility. The ASME stress limits given on pg. 11 allow the material to be stressed beyond yield on the basis that the material is ductile. To use these stress limits, it must be assured that the material maintains this ductility when irradiated. Has it been established that the material will maintain ductility at the irradiation levels experienced?

7. It is stated on pg. 27 that the test load applied to determine stress limits for the spacer grid and channel weld was cyclic force. Is the number of load cycles applied in the test representative of the number of cycles expected in the actual installation?

8. On pg. 29, has the fuel rod performance code VIK been approved by the NRC?

9. On pg. 29, the methods used to combine stresses from different loadings on the fuel rods are described. Since deformation of the channel is important, a description of the method used to determine the total deflection should also be provided. This should include the methods used if the structure goes plastic. What is the method used to determine the total deflection?

10. On pg. 15, sect. 5.1.1.1, Discussion: Does the model of fig. 5.2 include the confined fluid?

11. On pg. 31, sect. 5.2.1, Methodology, steps (1), (2), & (3):

    (a) How is bounding arrived at in step (1)?
    (b) What is the significance of not going through step (2) in all cases, irrespective of the results from step (1)?
    (c) More information is necessary to establish the validity of step (3), Please provide.

12. On pg. 31, sect, 5.2.1, Discussion, Step (3):

    A technical justification for step (3) is required.

13. On pg. 58, sect. 6.3.2.1, GOBLIN/DRAGON Model:

    Are the results of the three channel and one channel representations significantly different? How do they compare?

14. On pg. 69, sect. [Fuel Assembly Vertical Acceleration]:

    The significance of the last paragraph is not evident. Clarification is required.

15. On pg. 77, sect. 6.5.1.2, Resident Fuel Assembly Seismic response, (also applies to sects. 6.5.1.3 & 6.5.2):

    Further explanation is required to establish the validity of the conclusion(s).

16. The integrity evaluation of a mixed core is not addressed. Is there a possibility of mixed core being used? If so, how is integrity established?

These were first discussed in a conference call with representatives of NRC, LMIT, and ABB, participating. Table I has the list of the participants.

The discussion, mostly, consisted of ABB asking for the kind of information the reviewers were looking for and clarifications. In some cases, the discussion did go into technical details of the investigation and tests performed by ABB. Agreements among the participants, on the need and type of information, were generally reached. In almost all cases, the major points were ironed out and mutual agreements were arrived at. The NRC expressed the need for a formal response to the RAIs, with the type of information the reviewers were looking for. ABB-CE, subsequently, submitted a formal response entitled, ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel: Response to Request for Additional Information; CENPD-288-P-RAI. The responses to the RAI's are summarized below:

1. It basically states that ABB BWR fuel design used only elastic analysis. It further notes that, if in the future, the need for a elastic-plastic analysis arises, the analysis would be submitted to NRC under separate application.

2. ABB's response explains that CENPD-288-P deals with the methodology for evaluating fuel bundles for postulated safe shutdown earthquakes and loss-of-coolant accident (LOCA) events. Therefore, level D service limits applies and a fatigue evaluation was not required. However, integrity of spacer grid and channel weld is evaluated in low cycle fatigue test. It, further, adds that a fatigue evaluation is, indeed, performed for normal operating loads and it is described in their reference 5.

3. The applicant indicates that a test is done on the channel to demonstrate the integrity of the channel against buckling when subjected to seismic and LOCA loads and deflections. These loads do not cause significant compressive axial load on the channel. Local plate buckling by lateral buckling would be the main concern. However, the applicant further adds that the channel acceptance criteria for design are:

   a. That the maximum stress based upon elastic analytical methods do not exceed stress intensities.
   b. The maximum deflection is controlled by its ability to allow safe insertion of the control rods which are done elastically, and
   c. The stresses and deflections, calculated elastically, are within allowables for channel weld strength and channel buckling loads and deflections are below the limits established through test simulating seismic plus LOCA condition.

4. In summary the response to this RAI, reiterates the point that all the analysis involved are based upon elastic model and limits. No elastic-plastic model is presently used. In the future, if there is a need for elastic-plastic model to be used, the calculations would be submitted to NRC for concurrence.

5. ABB's response indicates that the impact loads are based upon a census of several plant licensing based analyses, determined by the plants' original equipment supplier.

6. The response indicates that a fuel assembly, typically, consists of a Zircaloy channel and fuel rods with stainless steel castings for the bundle and channel end pieces. Among these, the stainless steel castings are of relatively solid design and present no concerns. Therefore, the focus is the Zircaloy pieces. For these, the stress limits are indeed based upon minimum material properties for unirradiated material. This, however, is conservative because irradiation of Zircaloy significantly increases its yield and ultimate strengths. They are expected to rise by a factor of about 2 and 1.5 respectively at relatively low irradiation values.

   The applicant had tabulated data from reference 16. Virtually, all the strengthening occurs early in the fast fluence exposure. The data is extrapolated. It is reasonable and supported by data given in reference 17. The changes occurring in the higher fast fluence exposures are relatively small. The conclusion, therefore, is that change in ductility is not going to, adversely, effect the results obtained by using unirradiated strength values. These details were not available in the first response. This was one of the concerns which was cleared in the second response, which is further discussed later in this report.

7. With respect to the inquiry about the number of load cycles in the test being representative of the actual experienced, the response of higher than 1000 load cycles is deemed to be adequate and may even be conservative. For example, a 30 seconds duration earthquake may have, conservatively speaking, 150 significant load cycles. The simulation of safe shutdown earthquake (SSE) plus LOCA, suggestions of reference 9, was followed. The test also simulated, by scaling factor, the effects of temperature, irradiation, and test statistics according to the same reference.

8. For the response to the question of the computer code VIK-II being approved by NRC, the applicant referred to document CENPD-285-P, which was under review by NRC and expected to be approved in 1995. It is the same code being used in document CENPD-288-P, reference 1.

9. In response to the method of combining the deflections from different loadings, and specially if the structure goes plastic, the applicant indicated that square root of the sum of the squares (SRSS) method is used. All the calculations, however, are based on the structure remaining elastic, as pointed out earlier.

10. The response to this RAI has the information which confirmed the consideration of confined fluid. The water within the channel of the fuel assembly is represented as an additional mass and dynamic fluid coupling elements are incorporated to simulate the confined fluid between components within the reactor vessel.

11. The applicant response is summarized below:

Section 5.2.1 describes the methodology used to demonstrate that the fuel assembly will remain engaged in the lower support during and after a postulated SSE/LOCA event. It gives a description of the lift forces, which are considered. However, some of the force/forces are not evaluated, specifically, since it is not necessary for reload as discussed below in the evaluation process steps (A-D);

(A) The evaluation process is based upon the relative values of uplift forces on the resident and reload fuel assemblies. As long as the sum of the reload lift forces is enveloped by the corresponding lift forces on the resident fuel assembly, the reload assembly is assured of keeping engaged. Further, this is necessary only for significantly variable forces. This keeps the licensing basis for the plant valid. The point is that all the individual lift forces (not significantly effected by fuel type, e.g.; the vertical acceleration of the core plate due to seismic & LOCA) are not considered in the process.

(B) The evaluation process in this step compares the sum of all the individual maximum value of uplift forces, regardless of time phasing, to the total weight of the fuel bundle. If the total weight is greater than the sum of individual forces, the fuel assembly is assured of keeping engaged. However, for this process to be done the individual uplift forces must be known, as explained in the preceding section (A). That is why, this process is not carried through in all cases, if process (A) is sufficient for the purpose.

(C) This is a vigorous analysis step, which is hardly ever done for reload. It deals with detailed non-linear time history analysis in which all the forces acting on the assembly are considered. They include, all time dependent base-support plate motion due to seismic & LOCA, hydraulic lift forces, and friction force between control rods and channel. Further, hydraulic forces are a function of lift height, are also of different duration and phasing is not explicitly known. The SSE, LOCA and scram forces are handled by phasing them in to produce maximum uplift. Several cases are examined for conservativeness.

However, this method is very time consuming and expensive for a reload fuel vendor to perform. There are also limitations to the extent the data required could be obtained by a vendor. Obviously, therefore, (A) or (B) is preferred. Use of (C) is very limited and done only if (A) and (B) both fail.

12. The applicant's response to this inquiry is summarized. It discusses the effects of lateral stiffness on the resistance of the control rod channel on the control rod during insertion. In essence, if the channel is relatively soft, it gives relatively less resistance to insertion. Further, this fact has been verified in tests performed by ABB.

13. ABB's response suggests that there is no significant difference in results of the three channel and one channel representation of the reactor core based upon computer code "GLOBLIN." This assertion is based upon their sensitivity studies.

14. In response to this inquiry, it is said that the licensing basis for a specific plant gives the maximum vertical acceleration value. The maximum acceleration during the first 30 seconds, while the control rod is being inserted, is much lower than the maximum value given for the plant. The ratio is about 15 to 1. Due to this reason, a lower value is used in lift analysis of the fuel and the higher value is used for the stress evaluation. It is pointed out that in the first 30 seconds the fuel assembly can lift up and misalign, blocking the insertion of the control rods. On the other hand, even if the bundle were to lift up after the control rods have inserted, it can not hamper insertion and it would be immaterial for the insertion process. However, for stress calculation, the higher value is relevant and used. This idea is further stressed in the paragraph in question. This RAI was another one which was resolved after the second response.

15. In explaining and establishing the validity of the paragraph in question (Section 6.5.1.2), it is pointed out that the information about seismic and LOCA for a specific plant is limited. To make the process manageable and easier, the supplier used an alternate way to address it. The process involves establishing a spectrum for the analyses of the reload fuel bundle. This spectrum is selected as a candidate on the basis of being either typical for BWR plants or enveloping typical BWR plant spectra. The candidate should satisfy the following test-criteria. These criteria are: (a) that the candidate spectrum has a higher acceleration value at the first mode frequency of the reload fuel than at the first mode frequency of the resident fuel, and (b) that the acceleration response it produces in the reload fuel at its own first mode frequency is at least as high as the acceleration response it produces in the resident fuel, at its own first mode frequency.

16. In response to this RAI, the applicant submits that mixed-core is the scenario in most operating plants. This fact has been accounted for in the proposed methodology for seismic/LOCA evaluation. A very basic assumption in the process is the strong hydrodynamic coupling between the fuel assemblies. It is argued and supported by references 11 and 12, that BWR assemblies do move in phase. Therefore, a single "stick" model in the RPV model is adequate to represent all the fuel assemblies in the core. Further, a confirmatory evaluation for a mixed core has been done by the applicant. It confirmed the fact that fuel designs have relatively similar seismic characteristics.

However, a significant number of the responses referred to reference reports, proprietary to ABB. These reports were not readily available to the reviewer. ABB-CE was contacted again to provide these reports, so that the statements and conclusions could be substantiated. The reports were made available and the conclusions were substantiated. Some of the conclusions in this TER may be deemed to be based upon the references. Therefore, for completeness, the references are given in Section IV. It is pointed out here that most of the responses, along with the supporting documents, were acceptable after the first submittal, as discussed earlier. There were two outstanding that needed further information to arrive at a satisfactory conclusion.

To discuss the remaining two issues, a subsequent conference call was arranged. The list of the participants is given in Table II. During the discussion, ABB-CE agreed to provide additional detailed information on a fuel bundle. It is one of the fuel bundle ABB-CE supplies to the commercial power plants. To further enhance the confidence in the methodology, modeling techniques, and the reliability of the results, LMIT intended to, independently, model and analyze it for verification of ABB-CE's analysis results.

ABB-CE supplied the needed information. The Company also, formally, submitted a modified RAI response entitled: ABB Seismic/LOCA Methodology for Boiling Water Fuel: Response to Request for Additional Information; CENPD-288-P-RAI, Rev.1. This submittal was a revised version of the first. Basically, it modified the responses to questions 6 and 14, incorporating new information and added an appendix C to it. The modifications and additions adequately addressed the remaining concerns.

II. Analysis and Results

LMIT prepared a finite-element model for ABAQUS, a computer program. The model had material properties supplied by the vendor and met reviewer expectations. With the boundary condition and loads applied, it represents an authentic engineering analysis model. The ABAQUS analysis code, is a prevalent code and can be considered acceptable in the industry. The results of the ABAQUS analysis are within the engineering tolerance to the results obtained by the vendor and are summarized as follows:

The mode shapes and frequencies are acceptably close. The percentage of frequency difference, comparatively higher in higher modes, is usually normal. Their effect on the stresses and deflections are minimal and are judged to be insignificant in these cases. The results, in terms of stresses, acceptably approximate the vendors. The deflections are also quite close. Their percentage differences are given in Appendix A. Based upon these results, it is inferred that the results obtained by ABB-CE, using ANSYS codes, are reasonable. The summary of the ABAQUS analysis are given in Appendix A.

III. Conclusion

Based upon the review of the topical report CENPD-288-P: ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel, the responses to request for additional information in CENPD-288-P-RAI: ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel: Response to Request for Additional Information, along with its first revision CENPD-288-P-RAI, Rev. 1, ABB's responses to inquiry during conference calls, and the verification of Fuel Assembly ANSYS Model, it is concluded that the topical report does present an adequate and acceptable methodology to evaluate the ABB BWR fuel assembly subjected to postulated seismic and LOCA events.

IV References

1. ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel, CENPD-288-P, NRC transmittal letter NFBWR-94-018, May 26, 1994
2. NRC Facsimile Transmission from S. L. Wu (NRC) to D. Ebeling-Koning (ABB), June 15, 1995
3. ASME Boiler and Pressure Vessel Code, Section III, Appendix F, 1992 Edition
4. ASME Boiler and Pressure Vessel Code, Section III, Part NB, 1992 Edition
5. Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, ABB Report CENFPD-287-P (proprietary), CENPD-287-NP (nonproprietary), June, 1994
6. River Bend Station Updated Safety Analysis Report, Section 3.9, Table 3.9B-2aa, August 1987
7. BWR Fuel Assembly Evaluation of Combined Safe Shutdown (SSE) and Loss-of-Coolant Accident (LOCA) Loadings (Amendment No.3), GE Report NEDO-21175-3-A, October 1984
8. Fuel Rod Design Methods for Boiling Water Reactors, ABB Report CENPD-285-P (proprietary) CENPD-285-NP (nonproprietary), May 1994
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10. U.S. NRC Standard Review Plan Section 4.2, Appendix A, NUREG-0800, July 1981
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12. T. Ikeda, et al., "Analysis of Response of BWR Core Structures to Earthquakes," Journal of Nuclear Science and Technology, August 1984
13. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification, ABB Report RPB-90-93-P-A (proprietary), RPB-90-91-NP-A (nonproprietary), October 1991
14. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity, ABB Report RPB-90-94-P-A (proprietary), RPB-90-92-NP-A (nonproprietary), October 1991
15. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96, CENPD-283-P (proprietary), CENPD-283-NP (nonproprietary), March 1993

Appendix A

Verification of Fuel Assembly ANSYS Model

Verification of Results

prepared by:

Tom E. Rahl

checked by:

Jag N. Singh

April 1996

Verification of Results

1. Objective

The objective of this task is to verify the numerical validity of the fuel bundle assembly ANSYS computer model contained in CENPD–288–P (ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel).

The method of verification is to create a new model of the fuel assembly using a different computer code (ABAQUS), apply the same loading, and compare the results.

2. Discussion of ABAQUS Finite Element Model

Geometry and material properties used in the ANSYS model are used in the ABAQUS model. A sketch of the ABAQUS model created is given in Figure 1.0.

Boundary conditions applied to the model are listed in Table 1. They are graphically shown in Figure 2.0.

Table 1: Boundary Conditions

Node Numbers	Degrees of Freedom (1–fixed, 0–free)					
	x–trans	y–trans	z–trans	x–rot	y–rot	z–rot
2 thru 8	0	1	1	1	1	0
10 thru 19	0	1	1	1	1	0
1 & 9	1	1	1	1	1	0

Zero mass beam elements are used to model the channel and fuel rods. All masses are modeled as lumped mass elements.

The spacer grid square and the subchannel are such that the clearance between the two is very small. It is, therefore, reasonably assumed that the interface between the two is rigid. A very stiff spring is used in the model to represent the interface.

3. Load

A response spectrum of the load is used in the analysis. The spectrum is graphically shown in reference 1.

Modal analysis first is performed. The first five eigenvalues are determined along with their mode shapes. The response spectrum analysis is then performed, which determine response loads and displacements.

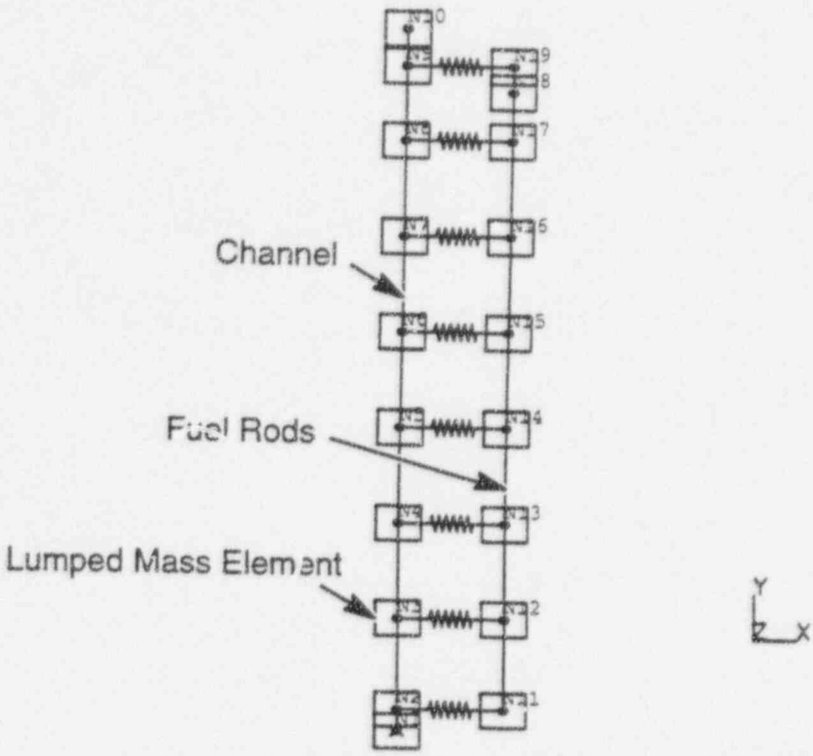


Figure 1.0 ABAQUS Finite Element Model

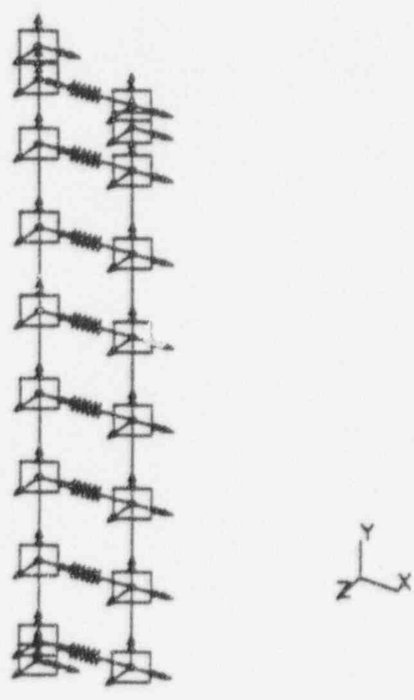


Figure 2.0 Graphical Representation of Boundary Conditions

4. Results Comparison

Comparisons of the results from the two models are presented in this section.

4.1. Modal Results

Percentage differences of the calculated eigenvalues for the two models are compared in Table 2.

Table 2: Eigenvalues

Mode Number	% Difference
1	0.8
2	2.8
3	4.9
4	21.2
5	7.9

Note that the higher modes (4 and 5) are very sensitive to the lateral stiffness assumed between the fuel bundle and the channel. However, these modes are not the primary load contributing. Therefore, the relatively high differences are acceptable.

Mode shapes of the analysis are presented in the attachment (pages A–8 thru A–12).

4.2. Displacement Results

Percentage differences of the calculated displacements of the two analyses are shown in Table 3.

Table 3: Displacements

Node Number	Displacements (% Difference)
1	0.0
2	2.9
3	2.9
4	3.2
5	3.0
6	3.1
7	12.7
8	3.5
9	0.0
10	1.7

4.3. Stress Results

Calculated stress differences for the two models are compared in Table 4.

Table 4: Bending Stress

Node Number	Stress (% difference)
1	0.0
2	13.6
3	1.9
4	1.1
5	1.4
6	1.1
7	2.7
8	0.4
9	17.6
10	0.0

Note that the relatively large differences (nodes 2 & 9) represent very low stress levels.

5. Conclusions

Based on the eigenvalue, displacement, and stress comparisons, it is observed that the ANSYS model (references 1 & 2) very closely correlates with the results of the ABAQUS model and therefore, accuracy of the ANSYS model is verified.

6. Computer Data

Computer Program Configuration Documentation for ABAQUS is given on page A-13 of the attachment.

7. References

1. ABB Combustion Engineering Nuclear Operations, "ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel", CENPD-288-P, May 1994.

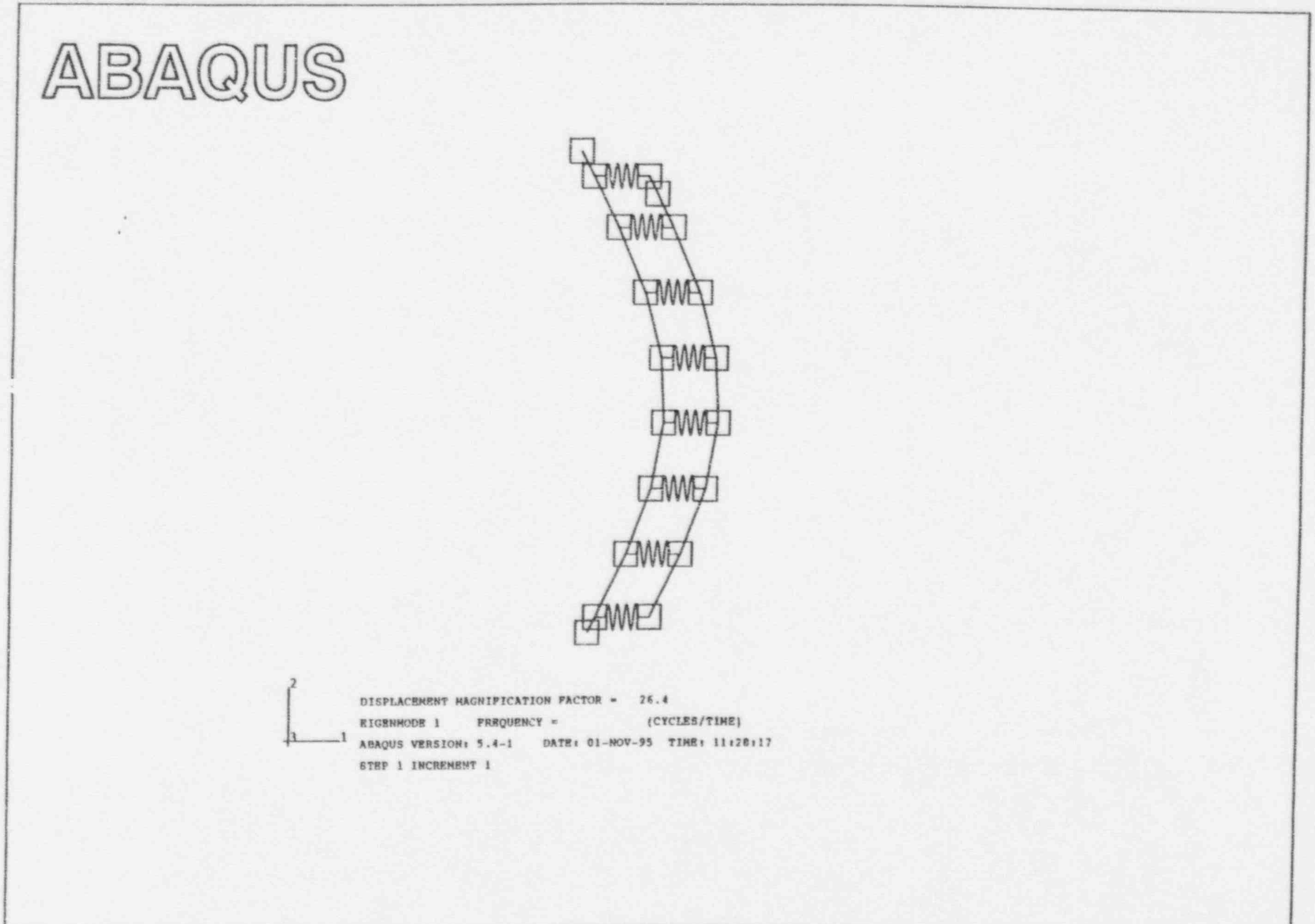
2. ABB Combustion Engineeri[illegible] [illegible]lear Operations, "Fuel Assembly Mechanical Design Methodolo[illegible]y for Boiling Water Reactors", CENPD-287-P, June 1994.

ATTACHMENT

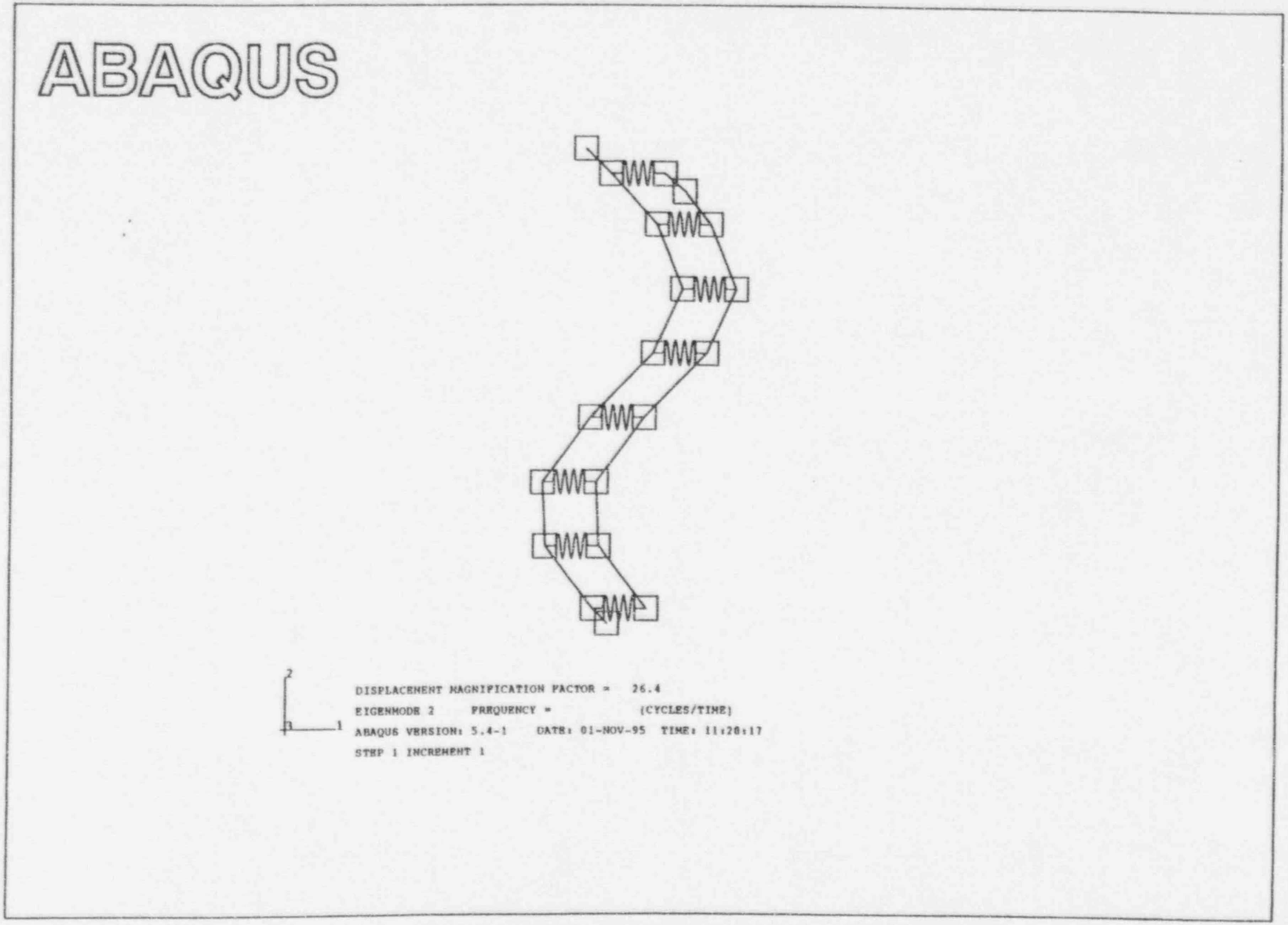
Mode Shape Plots

Computer Code Documentation

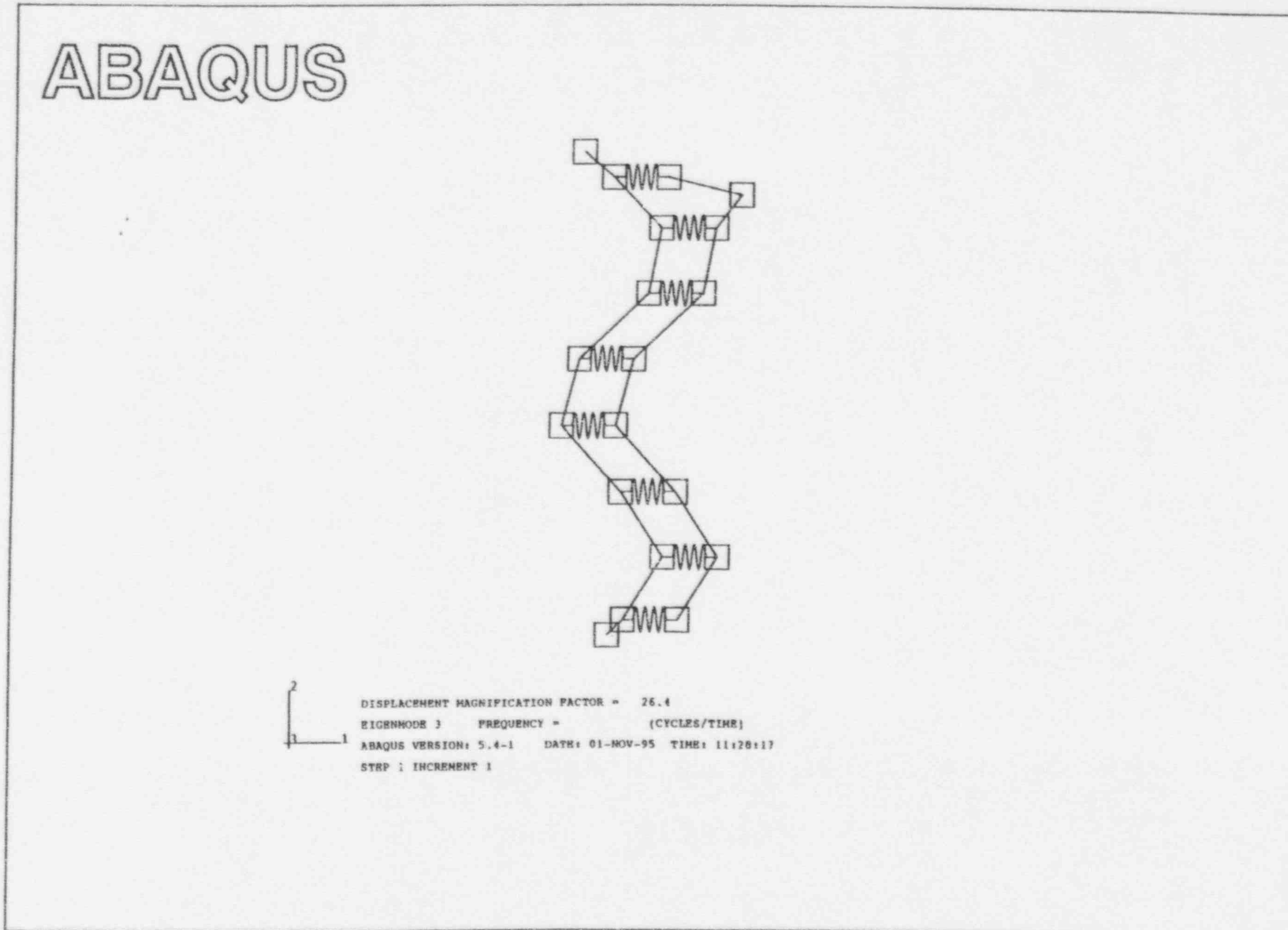
Plot of Eigenmode 1



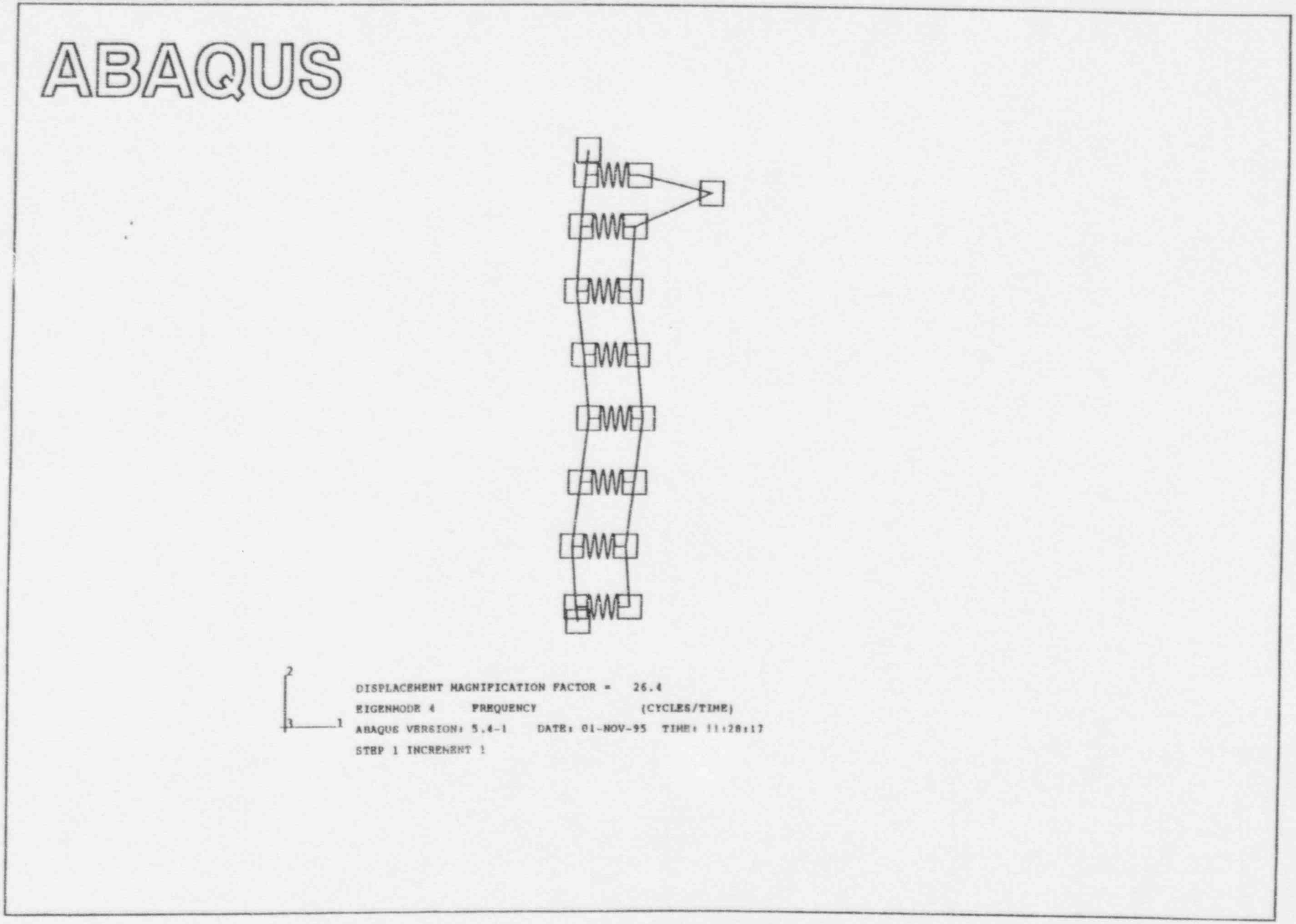
Plot of Eigenmode 2



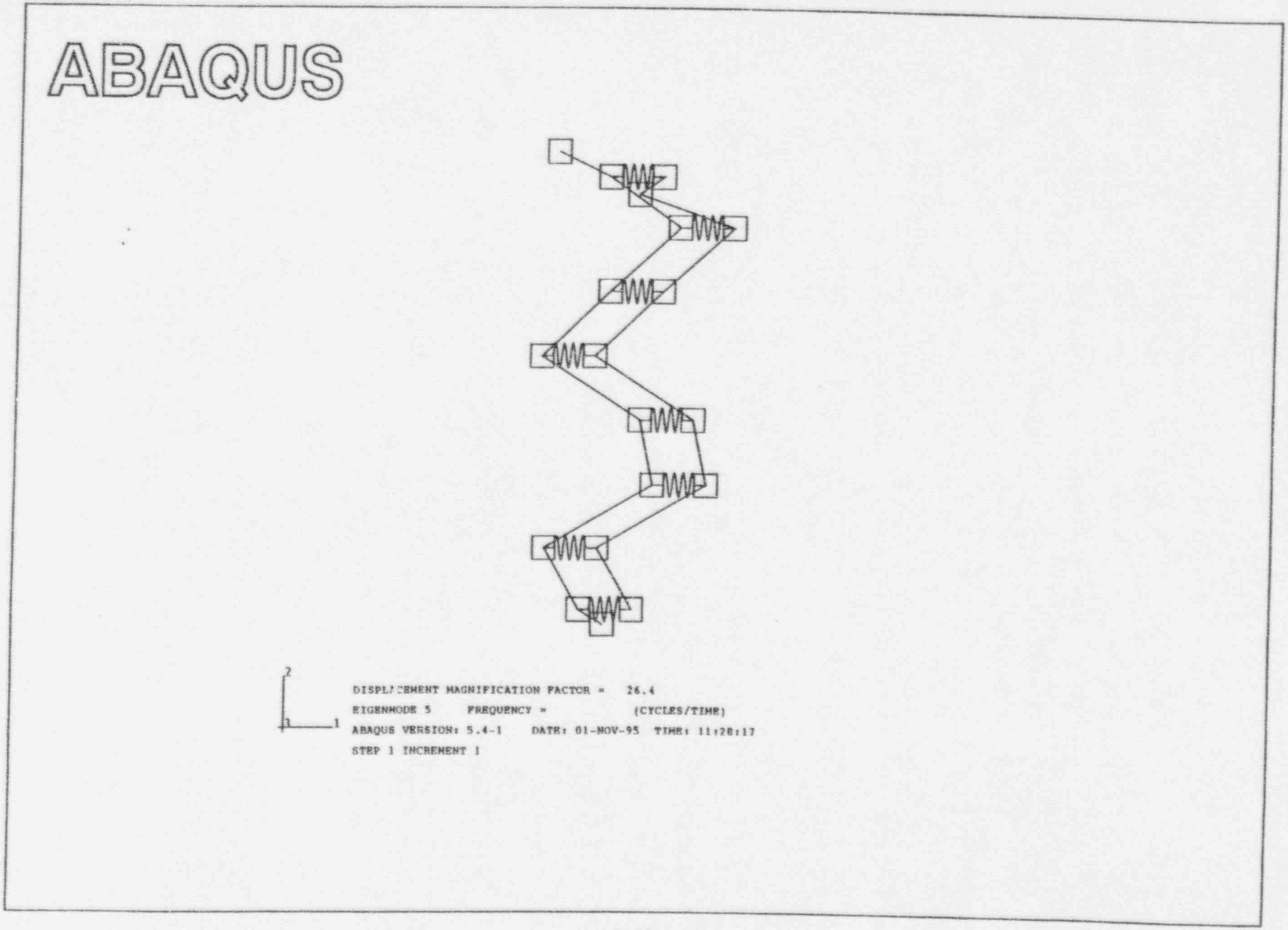
Plot of Eigenmode 3



Plot of Eigenmode 4



Plot of Eigenmode 5



APPLIED MECHANICS UNIT
COMPUTER PROGRAM CONFIGURATION DOCUMENTATION

The following documentation presents the traceability for computer programs used in the analysis reported here. This documentation should accompany any and all analysis reports transmitted by Applied Mechanics.

Task: Response Spectrum Analysis using Beam Elements

Charge No: 411355100

Report Title: Model Verification of Results ,Fuel Assembly ANSYS Model

Author: T.E. Rahl **Date:** Nov 1995

Program Used: ABAQUS **Version:** 5.4

Computer Used: Thayne **Model:** Alpha Work Station

Verification Manual/Test Problem Manual/Example Manual:
"ABAQUS/Standard Verification Manual, "Hibbitt, Karlsson & Sorensen, Inc.

CENPD-288-NP-A REPORT

Part II

Body of Report

ABB Combustion Engineering Nuclear Operations

ABB

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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

Proprietary Information Deleted

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 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.

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.

(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.

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.

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]

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.

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 arises 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]

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]

[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 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]

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 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]

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]

TABLE 5.1 AND TABLE 5.2

Proprietary Information Deleted

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.

TABLE 5.4

Proprietary Information Deleted

FIGURE 5.1 THROUGH FIGURE 5.10

Proprietary Information Deleted

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°F) 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.

<u>Lateral Load Cycling Test, Channel and Spacer Grid</u>

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]

<u>Channel Buckling Test</u>

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:

(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.

[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.

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

The general-purpose, finite element program ANSYS (Appendix A.2) was used for this structural analysis. [Proprietary Information Deleted]

6.3.1.2 Model Qualification

Time Step

[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]

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.

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]

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] 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 allowable values, 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 tie 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

<u>System Response</u>

[Proprietary Information Deleted]

<u>SVEA-96 Fuel Pressure Loads</u>

[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.

<u>Stresses from Seismic Displacement Loads</u>

[Proprietary Information Deleted]

<u>Stresses from LOCA Pressure Loads</u>

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.

<u>Total Channel Stress Intensity</u>

[Proprietary Information Deleted]

<u>Comparison with Stress Limits</u>

[Proprietary Information Deleted]

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

6.4.3.2 Spacer Grids

[Proprietary Information Deleted]

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)

\# Proprietary Information Deleted

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)

\# Proprietary Information Deleted

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)

\# Proprietary Information Deleted

TABLE 6.2

SUBBUNDLE NATURAL FREQUENCIES

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

\# Proprietary Information Deleted

TABLE 6.3

FUEL ASSEMBLY MODEL/TEST CORRELATION

[Proprietary Information Deleted]

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

\# Proprietary Information Deleted

TABLE 6.4A AND TABLE 6.4B

Proprietary Information Deleted

TABLE 6.5

ABSOLUTE FUEL HORIZONTAL ACCELERATIONS (EXAMPLE 1)

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

\# Proprietary Information Deleted

TABLE 6.6

FUEL DISPLACEMENTS (EXAMPLE 1)

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

\# Proprietary Information Deleted

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

\#   Proprietary Information Deleted

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	[# [#	#] #]

\# Proprietary Information Deleted

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)	[# [#	#] #]

[Proprietary Information Deleted]

\# Proprietary Information Deleted

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)	[# [#	#] #]

\# Proprietary Information Deleted

TABLE 6.11

SUMMARY OF MID-CORE ELEVATION 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	[#	#	#	#	#	#]

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

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

Proprietary Information Deleted

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 $|s_i - s_j|$) = [#]

\# Proprietary Information Deleted

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 [#]

\# Proprietary Information Deleted

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

\# Proprietary Information Deleted

TABLE 6.15

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

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

[Proprietary Information Deleted]

[Proprietary Information Deleted]

Proprietary Information Deleted

TABLE 6.16

REACTOR INTERNALS MAXIMUM HORIZONTAL NODAL FORCES (EXAMPLE 1)

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

[Proprietary Information Deleted]

\# Proprietary Information Deleted

TABLE 6.17

FUEL MODE 1 THROUGH 5 FREQUENCIES (EXAMPLE 2)

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

\# Proprietary Information Deleted

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)
[#	#	#	#	#	#]
[#	#	#	#	#	#]
[#	#	#	#	#	#]
[#	#	#	#	#	#]
[#	#	#	#	#	#]
[#	#	#	#	#	#]
[#	#	#	#	#	#]
[#	#	#	#	#	#]
[#	#	#	#	#	#]
[#	#	#	#	#	#]
[#	#	#	#	#	#]

\# Proprietary Information Deleted

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)
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]
[#	#	#	#	#]

\# Proprietary Information Deleted

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

\# Proprietary Information Deleted

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)	[# [#	#] #]

\# Proprietary Information Deleted

TABLE 6.22

CHANNEL STATIC PRINCIPAL STRESSES (EXAMPLE 2)

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

\# Proprietary Information Deleted

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 $|s_i - s_j|$) = [#]

\# Proprietary Information Deleted

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 $|s_i - s_j|$) = [#]

\# Proprietary Information Deleted

TABLE 6.25

SUMMARY OF FUEL ROD CLADDING PRIMARY 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
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 [#]

\# Proprietary Information Deleted

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 $\equiv \left(\frac{\text{Allowable - Calculated}}{\text{Allowable}}\right) \times 100$

\#   Proprietary Information Deleted

TABLE 6.27

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

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

[Proprietary Information Deleted]

[Proprietary Information Deleted]

Proprietary Information Deleted

TABLE 6.28

FUEL ASSEMBLY CHANNEL DISPLACEMENTS AND BENDING MOMENTS (EXAMPLE 3)

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

\# Proprietary Information Deleted

TABLE 6.29

Proprietary Information Deleted

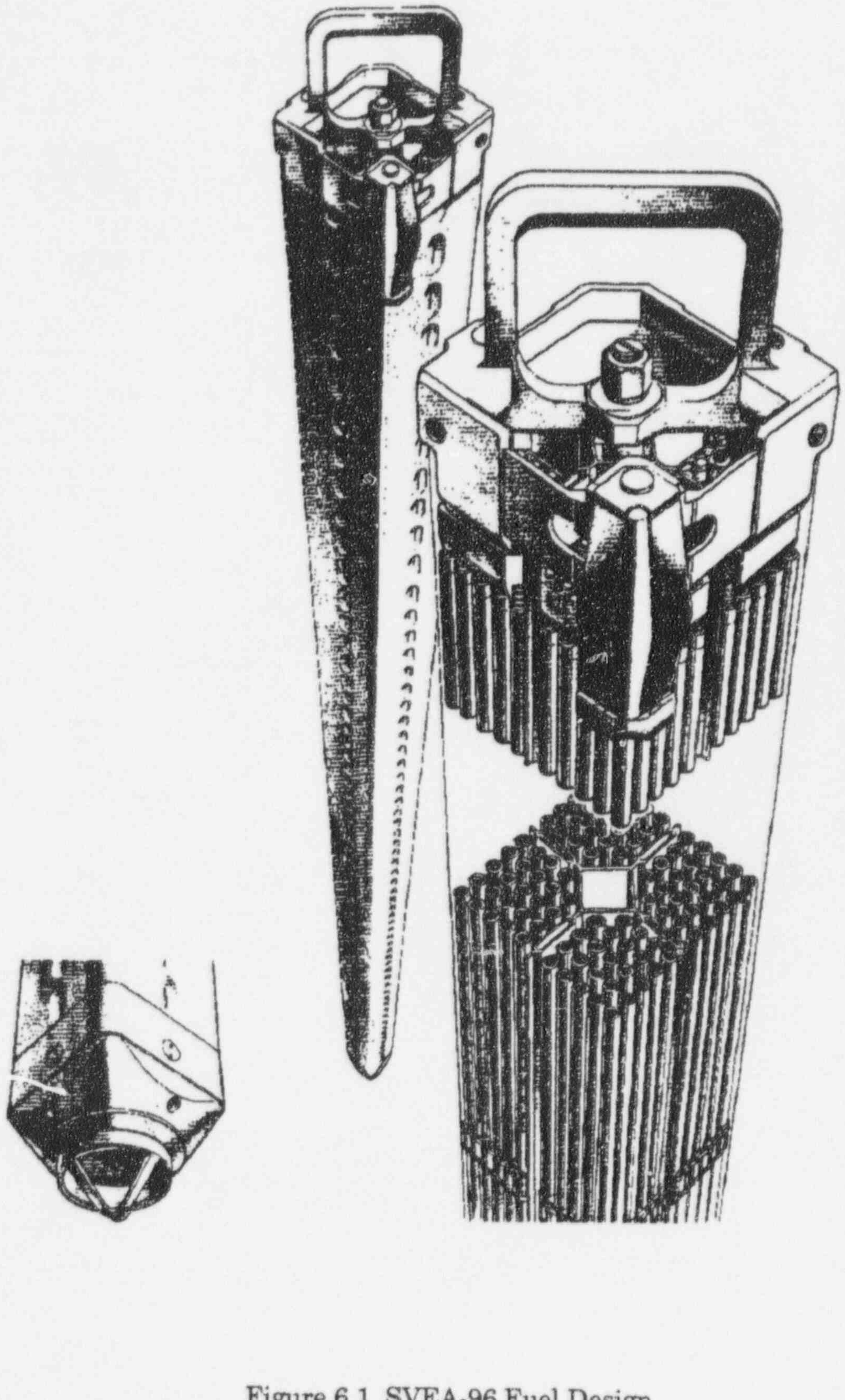

Figure 6.1 SVEA-96 Fuel Design

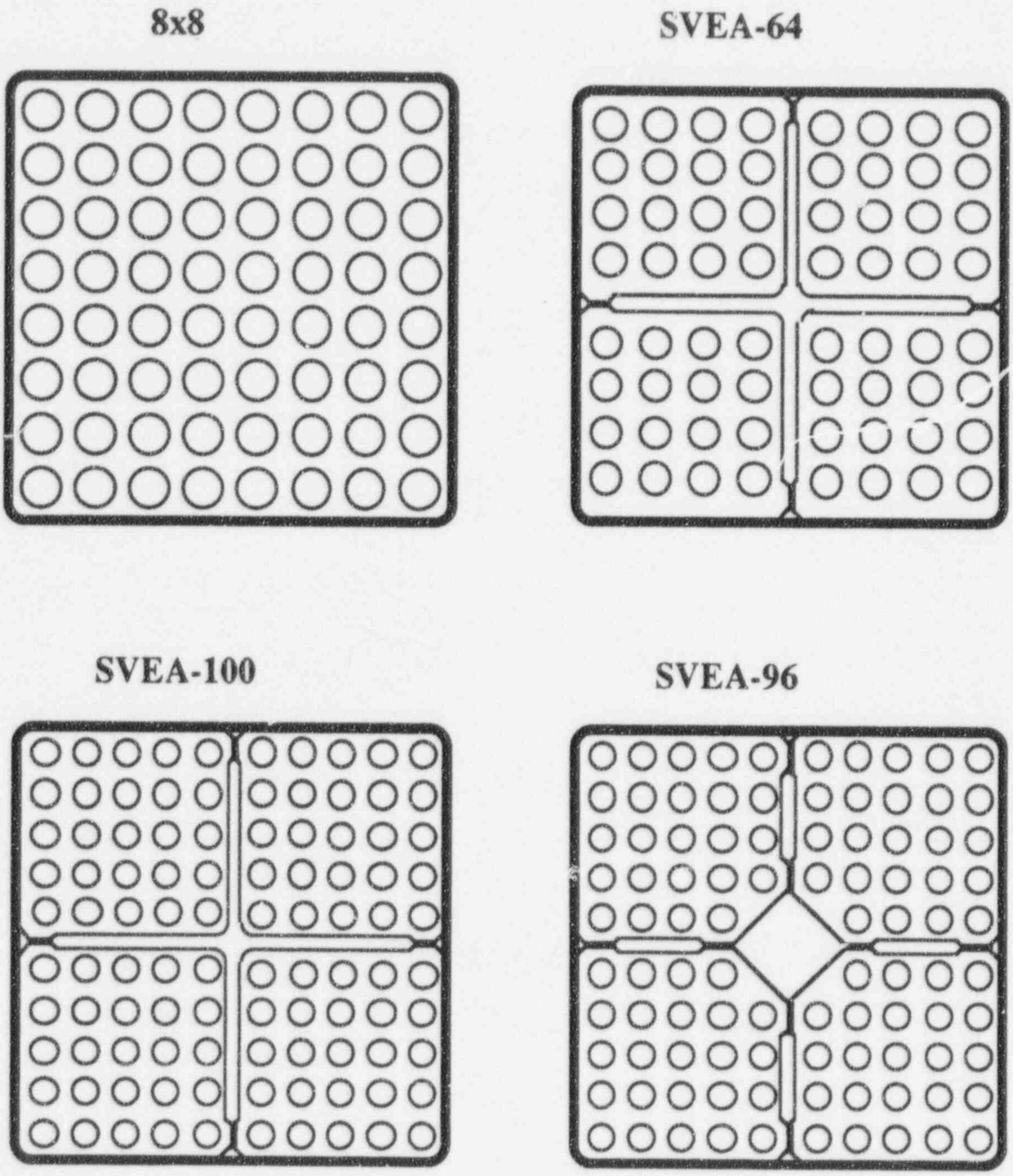

Figure 6.2 Cross Sectional View of Fuel Designs

FIGURE 6.3 THROUGH FIGURE 6.35

Proprietary Information Deleted

7 REFERENCES

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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.

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.

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.

- 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.

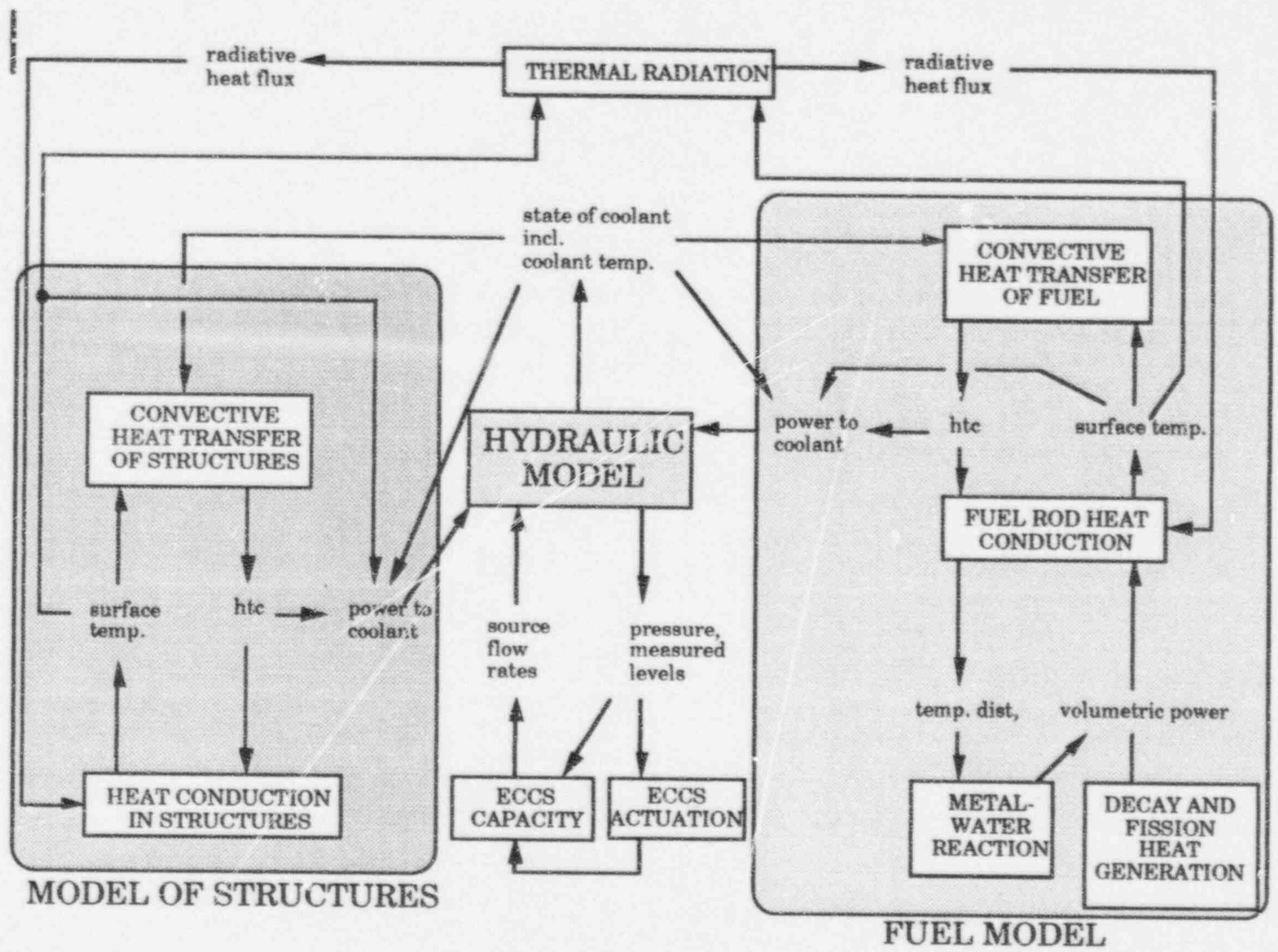

Figure A.1 Interaction of Models in GOBLIN-EM

APPENDIX B: RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

B.1 INTRODUCTION

This appendix contains responses to the NRC Requests for Additional Information regarding Reference B1 which was transmitted to ABB by the NRC letter identified in Reference B2. Appendix C provides additional details of the Example 2 calculations presented in Chapter 6 of the main body.

B.2 QUESTIONS AND RESPONSES

NRC Question B1

It should be noted that the allowable stress intensities given on pg. 11 are for elastic analysis; what allowables are used if the component is analyzed plastically?

ABB Response to Question B1

ABB BWR fuel always has been shown to meet design requirements using elastic analysis, which is acceptable under Section F-1322 of Appendix F (Reference B3). In the event that an elastic-plastic analysis were required to demonstrate compliance with the design bases for a specific application, the analysis would be submitted to the NRC under separate application.

NRC Question B2

Why is fatigue evaluation not addressed for the component for the combination of seismic and LOCA loads?

ABB Response to Question B2

The main body provides the methodology for evaluating BWR fuel subjected to postulated Safe Shutdown Earthquake (SSE) and Loss of Coolant Accident (LOCA) events. Therefore Level D Service Limits apply and a fatigue evaluation is not required. [Proprietary Information Deleted]

In addition, a fatigue evaluation is performed for normal operating loads as required by Reference B4. This evaluation is discussed in CENPD-287-P-A (Reference B5).

NRC Question B3

On pg. 12, a statement is made that buckling of the channels is precluded when the maximum calculated stresses do not exceed the allowable stress intensities. A separate buckling evaluation, such as is prescribed in F-1331.5 of Appendix F of the ASME Code Section III, is needed to assure that buckling is precluded. Is a separate buckling evaluation done?

ABB Response to Question B3

Testing of the channel is performed to demonstrate that the channel will not buckle when subjected to seismic plus LOCA loads and deflections. The seismic plus LOCA loads do not cause significant compressive (axial) loads on the fuel channel. The potential collapse

mode of concern is local plate buckling caused by excessive lateral deflection.

The channel acceptance criteria given in Section 4.2.2 of the main body is clarified here. It is restated:

"The channel deformation design acceptance criteria are:

(1) The maximum stresses calculated analytically do not exceed the allowable stress intensities.

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

(3) The maximum stresses and deflections calculated do not exceed limits for channel weld strength and channel buckling based on test loads and deflections representative of a seismic plus LOCA event."

The channel weld strength and channel buckling limits are typically measured in the test noted in Section 6.1.2.2 of the main body. [Proprietary Information Deleted]

<u>NRC Question B4</u>

The stress limits listed on pg. 11 infer that the analysis performed is elastic. If stresses in the channel structure exceed the yield stress, then the deflections would not be calculated accurately. An accurate determination of the deflection would in this case require plastic analysis. An accurate calculation for the deflection would be needed to assure safe insertion of the control rods. Is it assured?

<u>ABB Response to Question B4</u>

In the methodology described in the main body, an elastic analysis is performed, which assures safe insertion of the control rods. It is expected that the calculated primary stresses in the channel will remain elastic so that the calculated deflections are accurate. [Proprietary Information Deleted]

<u>NRC Question B5</u>

On pg. 18 It is stated that if the fuel assembly lifts it will then impact against the core support plate. It is stated that the resulting fuel vertical load would range from 2 to 5 g. It is not explained how this loading was determined. If the fuel assembly and the core support plate are very stiff structures, it is conceivable that the impact loading could be significantly higher than 2 to 5 g. This would, of course, depend on

the energy of the fuel assembly at impact. How was the impact deceleration determined?

ABB Response to Question B5

The range of peak vertical loads with fuel assembly lift given in Section 5.1.1.4 of the main body, are from a census of several plant licensing base analyses as reported in the plant specific Final Safety Analysis Report. The FSAR reported peak vertical loads were determined by the plant original equipment supplier for the operating utility. For example, Reference B6 reports a fuel assembly peak vertical fuel assembly load of 4.9 g, using the methodology of Reference B7.

[Proprietary Information Deleted]

NRC Question B6

It is stated on pg. 27 that fast neutron irradiation has little effect on the yield and ultimate strengths of the stainless steel and increases the yield and ultimate strengths of the Zircaloy. Hence, it is concluded that the use of unirradiated properties is conservative. This conclusion, however, does not account for possible effects of irradiation on the material ductility. The ASME stress limits given on pg. 11 allow the material to be stressed beyond yield on the basis that the material is ductile. To use these stress limits, it must be assured that the material maintains this ductility when irradiated. Has it been established that the material will maintain ductility at the irradiation levels experienced?

ABB Response to Question B6

The methodology described in the main body will ensure that irradiated fuel assemblies will not fracture if subjected to SSE plus LOCA loads. For example, a SVEA-96 fuel assembly is composed of a Zircaloy-4 channel, Zircaloy-2 fuel rods, and stainless steel castings for the bundle and channel end pieces. The stainless steel castings are of relatively solid design and have very large strength margins. Therefore, the focus of this response concerns the Zircaloy components.

Calculated fuel assembly stresses are compared to stress limits based on minimum material properties for unirradiated material. This is conservative because irradiation of Zircaloy significantly increases its yield stress and ultimate strength. The yield and ultimate tensile strengths are expected to increase by factors of at least 2 and 1.5, respectively, at relatively low irradiation values.

In example, Table B6-1 summarized measured BWR fully recrystalized Zircaloy-2 material data as a function of fast fluence and strain rate (from Reference B16). Virtually all of the increase in yield and

ultimate strength occurs early in the fast fluence exposure, so that it is reasonable to extend the data to fast fluences greater than 1 x 10^{26} n/m^2. This extrapolation is supported by the data reported in Reference B17 (Yasuda, et al.) which indicate most of the increase in yield strength of fully recrystalized annealed Zircaloy-2 occurs below a fast fluence exposure of 10^{24} n/m^2, and no significant change in the fluence dependency is exhibited up to 0.4 x 10^{26} n/m^2. These data support the qualification of the Zircaloy material models discussed in Appendix A of Reference B8. The data confirm that the reduction in material ductility with irradiation is not of concern (see Figure B6-1), since the irradiated ultimate strengths far exceed the acceptance stress intensity limit established from the lower bound of the unirradiated strength (given in Table 6.1C of this report).

<u>NRC Question B7</u>

It is stated on pg. 27 that the test load applied to determine stress for the spacer grid and channel weld was cyclic force. Is the number of load cycles applied in the test representative of the number of cycles expected in the actual installation?

<u>ABB Response to Question B7</u>

Yes, the number of load cycles applied in the test bounds the number of cycles expected at the actual plant. The seismic spacer grid and channel weld test is performed with a greater number of load cycles than what is expected during an SSE plus LOCA event. [Proprietary Information Deleted]

<u>NRC Question B8</u>

On pg. 29, has the fuel rod performance code VIK been approved by the NRC?

<u>ABB Response to Question B8</u>

The VIK-II code is described in CENPD-285-P-A (Reference B8). It is the same computer code that is used for the applications described in the main body.

<u>NRC Question B9</u>

On pg. 29, the methods used to combine stresses from different loadings on the fuel rods are described. Since deformation on the channel is important, a description of the method used to determine the total deflection should also be provided. This should include the methods used if the structure goes plastic. What is the method used to determine the total deflection?

ABB Response to Question B9

The method used to calculate the total deflection of the channel is described in Section 5.1.4.1 of the main body. [Proprietary Information Deleted]

NRC Question B10

On pg 15, sect. 5.1.1.1, Discussion: Does the model of fig. 5.2 include the confined fluid?

ABB Response to Question B10

Yes, the confined fluid is included in the model shown in Figure 5.2. The model is discussed in more detail in Section 6.3.1.1 of the main body. [Proprietary Information Deleted]

NRC Question B11

On pg 31, sect, 5.2.1, Methodology, steps (1), (2) & (3):

(A) How is bounding arrived at in step (1)?

(B) What is the significance of not going through step (2) in all cases, irrespective of the results from step (1)?

(C) More information is necessary to establish the validity of step (3), Please provide.

ABB Response to Question B11

Section 5.2.1 of the main body provides the methodology for demonstrating that ABB BWR reload fuel assemblies will remain engaged in the lower support structure following a postulated SSE/LOCA event. [Proprietary Information Deleted]

ABB Response to Question B11, Item (A)

[Proprietary Information Deleted]

ABB Response to Question B11, Item (B)

[Proprietary Information Deleted]

ABB Response to Question B11, Item (C)

[Proprietary Information Deleted]

NRC Question B12

On pg. 31, sect. 5.2.1, Discussion, step (3):

A technical justification for step (3) is required.

ABB Response to Question B12

[Proprietary Information Deleted]

NRC Question B13

On pg. 58, sect. 6.3.2.1, GOBLIN/DRAGON Model:

Are the results of the three channel and one channel representations significantly different? How do they compare?

ABB Response to Question B13

[Proprietary Information Deleted]

NRC Question B14

On pg. 69, sect. [SVEA-96 Fuel Assembly Vertical Acceleration]:

*The significance of the last * paragraph is not evident. Clarification is required.*

ABB Response to Question B14

This question refers to the fuel assembly lift calculation performed for Example 2, in Section 6.4 of the main body. [Proprietary Information Deleted]

NRC Question B15

On pg. 77, sect. 6.5.1.2, Resident Fuel Assembly Seismic response, (also applies to sects. 6.5.1.3 & 6.5.2):

Further explanation is required to establish the validity of the conclusion(s).

ABB Response to Question B15

Section 6.5 of the main body discusses the methodology to be used when the information concerning seismic and LOCA excitation for a

* The original NRC question had a typographical error "elastic" should be "last".

specific plant is limited. Additional explanation of this methodology is provided below in response to this question.

[Proprietary Information Deleted]

NRC Question B16

The integrity evaluation of a mixed core is not addressed. Is there a possibility of mixed core being used? If so, how is it integrity established?

ABB Response to Question B16

Most operating reactors have a core composed of a mix of fuel designs, either by the same vendor or from several vendors. The seismic/LOCA evaluation methodology described in the main body considers this fact when performing an evaluation for a specific fuel design in a particular plant.

[Proprietary Information Deleted]

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B4. ASME Boiler and Pressure Vessel Code, Section III, Part NB, 1992 Edition

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B6. River Bend Station Updated Safety Analysis Report, Section 3.9, Table 3.9B-2aa, August 1987.

B7. BWR Fuel Assembly Evaluation of Combined Safe Shutdown (SSE) and Loss-of-Coolant Accident (LOCA) Loadings (Amendment No. 3), GE Report NEDO-21175-3-A, October 1984.

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B9. ASME Boiler and Pressure Vessel Code, Section III, Appendix II-1520, 1992 Edition

B10. U.S. NRC Standard Review Plan Section 4.2, Appendix A, NUREG-0800, July 1981.

B11. Y. Sasaki, Y. Sasaki, H. Niwa, "Dynamic Analysis of Fuel Elements in Boiling Water Reactor," Vol. D, Paper 4-7, 4th Inter. Conf. on Structural Mechanics in Reactor Technology, 1977.

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B13. Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification, ABB Report RPB-90-93-P-A (proprietary), RPB-90-91-NP-A (non-proprietary), October 1991.

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B16. K. Petterson and E. Hellstrand, "Mechanical Properties of Zircaloy 2 After a High-Dose Irradiation", Studsvik/MZ - 78/110 (1978).

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B18. ANSYS Engineering Analysis System Users Manual, Rev. 4.4, Swanson Analysis Systems Inc., Vol. I, Houston, Pa, May 1989.

TABLE B6-1

Proprietary Information Deleted

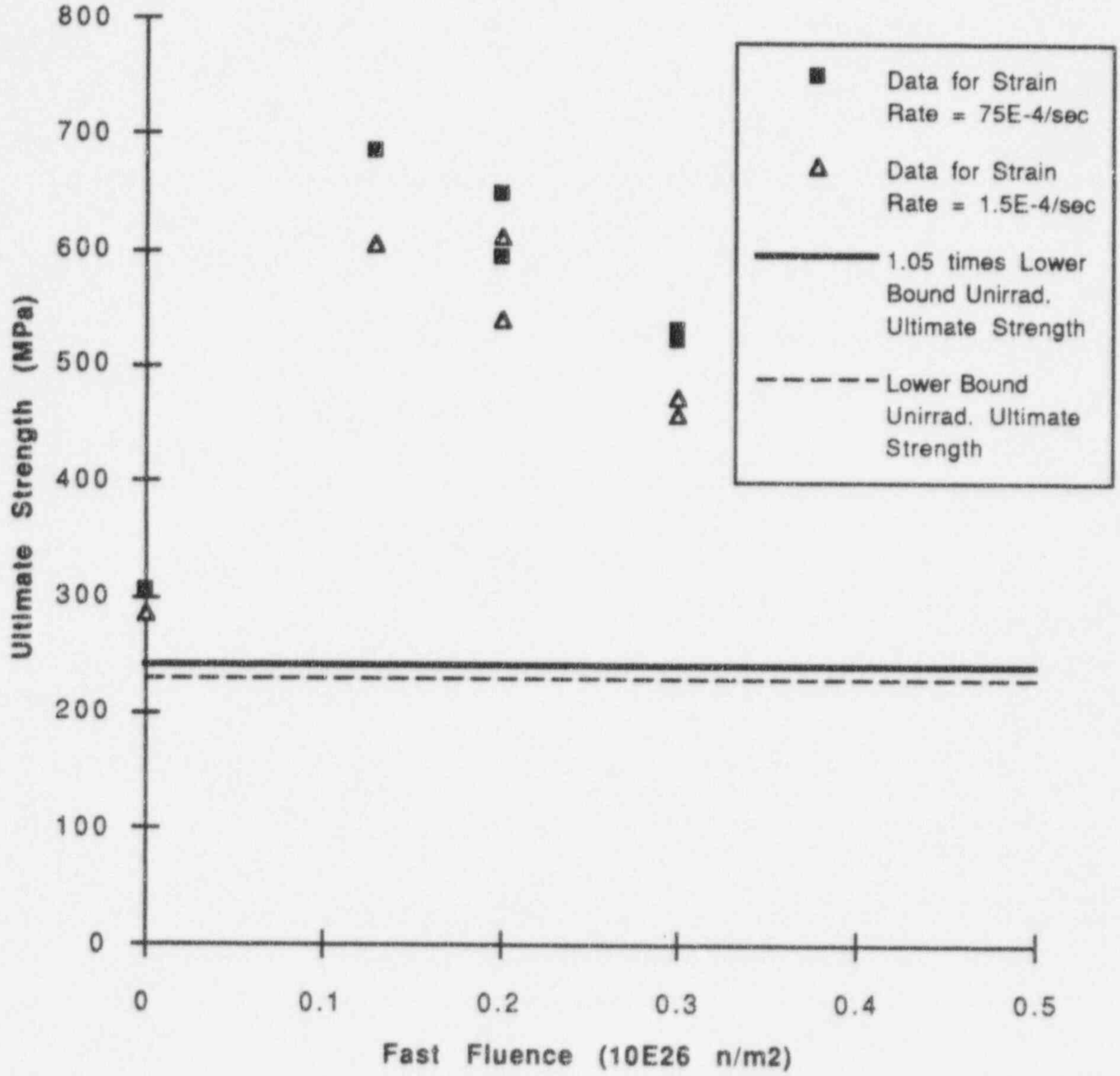

Figure B6-1 Comparison of BWR Fully Recrystalized Annealed Zircaloy-2 Ultimate Strength Acceptance Limit with Measured Data.

FIGURE B12-1

Proprietary Information Deleted

FIGURE B13-1 AND FIGURE B13-2

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APPENDIX C: DETAILED DATA FOR EXAMPLE 2

Table C-1 and Figures C-1 and C-2 summarize the sources of data used in the illustrative Example 2 of Chapter 6 in the main body of this report.

TABLE C-1

Proprietary Information Deleted

FIGURE C-1 AND FIGURE C-2

Proprietary Information Deleted

ABB Combustion Engineering Nuclear Operations
Combustion Engineering, Inc.
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