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REPORT TITLE: **Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions**

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WCAP - 15030-NP-A

Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions

Westinghouse Energy Systems

WCAP-15030-NP-A

Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 10, 1998

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WOG PROJECT OFFICE

Mr. Lou Liberatori, Chairman
Westinghouse Owners Group Steering Committee
Indian Point Unit 2
Broadway & Bleakley Ave.
Buchanan, NY 10511

SUBJECT: SAFETY EVALUATION OF TOPICAL REPORT WCAP-15029 "WESTINGHOUSE METHODOLOGY FOR EVALUATING THE ACCEPTABILITY OF BAFFLE-FORMER-BARREL BOLTING DISTRIBUTIONS UNDER FAULTED LOAD CONDITIONS" (TAC NO. MA1152)

The staff has completed its review of the subject report requested by Westinghouse Owners Group (WOG) by letter of June 19, 1998. The staff has found that this report is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and in the associated NRC safety evaluation, which is enclosed. The safety evaluation defines the basis for acceptance of the report.

In response to reported foreign experience with baffle bolt failures and concerns of potential degradation of baffle barrel bolts in domestic plants, the WOG proposed that licensees perform corrective action inspections and replacement of baffle bolts under the provisions of 10 CFR 50.59 for two lead plants (2 & 3 loop). In determining the applicability of the 10 CFR 50.59 licensing approach, WOG requested NRC review and approval of WCAP-15029, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," and WCAP-14748/9, "Justification of Increasing Postulated Break Opening Time." The latter WCAP report invokes the leak-before-break (LBB) concept and provides a justification for break-opening times greater than one millisecond. The LBB concept and break-opening times greater than one millisecond are applied, by reference, in the methodology guidance provided in WCAP-15029. WCAP-14748/9 was reviewed separately and the staff's safety evaluation was transmitted to WOG on October 1, 1998. Consequently, it is not included in this safety evaluation of WCAP-15029. Further, WOG requested that WCAP-9735 Rev.2, "Multiflex 3.0, A Fortran IV Computer Program for Analyzing Thermal Hydraulic Structural System Dynamics-Advanced Beam Model," which was previously submitted for staff review and approval, be withdrawn from review under the NRC licensing topical report program for referencing in licensing actions. Multiflex 3.0 is also applied in WCAP-15029 methodology guidance, by reference, but is not evaluated in this safety evaluation.

The staff will not repeat its review of the matters described in the WOG Topical Report WCAP-15029, when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. In accordance with procedures established in NUREG-0390, the NRC requests that WOG publish accepted versions of the submittal, proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed safety evaluation

L. Liberatori -2- November 10, 1998

between the title page and the abstract and an -A (designating accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusion that the submittal is acceptable are invalidated, WOG and/or the applicant referencing the topical report will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Sincerely,

Thomas H. Essig, Acting Chief
Generic Issues and Environmental Branch
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Safety Evaluation

cc w/enc: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WCAP-15029, "WESTINGHOUSE METHODOLOGY FOR EVALUATING THE ACCEPTABILITY OF BAFFLE-FORMER-BARREL BOLTING DISTRIBUTIONS UNDER FAULTED LOAD CONDITIONS"

INTRODUCTION:

Over the past year, the NRC and the Westinghouse Owners Group (WOG) have been interacting on the issue of potential failure of baffle-former-barrel bolting. This work is in response to reported foreign experience with baffle bolt failure. Baffle bolt cracking experience in PWRs is described in NRC Information Notice 98-11, *Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants,* dated March 25, 1998.

By letter dated June 19, 1998, (Reference 1) WOG submitted its assessment of the safety significance impact of potential baffle bolt cracking on domestic Westinghouse designed reactor vessel internals. The letter indicated that a large number of the later designed domestic Westinghouse plants contain reactor vessel internals design features that reduce the magnitude of the bolt loads induced during a faulted event to comparatively small values. These design features are described as bolt cooling holes and baffle plate pressure relief holes which minimize the potential for age related effects on the bolts to significantly affect plant safety and operation. For the remaining Westinghouse domestic plants, WOG indicated there is a large variation in plant features and a significant variation among the plants in the potential for bolt degradation and consequences of bolt degradation on plant safety and operation. The assessment was based on both a deterministic and risk-informed evaluation. WOG concluded that its assessment confirmed its belief that the issue is not an immediate safety concern, and that it is appropriate to treat the baffle bolt cracking as an aging management issue.

In its letter dated June 19, 1998, WOG proposed that licensees perform corrective action inspections and replacement of baffle bolts under the provisions of 10 CFR 50.59 for two lead plants (2 & 3 loop). In determining the applicability of the 10 CFR 50.59 licensing approach, WOG requested NRC review and approval of WCAP-15029, *Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,* and WCAP-14748/9, *Justification of Increasing Postulated Break Opening Time.* The latter WCAP report invokes the leak-before-break (LBB) concept and provides a justification for break-opening times greater than one millisecond. The LBB concept and break-opening times greater than one millisecond are applied, by reference, in the methodology guidance provided in WCAP-15029. WCAP-14748/9 was reviewed separately and the staff's SE was transmitted to WOG on October 1, 1998 (Reference 2). Consequently, it is not

Enclosure

included in this SE of WCAP-15029. Further, WOG requested that WCAP-9735 Rev.2, *Multiflex 3.0, A Fortran IV Computer Program for Analyzing Thermal Hydraulic Structural System Dynamics-Advanced Beam Model*, which was previously submitted for staff review and approval, be withdrawn from review under the NRC licensing topical report program for referencing in licensing actions. Multiflex 3.0 is also applied in WCAP 15029 methodology guidance, by reference, but is not evaluated in this SE. Evaluation of the Multiflex 3.0 methodology is not a requisite for concluding that WCAP 15029 is acceptable.

DISCUSSION

The WCAP-15029 topical report provides systematic guidance in the form of a flow diagram and associated methodology narrative that identifies the sequence of analyses for evaluating the acceptability of baffle bolts and bolting distributions under faulted conditions. The methodology applies referenced prescribed analytical methods and assumptions to determine acceptable baffle bolt loading, bolting distributions, allowable stress limits for certain bolt materials, and options to address control rod insertion ability and core coolabilty criteria. The topical report indicates that it supersedes a referenced and previously NRC-approved methodology (Reference 4), that demonstrated safe plant operation during faulted conditions, by broadening the previous methodology to address the safe operation of Westinghouse plants with reduced baffle-former-barrel bolting. Further, the report indicates that the broadened methodology consists of refinements and enhancements of previously used analyses, and that the prescribed analyses methods, assumptions and options use realistically simulated faulted conditions while retaining an adequate measure of conservatism.

The broadened methodology applies the LBB concept, break-opening times greater than one millisecond, and the Multiflex 3.0 program with its advanced fluid-structure interaction simulations in the core barrel baffle-former region. The methodology incorporates the application of the best-estimate WCOBRA/TRACK code on intermediate pipe break sizes to establish the depressurization transient that is used to demonstrate that two-phase loads are less limiting than single phase loads for smaller breaks under the LBB concept. The validity of the demonstration is left to the licensee in the application of its plant conditions and current licensing bases in using the methodology.

LOCA initiation gives rise to three types of loads: 1) Acoustic depression waves which traverse the reactor coolant system from the location of the break at the speed of sound. Peak loads from these acoustic pulses occur at about 50 msec after break initiation; 2) Mass flow induced loads from fluid motion caused by the depressurization forces. These loads peak between 100 and 200 msec and could be limiting for core components subject to cross flows following depressurization; and 3) two phase flow frictional and pressure loads. These loads are significant in the bypass region where high pressure differentials are created during depressurization and appear between 5 and 50 msec after LOCA initiation. The first and second loads are estimated using the Multiflex 3.0 code. The third load is covered below.

The logic of the methodology depicted in the WCAP-15029 flow chart is an appropriate process for evaluating acceptable bolting loads and distributions. The Multiflex 3.0 program is described as a more sophisticated analysis tool for LOCA hydraulic force calculations than the currently

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approved version, Multiflex 1.0. WCAP-15029 indicates that the Multiflex 3.0 program enhancements of Multiflex 1.0 include: the use of a two dimensional flow network to represent the vessel downcomer region in lieu of a collection of one dimensional parallel pipes; the allowance for nonlinear boundary conditions at the vessel and downcomer interface at the radial keys and the upper core barrel flange in lieu of simplified linear boundary conditions; and the allowance for vessel motion in lieu of rigid vessel assumptions. WCAP-15029 indicates that these modifications are included in the Multiflex 3.0 program that is used to estimate the LOCA hydraulic forces on the vessel and consequential forces induced on the fuel and reactor vessel internal structures. The staff concurs with the WOG that Multiflex 3.0 provides a more accurate and realistic modeling approach. On this basis, and considering that Multiflex 3.0 is based on the previously approved Multiflex 1.0, the staff considers the application of Multiflex 3.0 with the WCAP 15029 methodology reasonable and acceptable.

The core structural integrity during a LOCA transient or a seismic event is analyzed using the "Verification Testing and Analysis of 17x17 Optimized Fuel Assembly" code which has been approved by the NRC and is described in Reference 4. However, the two phase loads on the baffle plates resulting from rapid depressurization are not normally calculated during LOCA analyses. Early Westinghouse calculations using the SATAN code, Reference 5, have shown that such loads were, in general, comparable to the single phase acoustic loads. These loads depend on the reactor vessel depressurization rate, which in turn depends on the size, location and type of the break. In the proposed methodology, a single depressurization calculation will be performed for each plant type (two- three- or four-loop) to be bounded. If, as expected, the two phase loads are not limiting, they will be considered as having been adequately addressed. Should the two phase loads turn out to be comparable with the acoustic loads then a more detailed calculation will be required.

The break sizes under consideration range from the accumulator piping (about 10 inches in diameter) to the pressurizer surge line (about 14 inches in diameter). The code chosen for these calculations is WCOBRA/TRAC. The code has been approved for best estimate large break LOCA calculations for three- and four-loop plants, Reference 6. A two loop (upper plenum injection) best estimate version, has been submitted for review, Reference 7. The staff has completed its review of WCOBRA/TRAC as a best estimate LOCA code for the two loop plant baffle load application, and the SE is being issued separately. The code has been used for breaks ranging from split breaks (less than 60 inch^2) to double ended guillotine breaks with a Moody coefficient of 1.0. A finite element representation is shown (octant geometry) which accounts for the baffle-former-barrel, and baffle to baffle bolts and the former holes for the bypass flow. A specific bolt configuration is used for each case. The acceptability of that configuration depends on the resulting bolt loadings and allowable stress and deflection limits for the plant-specific application.

The use of best estimate WCOBRA/TRAC code for the estimation of two-phase loads on the baffle bolts contained in the methodology is acceptable provided that: 1) the limiting baffle bolt loading will be determined by analysis for a class of plants and a specific break; 2) the noding to be used in the representation of the loading is demonstrated to be adequate by performing nodalization sensitivity studies or by some other acceptable methodology.

The methodology developed by WOG for Westinghouse plants is proposed for use in determining the acceptability of replacement bolts and bolting distributions and, in part, to assess existing bolting using appropriate irradiated bolt material properties. The bolt replacement activities are subject to the results of plant inspections for baffle bolt cracking degradation. The topical report references the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, for materials and formulations for determining allowable stress limits to be used in selecting replacement bolts for normal, upset and faulted conditions. The ASME B&PV Code does not provide allowable stress limits for irradiated materials. The WCAP report contains proposed irradiated stress allowable limits for use in determining the allowable primary membrane and primary membrane plus primary bending stress limits for baffle bolting under faulted conditions in accordance with the formulation requirements of Appendix F of Section III of the ASME B&PV Code.

It is not evident that this approach is entirely justifiable. Implicit in the ASME Code stress allowable limit is the fact that the bolting materials should exhibit ductility that provides a strain margin between yield and ultimate strength. For irradiated austenitic stainless steel, the available data indicate a severe loss of ductility, with the yield strength and the ultimate strength in some cases being coincident. It is not clear that the stress allowable limits proposed for irradiated stainless steel account for the lack of ductility of the material.

The inservice inspection (ISI) methods used for inspecting the baffle bolts will have certain sensitivity and probability of detection (POD) characteristics. In using irradiated tensile strength for calculating safety margins, the bolt load-bearing cross-section area should be reduced based upon the sensitivity of the ISI methods and the potential for degraded bolts to escape detection by ISI. In addition, since the available data for irradiated stainless steel indicates that the ductility (and presumably the fracture toughness) of irradiated bolts is severely degraded, a fracture mechanics approach would be necessary to demonstrate that the degraded bolts exhibit adequate toughness with a postulated flaw size undetected by ISI.

The report indicates these proposed stress allowable limits are based on yield strength and ultimate strength values obtained from irradiated bolting materials. The report indicates that the selected values are conservative considering the source irradiated materials had significantly lower fluence levels than bolts in the baffle region. The conservatism is based on experience which demonstrates that for certain materials the yield strength and ultimate strength increase with fluence level. The determination of adequate conservatism of bolting material properties and characteristics, resides with the licensee in applying appropriate stress and deflection limits to the baffle assembly under faulted conditions. This determination should include consideration of the subject plant's historical operating conditions and current licensing bases. The determination should be based on conservative yield strength and ultimate strength values representative of the plant's existing bolting material properties.

Although the methodology provides systematic guidance for an acceptable logical analytical approach for determining baffle bolt loading under faulted conditions, it has not identified any requirements for a baffle bolt post-replacement monitoring program. The implementation of an inspection program is appropriate for baffle bolt cracking aging management. The inspection program must be capable of detecting baffle bolt cracking and degradation consistent with the

results of the plant's bolt inspection and replacement activities and applicable industry experience. Further, the inspection program should implement a root-cause evaluation of any replacement baffle bolt degradation discovery.

CONCLUSION

The use of the WCAP-15029 methodology guidance is acceptable in accordance with the following limitations:

1. The bolt loading should be determined by analysis for a class of plants and a specific break;

2. The noding to be used in the representation of the loading is demonstrated to be adequate by performing nodalization sensitivity studies or by some other acceptable methodology;

3. The methodology should not be used to assess existing bolting without demonstration of adequate conservatism in projected bolting material properties (i.e., yield and ultimate strength) to ensure that sufficient ductility is present in existing irradiated stainless steel bolting materials;

4. The use of the methodology for existing irradiated stainless steel bolting should account for limitations in available ISI methods with regard to the probability of detection characteristics;

Finally, in consideration of the WOG assessment and conclusion that the baffle bolt issue is not an immediate safety concern and that it is appropriate to treat baffle former bolt degradation as an aging management issue, subsequent to replacement of baffle bolts, licensees are expected to develop an appropriate inspection monitoring and aging management program for baffle bolting.

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REFERENCES

1. Westinghouse Owners Group Letter, dated June 19, 1998, "Baffle Barrel Bolt Program-Response to April 9, 1998, NRC Letter (MUHP5135)," by Louis F. Liberatori, Jr.

2. NRC letter, dated October 1, 1998, "Safety Evaluation Related to Topical Report WCAP-14748/49-Justification for Increasing Break Opening Times in Westinghouse PWRs," to Lou Liberatori, Chairman, WOG Steering Committee

3. WCAP-15029, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions" by P. E. Schwirian, et al., Westinghouse Electric Company, 1998

4. WCAP-9401PA, "Verification Testing and Analysis of 17x17 Optimized Fuel Assembly," by M. D. Beaumont, et al., Westinghouse Electric Company, August 1981

5. WCAP-8302, "SATAN-IV Program: Comprehensive Space Time Analysis of Loss of Coolant," by F.M. Bordelon, et al., Westinghouse Electric Company

6. WCAP-12945P, Volume 4, "Code Qualification Document for Best Estimate LOCA Analysis Volume IV: Assessment of Uncertainty," by S. B. Bajorek, et al., Westinghouse Electric Company, 1993

7. WCAP-14449P "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," by S. I. Dederer, et al., Westinghouse Electric Corporation, 1993

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Date: November 10, 1998

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EXECUTIVE SUMMARY

Reference 1-1 describes the U.S. NRC approved analyses methodology for demonstrating safe plant operation of Westinghouse pressurized water reactors (PWRs) during faulted conditions. Inherent in this methodology is the assumption that the bolting in the baffle-former-barrel region surrounding the core is in an as-designed condition. Ultrasonic (UT) measurements in a number of foreign plants, however, indicate that some bolt cracking is possible. The purpose of the present report is to broaden the analysis methodology of Reference 1-1 to address this possibility, and provide a comprehensive analysis methodology for assuring safe plant operation of Westinghouse PWRs with acceptably reduced baffle-former-barrel bolting.

Section 1 summarizes the background of the bolt-cracking issue and the programs which have been pursued by the Westinghouse Owner's Group (WOG) to address it in the United States. In 1997, the WOG initiated the Baffle-Barrel-Bolting (BBB or B**3) Program, which incorporates the experiences gained from previous activities into a comprehensive new, multifaceted program. An important aspect of this program is the use of analysis methods and assumptions which realistically simulate faulted conditions while retaining an adequate measure of conservatism. Among these are the use of the leak-before-break (LBB) technology, break-opening-times greater than one millisecond, the MULTIFLEX 3.0 program, and advanced fluid-structure interaction simulations of the structure and fluid in the baffle-former-barrel region. Approval of these, analysis refinements, described in the current report, is requested of the U.S. NRC.

The overall acceptable bolting analysis methodology is described in Section 2. A flow chart (Figure 2-1) outlines the overall analyses procedure, which includes consideration of faulted events, normal/upset (N/U) conditions, options for demonstrating core coolability, and acceptance criteria. The proposed improvements in analysis methodology and their verification are discussed in detail in this section and in the appendix. Those aspects of the overall analysis procedure which are common to the Reference 1-1 analysis procedure are also identified for completeness. This methodology can be used to determine replacement bolt locations and number, or to assess existing bolt configurations, using the appropriate bolt material properties in both cases.

Because normal/upset performance may be affected by revised bolting, such conditions are also analyzed. These include evaluations of low cycle fatigue (thermal cycling), high cycle fatigue (flow-induced vibration) for the bolting, baffle-jetting and core bypass flow. The analyses which deal with these topics are outlined in Section 3. These analyses are performed to assure that the revised bolting patterns satisfy the established design requirements for N/U conditions.

Section 4 deals with the acceptance criteria and with options for demonstrating core coolability with revised bolting distributions. For the bolts themselves, the acceptance criteria are established stress allowable criteria. For the fuel, acceptable demonstrations must be made of fuel rod integrity, control rod insertability, and core coolability. For the latter, three options are proposed: a) fuel grid impact loads below grid strength allowables, b) fuel grid impact loads above allowables only in peripheral fuel assemblies, and c) fuel grid impact loads above allowables in interior fuel assemblies. The option chosen for a given plant or plant grouping determines the type of core coolability analysis required.

1. INTRODUCTION

In Westinghouse style, Pressurized Water Reactors (PWRs), the reactor core is surrounded by a series of vertical plates, called baffle plates, which form a boundary for the flow of reactor coolant and provide lateral support for the fuel assemblies. These plates are supported by horizontal supports, called former plates, which are bolted to the baffle plates. The former plates are in turn supported by the cylindrical core barrel. The bolts which attach the baffle plates to the formers are referred to as "baffle-former bolts" or "baffle bolts".

Since 1989, indications of cracking of the 316 CW stainless steel baffle-former bolts have been observed in a number of plants in Europe. These indications were detected by ultrasonic inspection. More recent inspections have revealed additional bolt cracking indications in some plants. In addition to the ultrasonic examinations, bolts have been removed and metallographic examinations have been performed in order to investigate the cause of the degradation. It has been established that the cracking is intergranular in nature.

European and Westinghouse domestic plants have many, similar characteristics, and may therefore possibly experience similar aging effects. Thus, the Westinghouse Owner's Group (WOG) Baffle-Former Bolt (BFB) Program was initiated. The objectives of this program were:

- To investigate the potential for Westinghouse plants to experience a similar aging effect.
- To evaluate the potential safety considerations which could arise as a result of baffle-former bolt cracking.
- To implement a coordinated program to manage this potential aging effect for Westinghouse reactors.

Among the efforts pursued in the Baffle-Former-Bolt Program was a decision analysis to evaluate strategies for managing the potential bolt cracking issue. One of the recommendations of the decision analysis was that additional analysis should be pursued using analyses methods which reduce the conservatism inherent in the current licensing basis approach, and to pursue U.S. NRC approval of these methods for use in demonstrating acceptability for reduced bolting distributions.

In June, 1997, the Westinghouse Owners Group authorized the Baffle-Barrel-Bolting (BBB or B**3) Program, one task of which is to develop an overall analysis methodology for evaluating the acceptability of reduced bolting distributions in the baffle-former-barrel region. Those distributions determined to be acceptable can be used to evaluate the acceptability of existing bolt performance or to provide guidance for the locations and number of replacement bolts. The purpose of this document is to describe and demonstrate the validity of this analysis methodology for the most limiting faulted conditions. For many aspects of this analysis methodology development, the U.S. NRC-approved analysis techniques described in Reference 1-1 will be employed and identified. Alternate or additional analysis techniques

utilized for the determination of acceptable bolting distributions will be presented in more detail.

The primary differences between the methodology presented in Reference 1-1 and in the present document lie in the treatment of the loss-of-coolant-accident (LOCA) event. The LOCA analysis methodology described herein differs from the current established analysis methodology primarily in the use of more sophisticated analysis tools relying upon more realistic bounding assumptions. As with current analyses of the reactor vessel and internals, leak-before-break (LBB) exclusions allowed under GDC-4 will be credited when selecting the break location and size. However, instead of relying upon the unrealistic but conservative assumption of a one millisecond break opening time which has been applied to the current approved MULTIFLEX 1.0 (Reference 1-2) analysis tool, a more realistic but still conservative break opening time greater than one millisecond will be selected (Reference 1-3). Further, the MULTIFLEX 1.0 analysis tool for the LOCA hydraulic forces calculation is to be replaced by the more sophisticated MULTIFLEX 3.0 (Reference 1-4) analysis tool. MULTIFLEX 3.0 uses a two dimensional network to represent the vessel downcomer, rather than a collection of one dimensional parallel pipes. MULTIFLEX 3.0 also allows for non-linear boundary conditions at the vessel and downcomer junctions at the radial keys and upper core barrel flange, rather than the simplified linear boundary conditions imposed in the MULTIFLEX 1.0 model. MULTIFLEX 3.0 allows for vessel motion, rather than the rigid vessel assumption used in the MULTIFLEX 1.0 methodology. The combination of these analytical methods will result in more realistic bounding calculations of the LOCA hydraulic forces on the vessel for purposes of calculating the accident analysis consequences on fuel and vessel internals.

MULTIFLEX 3.0 is intended primarily to analyze the single phase blowdown transient over the first five hundred milliseconds. For enclosed vessel subregions - primarily the baffle-former-barrel region - two-phase depressurization loads may also occur later in the transient as the system pressure reaches saturation and boiling commences. These two-phase loads were previously addressed in current plant designs using depressurization transients calculated using the approved appendix K analysis tools such as SATAN (Reference 1-5) for large double ended breaks. Since that time, LBB exclusions on the main loop piping have reduced the break size under consideration to well below those typically analyzed using appendix K large break LOCA analysis tools like SATAN, but above those typically analyzed using appendix K small break LOCA analysis tools like NOTRUMP (Reference 1-6). To ensure a technically accurate but conservative calculation, the approved best-estimate WCOBRA/TRAC (References 1-7 and 1-8) tool is to be used on these intermediate break sizes to establish the depressurization transient. The results of this depressurization transient are used to demonstrate that two-phase loads are significantly less limiting than single phase loads for the smaller break sizes considered under LBB.

While MULTIFLEX 3.0 captures the effects of fluid-structure interaction in the region of the downcomer, the calculation of hydraulic conditions inside the baffle former region assumes rigid boundaries. To account for the effects of fluid-structure interaction in the baffle-former region, a finite element model has been developed which uses the MULTIFLEX 3.0 rigid boundary hydraulic pressures as forcing functions. Fluid acoustical and fluid-structure interaction simulations transmit the effects of these forcing functions to the baffle-former barrel

region. This model allows for the effects of missing or broken bolts in calculating baffle plate deflections and bolt stresses for the candidate bolting distributions.

Finally, the combined effects of core plate motions and baffle deflections are used to calculate LOCA loads on fuel assemblies and, if applicable, combined by the square root of sum of squares (SRSS) technique with seismic loads. The consequences are then evaluated relative to fuel acceptance criteria: a) fuel rods must remain intact such that fuel pellets are not allowed to escape where they could achieve a configuration in which a core coolable geometry cannot be demonstrated, b) control rod guide tubes must not be deformed to the point at which control rod insertion cannot be demonstrated where it is credited in accident analysis consequences, and c) any calculated fuel grid crushing would have to be dispositioned relative to accident analysis consequences through the appropriate 10 CFR 50.46 LOCA analysis tool using either approved appendix K LOCA analysis methods, or approved best-estimate LOCA analysis methodology.

2. FAULTED (LOCA/SEISMIC) LOAD ANALYSIS METHODOLOGY

Reference 1-1, WCAP-9401, describes the U.S. NRC-accepted Westinghouse analytical methodology for assuring safe plant operation during faulted LOCA and seismic (SSE) conditions. This methodology was developed and implemented to address the core coolability, control rod insertability, and fuel structural integrity issues which arise because of the transient loads which may occur during LOCA and seismic events. This methodology is applicable to design basis LOCA and seismic events using the approved analysis codes and assumptions, and assumes that all original baffle-barrel bolting components are and will remain in their as-designed conditions.

In Section 1.0, background on foreign plant ultrasonic inspections of baffle-former bolts was identified. While no ultrasonic indications of baffle-former bolt cracking have yet been found in U.S. domestic plants, the possibility of such cracking increases as plants age. This, in turn, raises the question whether previous analyses, based on completely intact baffle-former-barrel bolting distributions, represent the most limiting conditions if some bolts are assumed to be cracked. It is to address this potentially more limiting condition that the faulted load analysis procedure outlined below and in Figure 2-1 is proposed. In most respects, this procedure is the same as the procedure given in Reference 1-1, with some modifications. In the outline below and in Figure 2-1, these modifications are identified.

<u>LOCA Portion of Figure 2-1</u>

a. Break Type, Break-Opening Time (BOT), Baffle-Former-Barrel (BFB) Configuration Determination (Section 2.1.1):

This block of Figure 2-1 constitutes the starting point or points of the LOCA load analyses sequence. The break type and location are selected on the basis of leak-before-break (LBB) applicability; LOCA load analyses need be performed only for the most limiting break or breaks. The break opening time is determined from the design basis of Reference 1-3. Depending on whether multiple baffle-former-barrel configurations are to be covered by the analysis, iterations may be necessary. If only one BFB design configuration is to be considered, iterations are not necessary. If more than one configuration is to be covered, the iteration loop shown in Figure 2-1 is repeated until a determination of the most conservative BFB configuration is made based on calculated core barrel/core plate motions and baffle plate pressure loads. This configuration is then used as the basis for subsequent calculations.

LOCA Forcing Function Analysis (Section 2.1.2):

To determine the loads acting on the components of interest - in this case the baffle-barrel-bolting and fuel - appropriate thermal-hydraulic analyses of the primary system between the break and these components are necessary. For a LOCA event, there are two phenomena which can yield significant loads on the baffle-former-barrel region. These phenomena occur at different points in the LOCA transient and are characterized by significantly different time

scales; they are referred to as a) the sub-cooled and b) the two-phase parts of the LOCA transient, respectively. These are discussed below:

b.a Sub-Cooled LOCA Load Analysis (Section 2.1.2.1):

This portion of the transient encompasses the period during which acoustic waves, generated by the abrupt break-opening, travel through the primary system and create single phase pressure differentials loads on the primary system components. MULTIFLEX 3.0 is used to calculate the acoustic pressure-time histories from which these loads are determined. For the baffle-former-barrel region and fuel, there are two important aspects of these acoustic loads: 1) the loads acting on the core barrel, and 2) the loads acting on the baffle plates and the former plates. Both of these loads can contribute to motions which may lead to impacting between the fuel assembly grids and baffle plates, and between the grids of adjacent fuel assemblies. Typically, this portion of the LOCA transient persists for no more than 500 milliseconds.

b.b Two-Phase LOCA Load Analyses (Section 2.1.2.2):

Following the acoustic or single phase stage of the LOCA transient, the primary system continues to depressurize due to the discharge of fluid through the break and saturation pressures are ultimately reached. In the baffle-former-barrel region, the low flow areas of the formers coupled with two-phase conditions can create a situation in which fluid cannot easily escape this region. This, in turn, can lead to baffle-former-barrel region fluid pressures in excess of core pressures, thus creating loads on the baffle plates in the direction of the core. For large breaks, SATAN is used to calculate these loads. For the smaller breaks which must be considered after the application of LBB, COBRA/TRAC is used for these load calculations (see Section 2.1.2.2 for details).

Typically, this portion of the transient begins no earlier then 10-15 seconds after break-opening, and can persist for 100 seconds or more, depending on break size. Smaller breaks tend to persist longer but generate smaller loads than larger breaks. Because of the long time period and low rates of change of this portion of the LOCA transient, break-opening-time is not a significant factor in the determination of two-phase loads.

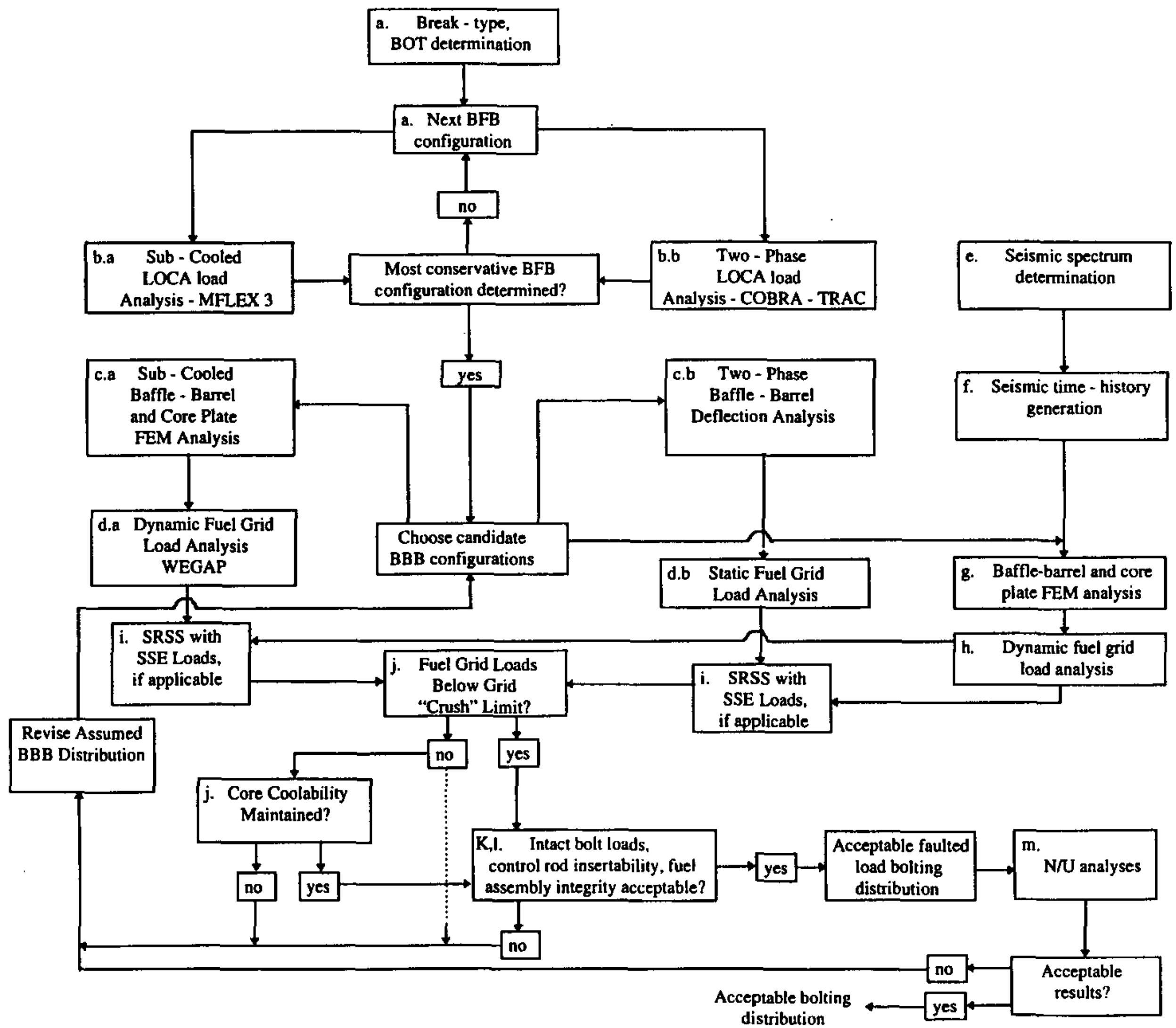


Figure 2-1 Seismic/LOCA Analysis Approach for Determining Acceptable Baffle - Barrel-Bolting

c. Baffle-Former-Barrel Region and Core Plate Finite Element Analyses (Sections 2.1.3 and 2.1.4)

The sub-cooled and two-phase portions of the LOCA transient both produce pressure loads on the baffle plates. However, the sub-cooled part of the transient has an additional loading component: core barrel/core plate motion. In the sub-cooled portion of the transient, asymmetric loads on the core barrel are produced when the LOCA-induced pressure wave preferentially loads the downloop side of the vessel downcomer, thus inducing core barrel and core plate motion. The time scale of the two-phase portion of the transient is sufficiently large (10-100 seconds) that asymmetric loads on the core barrel are small; thus, no significant core barrel motion is induced here. The finite element method (FEM) analyses associated with the sub-cooled and two-phase regions are outlined below:

c.a Sub-Cooled Baffle Displacement and Bolt Load Analysis (Section 2.1.3)

As noted above, there are two aspects of this analysis: a) core plate motion, and b) direct pressure loads on the baffle plates.

- Core Barrel/Core Plate Displacement - Time (Section 2.1.3.1). In this analysis, the asymmetric pressure loads in the vessel downcomer calculated by MULTIFLEX 3.0 are used as input to a finite element analysis of the core barrel/core plate system. The finite element model used to perform this calculation includes the core internals, reactor vessel and fuel.

- Baffle - Former Pressure Loads (Section 2.1.3.2) The baffle-former-barrel region is comprised of a series of fluid volumes separated by former plates containing holes with relatively small flow areas. The fluid dynamics of this region are therefore significantly different than the core region, resulting in significant lags in response time. This, in turn, can lead to significant transient pressure loads on the baffle plates, former plates, and bolting.

c.b Two-Phase Baffle Displacement and Bolt Load Analysis (Section 2.1.4)

As noted above, only the baffle pressure loads, not core plate motions, are significant during this part of the transient. Because the time scale is large and the rates of change correspondingly small here, these pressure loads can be considered quasi-static in nature.

d. Fuel Grid Load Analysis (Section 4.1.1)

The baffle motions and, for the sub-cooled portion of the transient, the core plate motions result in relative displacements between the baffle plates and fuel assemblies. For the sub-cooled portion of the LOCA transient, this may result in fuel grid impact loads (Section 4.1.1.1, or d.a on Figure 2-1). For the two-phase portion of the LOCA transient, the fuel grid loads may be considered quasi-static in nature (d.b in Figure 2-1).

<u>Seismic Portion of Figure 2-1</u>

e. Seismic Spectrum Determination (Section 2.2.1)

Generally, each of the several plants in a given group to which the analysis results will apply will have unique seismic spectra. For the analysis results to apply to all plants in the group, a conservative seismic spectrum must be chosen. There are several options which may be chosen: 1) select the most conservative spectra in the group, 2) define "composite" spectra for the entire group. Other conservative approaches may also be employed.

f. Seismic Time-History Generation (Section 2.2.2)

For subsequent time-history analyses, the chosen spectrum must be converted to an equivalent acceleration time-history. This is done by time-history synthesis techniques, for use as input to the time-history analysis described below.

g. Baffle-Former-Barrel and Core Barrel/Core Plate Finite Element Time-History Analyses (Section 2.2.3)

These analyses are similar to the analysis of baffle and core barrel/core plate motions described above in the LOCA Branch and in Sections 2.1.3.1 and 2.1.3.2, with the exception that seismic acceleration-time histories are the driving force rather than asymmetric pressure loads on the core barrel. Unlike the sub-cooled portion of the LOCA event, core barrel motion is the only cause of baffle plate and fuel assembly motion in the seismic event, and of the bolt loads and fuel grid impacting loads these motions engender.

h. Fuel Grid Load Analysis (Section 4.1.1)

This is exactly the same type of analysis performed for the sub-cooled potion of the LOCA transient, the only difference being that the input baffle and core plate motions are derived from the seismic, rather than the LOCA, event.

<u>Load Acceptability Portion of Figure 2-1</u>

Following the LOCA and seismic analyses, evaluation of the calculated loads must be performed to determine acceptability. The several aspects of these evaluations are discussed below.

i. SRSS of Seismic and LOCA Loads

For some plant design bases, LOCA and seismic loads must be combined in a square-root-of-sum-of-squares (SRSS) method. For other plant design bases, separate consideration of LOCA and seismic loads is permitted. The plants comprising the analyses group must decide which of these approaches is appropriate.

j. Core Coolability Evaluation (Section 4.2)

The primary criterion to be considered in the determination of acceptable baffle-barrel-bolting distributions is core coolability. To demonstrate adequate core coolability, one of three options may be selected:

- All fuel grid loads below grid strength allowables (Section 4.2.1)
- Some fuel grid loads in peripheral assemblies greater than grid strength allowables (Section 4.2.2)
- Some fuel grid loads in core interior assemblies greater than grid strength allowables (Section 4.2.3)

If all calculated fuel grid loads are below grid "crush" allowables (Section 4.2.1), no core cooling evaluation is needed. If some calculated fuel grid loads are above grid "crush" allowables, the appropriate option can be selected for evaluation or, if possible, a revised bolting distribution can be assumed. Both of these options, however, necessitate evaluations of core coolability as described in Sections 4.2.2 and 4.2.3.

k. Control Rod Insertability, Fuel Rod Integrity (Sections 4.1.2 and 4.1.3)

In addition to determining the extent of fuel grid "crush," evaluations must be performed to ensure that adequate control rod insertability is maintained, and that no other damage to the fuel occurs. Historically, these criteria have been less limiting than grid "crush." Nevertheless, these evaluations must be performed.

l. Intact Baffle-Barrel Bolt Loads (Section 4.3)

Bolt stresses are calculated as part of the baffle-former-barrel finite element analyses in Sections 2.1.3.3.2 and 2.1.4.3.2. To assure that candidate bolting distributions are viable, these stresses must be compared against stress allowables which consider the bolt geometries, materials, properties and environmental effects on properties.

It should be recognized that bolt stress criteria are applied only to ensure that the bolting distribution being considered is one that will survive LOCA and seismic events. If bolt stress criteria are not satisfied for some bolts, this means only that the bolting distribution being considered is not viable and must be revised; there is no direct effect on safety. The desired goal is to determine a viable bolt loading pattern, including the possible case in which a highly stressed bolt is assumed to be broken. For this case, the analysis must ensure that the stresses in the remaining bolts are acceptable and that "unzippering" does not occur.

m. Normal/Upset (N/U) Acceptability (Section 3.0)

Normal/upset conditions which are potentially affected by reduced bolting are evaluated to address plant operability concerns. The particular normal/upset (N/U) conditions potentially affected by reduced bolting are discussed in Section 3.0.

"Acceptable" bolting distributions are those which meet all the criteria outlined above and discussed in subsequent sections for faulted and normal/upset conditions. Bolting distributions which do not satisfy all relevant criteria are rejected, or revised as shown in Figure 2-1. Revised bolting distributions must, of course, undergo all the stages of analysis described above and illustrated in Figure 2-1.

2.1 LOSS-OF-COOLANT ACCIDENT (LOCA) ANALYSIS METHODOLOGY

10 CFR 50, GDC-4 requires that systems and components which perform functions important to safety be designed to survive the dynamic effects associated with postulated pipe rupture (LOCA), as well as other postulated events described therein. The postulated LOCA loads and any required seismic loads tend to be the most limiting structural design constraint for the reactor vessel internal components and fuel, and are therefore the limiting loads for assessing the acceptability of baffle-barrel bolting configurations.

2.1.1 Break Considerations

Break modeling in the LOCA analysis is sensitive to three criteria: break size, break location, and break opening time. Historically, circumferential pipe breaks have resulted in more limiting loads than longitudinal pipe breaks, and as a result, circumferential breaks are considered to be the limiting breaks for analysis. Circumferential break size is a function of the piping diameter under consideration, along with any restraints on pipe separation. The original break locations postulated for the design of Westinghouse NSSS are described in Reference 2-1. These break locations, without further consideration, would result in break areas equivalent to the piping cross sectional area. However, the consideration of piping whip restraints along with the piping separation loads generated by the LOCA result in substantial reductions in the maximum pipe break areas. These calculations have in some cases been performed on plant specific bases. A sample of generic bounding break opening areas is tabulated in Reference 2-2, on the basis of numerous plant specific calculations. These break areas can be used as bounding for calculations which do not yet credit leak-before-break or other exemptions or exclusions regarding the main coolant loop piping.

2.1.1.1 Leak-Before-Break Application

Leak-Before-Break (LBB) licensing allows for the exclusion of piping for dynamic effects associated with postulated pipe rupture under General Design Criteria 4 (GDC-4). These excluded effects include pipe whip and both internal and external (cavity pressurization) pressure loads resulting from such breaks. The generic technical basis for LBB was approved in Reference 2-3 (U.S. NRC Generic Letter 84-04).

For purposes of calculating acceptable baffle-barrel-bolting configurations, LBB exclusion of at least the main loop piping (or other NRC approved GDC-4 exemptions) will be considered for a particular analysis. In order for a particular calculated acceptable bolting configuration to be applicable to a particular plant, that plant must demonstrate that the largest piping not excluded from GDC-4 by LBB or otherwise exempted from GDC-4 is no greater than that of the piping breaks analyzed. A plant may always choose to use as bounding a bolting configuration calculated for a pipe break size greater than the largest size excluded under LBB. This allows for plants with differing levels of LBB licensing exclusions to be grouped together under the same bolting configuration calculation, provided that other similarities exist which make such groupings possible.

2.1.1.2 Break-Opening Time

Reference 1-3 provides justification for a proposed design basis which employs break-opening times larger than 1 millisecond as allowed by Reference 2-4. This justification is based on the results of break-opening experiments, calculations of break-opening by Westinghouse and others, engineering practices of domestic and foreign nuclear suppliers, conservatism inherent in MULTIFLEX, and regulatory considerations.

The break-opening time design basis proposed in Reference 1-3 includes proposals for breaks resulting from postulated longitudinal and circumferential breaks. In Westinghouse plants, however, longitudinal breaks have been found to be non-limiting; therefore, only that part of the proposed design basis dealing with circumferential breaks is relevant to the determination of acceptable bolting distributions.

2.1.1.3 Break Location

While LBB determines which lines are excluded from consideration under GDC-4, Reference 2-1 provides the appropriate guidance on where to locate the pipe breaks in the Westinghouse primary coolant loop. The break locations considered in the analysis of each bolting configuration will be the two largest lines not excluded or exempted from consideration under GDC-4, one of which will be on the RCS cold leg, the other on the RCS hot leg. Typically, where the main loop piping is excluded, these breaks will be in the three main branch lines, the accumulator line on the cold leg and the pressurizer surge line or RHR line on the hot leg. As indicated in Reference 2-1, the appropriate location for the branch line break is at the safe end of the branch line nozzle.

2.1.2 LOCA Pressure Loads

The LOCA event results in blowdown loads on the reactor vessel and internals which can be characterized as three distinct phenomena, which are significant to varying degrees on different vessel components. The first type of load introduced by a LOCA is a sub-cooled acoustic de-pressurization wave which traverses the reactor coolant system from the break location at the speed of sound. Pressure differentials caused by these acoustic pulses cause loads across various vessel internal components, and they tend to peak very early in the transient, typically peaking within the first 50 milliseconds for breaks the size of large branch lines. The second

type of load introduced by a LOCA is also primarily a sub-cooled phenomena, that is, the peak loads are seen at sub-cooled fluid conditions, although it continues to act under two-phase fluid conditions. This second type of load is a frictional mass-flow induced load. Unlike the acoustic load, the frictional mass-flow loads tend to peak later in the transient, typically 100 to 200 milliseconds for breaks of one square foot area and larger. These frictional mass-flow induced loads only tend to be limiting for vessel internal components subjected to high cross-flow loads. As such, they are typically only of concern for components such as control rod guide tubes and instrument support columns located in the upper plenum where strong crossflow loads are generated between the intact and broken loop outlet nozzles. The third type of load is a two-phase load which occurs much later in the transient, typically 5 to 50 seconds for breaks the size of large branch lines and larger. The two phase loads are only significant in highly enclosed sub-regions like the barrel-baffle-former. The small areas available for fluid to escape such sub-regions results in high pressures within the sub-region relative to the pressure in the rest of the system. This puts a load onto the structures enclosing the sub-region, e.g., the baffle plates.

2.1.2.1 Sub-Cooled - MULTIFLEX 3.0

To address the acoustic de-pressurization wave and frictional mass-flow induced loads, an appropriate methodology has been developed and previously approved in the MULTIFLEX 1.0 code (Reference 1-2). This code has been further enhanced to increase the accuracy of the calculation in relation to the modeling of fluid-structure interaction in the vessel downcomer. This enhanced version of the code is designated MULTIFLEX 3.0 (Reference 1-4). The methodology used in modeling the reactor coolant system is described in detail in References 1-2 and 1-4. Reference 1-4 contains example noding diagrams and sample input and output for a three loop plant. For the baffle former bolting analyses, the specific modeling approach to be used in establishing appropriate bounding LOCA forces for each plant group is described as follows.

For each plant grouping, a representative plant is selected for purposes of establishing the physical geometry and hydraulic loss coefficients for the model. The structural data provided for modeling the flexibility of the core barrel and vessel are selected to represent the limiting configuration from that group of plants being analyzed. The break sizes and locations are postulated to be the most limiting breaks applicable to each group of plants. The dominant contributor to the magnitude of the LOCA forces generated (beyond break size, plant size and reactor internals configuration) is the fluid density. Higher fluid density contributes to more severe loads. Therefore, the operating conditions are selected to bound the lowest temperature cold leg, and the highest initial system pressure. As an additional conservatism, the operating conditions selected are also at the lowest allowable thermal design flow temperature minus uncertainty, and the highest allowable pressurizer pressure plus uncertainty. This choice of parameters will result in conservatively calculated LOCA hydraulic forces for each plant type to be modeled.

The MULTIFLEX 3.0 generated hydraulic transient data is converted to horizontal and vertical loads on the reactor vessel and internals using the LATFORC and FORCE2 codes as discussed in Reference 1-2. The LATFORC and FORCE2 inputs are based upon the selected representative

plant for each group of plants being analyzed. The use of these two codes is the same as in the existing analysis methodology (Reference 1-1). In addition to the standard horizontal and vertical forces generated by LATFORC and FORCE2, the baffle-former-barrel region pressures and core pressures from MULTIFLEX 3.0 are provided independently for application to the finite element structural analysis of this region.

2.1.2.2 Two-Phase - WCOBRA/TRAC

The concept of a fast blowdown associated with the generation of two-phase loads on enclosed sub-regions was originally conceived in association with large double-ended guillotine (DEG) breaks of the primary reactor coolant loop piping. The core de-pressurization transients which resulted in these loads were originally calculated using the approved SATAN (Reference 1-5) large break LOCA model. The pressure at the top and bottom of the core nodes in the SATAN model were used as boundary conditions for a more detailed baffle-former-barrel region analysis. The results showed loads that were comparable to, but generally bounded by, the single-phase acoustic loads.

Two-phase load calculations have only rarely been performed since the 1970's. The primary exception to this has been in upflow conversion programs. The upflow configuration design changes have been verified as acceptable based upon SATAN double-ended guillotine (DEG) break blowdown depressurization transients. The vast majority of these analyses have pre-dated leak-before-break (LBB) licensing which excludes dynamic effects associated with postulated main coolant loop piping rupture from the general design criteria design basis. It is therefore necessary to consider the two-phase blowdown loads resulting from postulated ruptures of the branch line piping, for the baffle-former-barrel region.

Based upon physical arguments alone, the two-phase loads would be expected to rapidly diminish for reduced break sizes associated with branch line breaks. The factors which make the two-phase loads large for DEG breaks of the main coolant loop piping are the rapidity of the de-pressurization through the large break, compared to the slower de-pressurization of the baffle-former region through the holes in the formers. As the postulated pipe ruptures are brought down to branch line area sizes, the time of de-pressurization increases substantially, allowing more time for flow through the former holes to drop the pressures in the baffle-former sub-region closer to equilibrium with the rest of the vessel. As a result, the two-phase loads for branch line breaks postulated as limiting under LBB licensing are expected to be substantially non-limiting in comparison to the single phase acoustic loads.

To demonstrate this analytically, Westinghouse will perform a single depressurization transient calculation for each plant type to be bounded. For the purposes of the baffle-barrel bolting program, the two-phase loads are expected to be non-limiting. A single depressurization calculation will be performed for each analyzed plant grouping: the two-loop, three-loop, and four-loop plants will each have one representative core depressurization calculation analyzed for two-phase loads. Providing that this analysis shows two-phase loads are substantially non-limiting compared to the single phase loads, the issue of two-phase loads will be considered to be adequately addressed. In the event that the single phase and two-phase loads are of

comparable magnitude, a more detailed calculation may be required to address the two-phase LOCA load issue.

The break sizes currently under consideration for the baffle barrel bolting program range between 10" accumulator line piping and 14" pressurizer surge line piping, a range of 60 in^2 to 100 in^2 in pipe break area. The WCOBRA/TRAC code has been introduced and licensed for use in both Appendix K large break LOCA calculations for the upper plenum injection plants (primarily two loop) using the SECY-83-472 methodology, and for best-estimate large break LOCA calculations for three and four loop plants (Reference 1-7). A two loop upper plenum injection best-estimate WCOBRA/TRAC model is currently under review (Reference 1-8). The WCOBRA/TRAC code has been used for split breaks as small as 0.1*ACL (<60 in^2), and as large as a DEG break with a Moody discharge coefficient of 1.0. Therefore, WCOBRA/TRAC was selected as the code to analyze the two-phase portion of the LOCA pressure transient.

The two-phase LOCA loads can be used as the design basis for acceptable baffle-barrel-bolting distributions, although they are not expected to be limiting relative to the sub-cooled acoustic loads. Therefore, it may be necessary only to confirm the non-limiting nature of the two-phase LOCA load transient by comparison to the equivalent break size single phase acoustic LOCA load transient. For this reason, a single representative calculation for each basic plant type (two loop, three loop, and four loop) is expected to be adequate and appropriate.

These two-phase loads will be addressed by using a best-estimate LOCA (WCOBRA/TRAC) model for a representative plant in the appropriate plant group under consideration. This methodology is described in detail in References 1-7 and 1-8. For three and four loop plants, the modeling is discussed in section 20 of Reference 1-7. For two loop plants, the appropriate modeling is discussed in section 3 of Reference 1-8. Breaks of these sizes would be modeled appropriately as split breaks, which are discussed in detail in Section 22-6-2 of Reference 1-7.

2.1.3 Sub-Cooled Baffle-Former-Barrel Region and Core Plate Finite Element Analyses

The sub-cooled and two-phase portions of the LOCA transient both produce pressure loads on the baffle plates. However, the sub-cooled part of the transient has an additional loading component: core barrel/core plate motion resulting from the asymmetric loads on the reactor internals. In the sub-cooled portion of the transient, asymmetric loads on the core barrel are produced when the LOCA-induced pressure wave preferentially loads the downloop side of the vessel downcomer, thus inducing core barrel and core plate motion. The time scale of the two-phase portion of the transient is sufficiently large (10-100 seconds) that asymmetric pressure loads on the core barrel are not significant; thus, no significant core barrel motion is induced here. The finite element (FEM) analyses associated with the sub-cooled region of the transient is outlined in the following sections.

2.1.3.1 Core Barrel/Core Plate Displacement-Time Analysis

This analysis models the dynamics of the reactor internals, core and reactor vessel to calculate core plate displacement-time histories resulting from LOCA-induced asymmetric loads. These loads are calculated by the MULTIFLEX 3.0 program (Reference 1-4).

Historically, the WECAN program (Reference 2-5) has been used to perform this analysis. Other finite element structural analysis programs which have been suitably qualified and verified, such as the ANSYS program (Reference 2-6), may also be used for this purpose. The type of modeling needed to perform this kind of analysis is described in Section 3.0 of Reference 1-1. The analytical approach itself is discussed in Section 3.3 of Reference 1-1.

2.1.3.1.1 Finite Element Modeling of the Vessel and Internals

Figures 3-7 (a, b, c, d) of Reference 1-1 illustrate the finite element representation of the vessel, core internals, and fuel which comprise this model. Sections 3.2.2, 3.2.3, and 3.2.5 of Reference 1-1 discuss this model. For determining acceptable bolting distributions, this methodology is the same as described in Reference 1-1.

2.1.3.1.2 Forcing Function Input

For the reactor vessel/internals core displacement calculation, horizontal and vertical forces calculated using the LATFORC and FORCE2 codes (Reference 1-2) are provided. These include the horizontal (X-Y) forces on the vessel, core barrel, and thermal shield (if applicable) at ten different vertical elevations. In addition, the vertical forces provided are described in Table 2-1. Note that not all models include every force, especially since many plant groups will not include a diffuser plate.

2.1.3.1.3 Fluid-Structure Interaction Modeling

The fluid-structure interaction model used in the calculation of acoustic and mass-flow blowdown loads includes a flexible core barrel. The most significant loads in terms of generating the core plate displacements are the loads on the core barrel. As a result, the core barrel is modeled as a flexible structure to allow fluid-structure interaction to reduce these loads. Reference 1-2 discusses in detail the fluid-structure interaction model employed in MULTIFLEX 1.0. Reference 1-4 discusses the enhancements to the core barrel model (network downcomer, non-linear boundary conditions, vessel motion) employed in a MULTIFLEX 3.0 analysis. Note that only the core barrel and downcomer see a flexible wall model. The baffle-former-barrel region continues to be treated with rigid boundary conditions.

2.1.3.1.4 Typical Results

Figure 3-19 of Reference 1-1 illustrates typical core plate motions for the plant and fuel considered in that document. Although the details of these motions will be plant and break dependent, the fundamental character of the results will not be significantly different for the determination of acceptable bolting distributions.

Table 2-1 Typical Example of Standard FORCE2 Force Descriptions		
Force #	**Force Description**	**Element #**
1	Friction Force on inside/outside of the Thermal Shield (TS) or Neutron Pads (NP)	3, 4
2	Friction (Force) on top of the TS or NP	1
3	Pressure (Force) on bottom of TS or NP	2
4	Friction on inside/outside of Core Barrel	8, 9
5	Pressure on top of Core Barrel & Flange	5, 6
6	Pressure on bottom of Core Barrel	7
7	Pressure on bottom of Upper Barrel Flange	10
8	Pressure on Lower Support Plate (LSP)	12
9	Pressure on Lower Core Plate (LCP)	13
10	Pressure on Bottom Fuel Nozzle	15
11	Forces on Grids and Thimbles	16-26, 28
12	Pressure on Top Fuel Nozzle	27
13	Pressure on Fuel Rod per Rod	29
14	Pressure on Upper Core Plate (UCP)	30
15	Pressure on Upper Support Plate (USP)	14
16	Total Force on Reactor Vessel	11
17	Pressure on Formers	31
18	Diffuser Plate Force	32

2.1.3.2 Baffle-Former Displacement-Time Analysis

The second component of baffle motion is the displacement of the baffle plates relative to the formers and core barrel. The movement of the core plates and core barrel (Section 2.1.3.1) affect this motion and are incorporated as forcing functions into the baffle-former displacement-time analysis. The more important driving forces for this analysis, however, are the pressure differences which act directly across the baffle plates. These occur because of the time lag between the arrival of the LOCA rarefaction wave in the reactor vessel and downcomer (Section 2.1.2.1) and the depressurization of the baffle-former-barrel region. The latter is relatively slow-acting because of the restricted hydraulics (low flow areas) of the former plates. The analysis approach used in this region is described below.

The analysis described in this section is needed only for the determination of acceptable baffle-former-barrel bolting. In the nominal condition, with all bolts intact, baffle deflections relative to the barrel are insignificant.

2.1.3.2.1 Finite Element Modeling of the Baffle-Former-Barrel Region

A single octant model of the baffle-former-barrel region is used in this analysis. Figure 2.1.3.2.1-1 shows a typical model for a two loop plant. The baffles and formers are represented as elastic plate elements. The core barrel is modeled only as an external boundary. Baffle-former, barrel-former, and edge bolts are represented as pipe elements (beam elements can also be used). Baffle-former and barrel-former bolts attach the baffle and barrel to the former plates, as shown in Figure 2.1.3.2.1-2. Edge bolts are non-structural and are installed to maintain a sufficiently small gap between adjacent baffle plates to preclude baffle-jetting. In the finite element model, sufficient nodalization is provided to permit representation of all baffle-former, barrel-former, and edge bolts in the simulated octant. Candidate "acceptable" bolting distributions are analyzed by putting the appropriate bolt elements in only at predefined locations.

Gap elements with springs are used to model several possible eventualities. One is baffle plate impacting with the top and bottom nozzles of the fuel assemblies. This may occur for reduced baffle-former bolting distributions for which some baffle plates are undersupported at the top and bottom former levels.

Gap elements of this type are also used at the interfaces between the baffle plates and formers and between the core barrel and formers. These gap elements are used whether or not baffle-former and barrel-former bolts are assumed to be present at given locations. The reason for doing this is because the possibility exists that, whether or not a bolt is present at a given joint, the joint may open during the LOCA event. Figure 2.1.3.2.1-3 illustrates a bolted joint model which permits proper simulation of linear and rotary joint stiffnesses, bolt preload, and joint separation, should it occur.

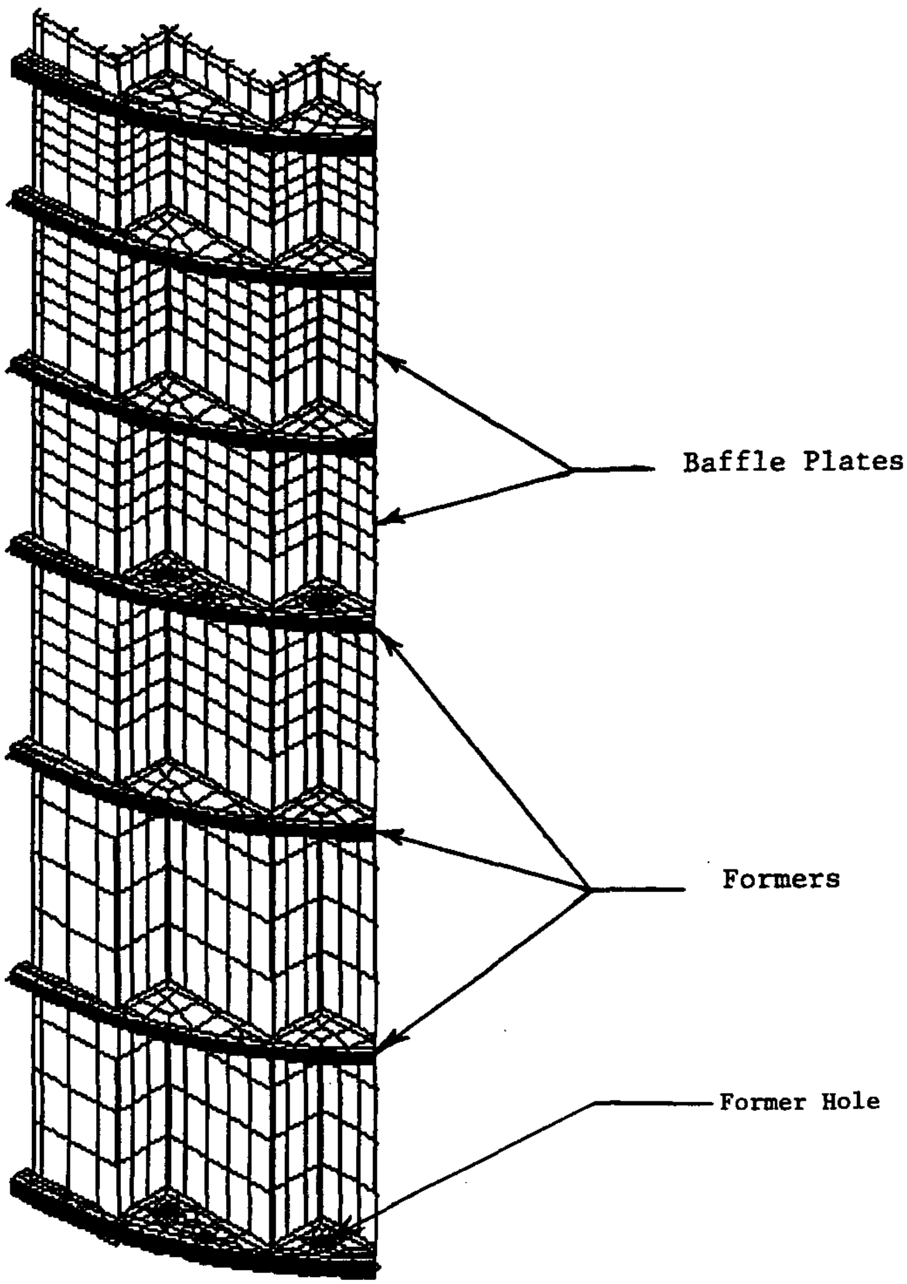


Figure 2.1.3.2.1-1 Typical Baffle-Former Model

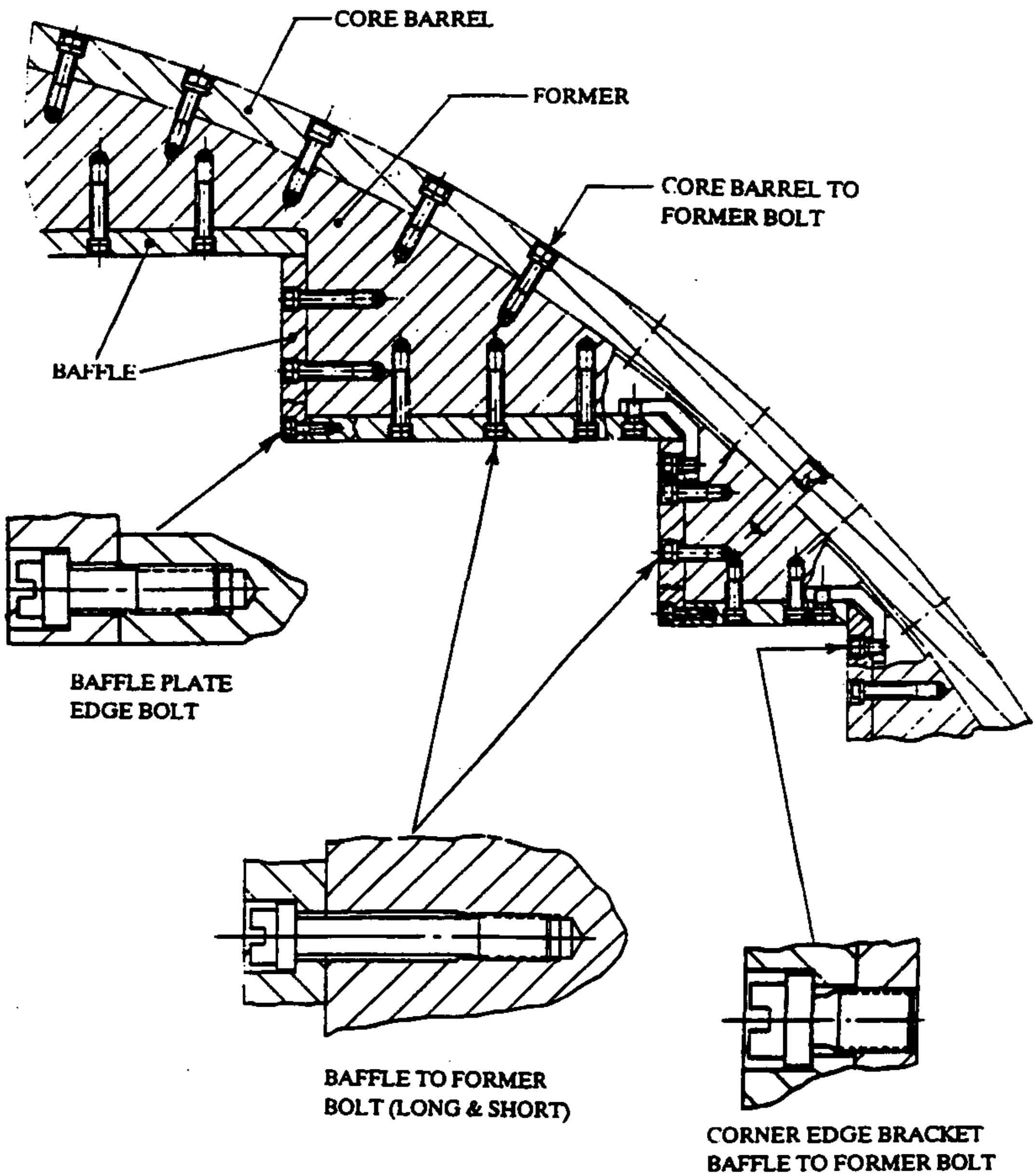


Figure 2.1.3.2.1-2 Baffle-Former-Barrel Bolting

(a,c,e)

Figure 2.1.3.2.1-3 Bolted Joint Simulation

2.1.3.2.2 Forcing Function Input

The baffle-former-barrel region is subjected to two excitation mechanisms: 1) the motion of the core barrel and core plates, and 2) the acoustic pressures caused by the break. The core barrel and core plate motions are calculated as described in Section 2.1.3.1 above. The acoustic pressure excitation is taken directly from MULTIFLEX 3.0 output, as discussed in Section 2.1.2.1 above. Note that these calculations assume rigid baffle and former plates. The flexible modeling in MULTIFLEX 3.0 (Reference 1-4) applies only to the downcomer and core barrel regions of the model. The manner in which these acoustic pressures are applied and fluid-structure interaction modeling are discussed in the next section.

2.1.3.2.3 Fluid-Structure Interaction Modeling

Fluid-structure interaction between the baffles and formers and the fluid between the baffles and core barrel is an important part of the acoustic pressure attenuation process in the baffle-former-barrel region. In essence, fluid-structure interaction permits the fluid to reduce the LOCA loads that would otherwise have to be absorbed by the bolts.

The fluid-structure interaction method that is used for this purpose was developed as part of the upflow conversion program (Reference 2-7). It has also been reported in the open literature and compared to MULTIFLEX 1.0 in Reference 2-8. For baffle-former-barrel modeling, MULTIFLEX 1.0 and MULTIFLEX 3.0 are the same, so that the comparisons in Reference 2-9 serve as a verification of the fluid-structure interaction methodology against either MULTIFLEX 1.0 or MULTIFLEX 3.0.

The baffle-former-barrel region may be considered to be a series of open volumes separated by thin plates (formers) with low flow areas. A typical model that was developed with the WECAN computer program (Reference 2-5) for the Upflow Conversion Program (Reference 2-7) is shown in Figure 2.1.3.2.3-1. The meaning of the various elements of this figure are described below:

1. White rectangles: These are the fluid volumes simulated by brick elements such that the normal stresses (pressures) are the same on all faces, and shear stresses are negligible. The elements behave as a compressible fluid with the appropriate bulk modulus.

2. Fluid mass nodes 51 - 67: These represent the fluid in the former holes and in the top and bottom baffle/core plate gaps.

3. Dampers attached to fluid mass nodes 51 - 67: These simulate the hydraulic drag exerted on the fluid in the former holes and core plate gaps.

Baffle nodes 101 - 115: In the Reference 2-8 effort, the baffles were simulated as spring-mass systems, with the baffle masses attached to nodes 101 - 115.

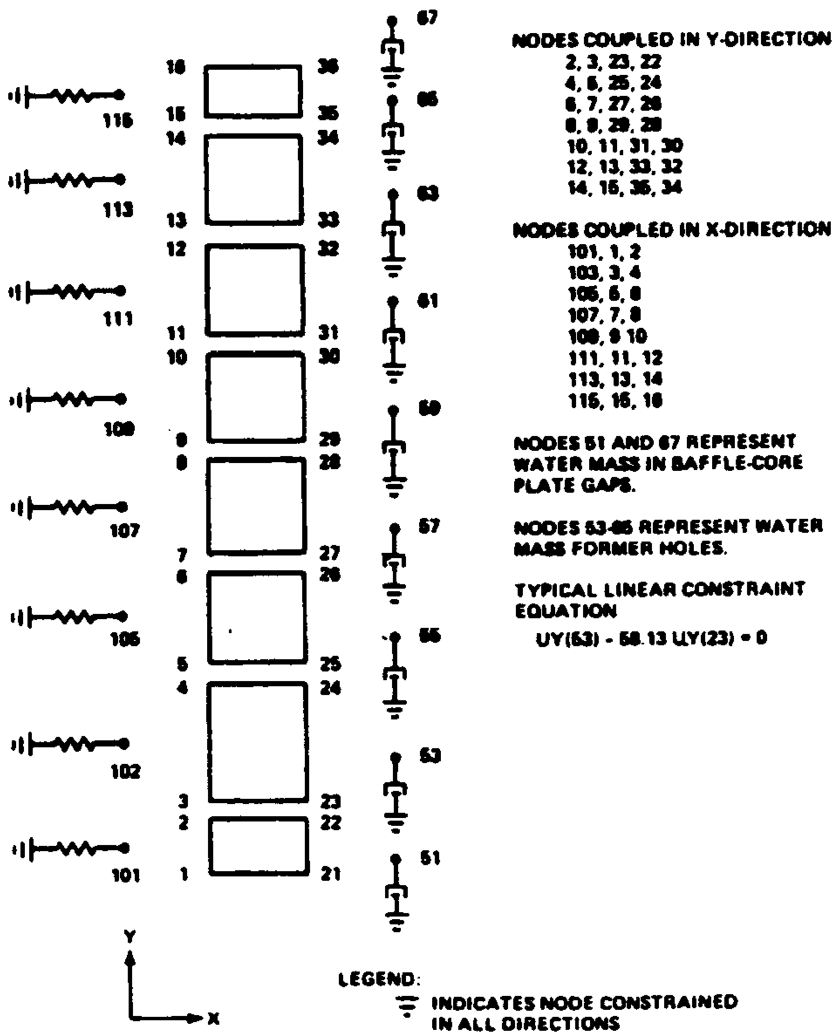


Figure 2.1.3.2.3-1 Baffle-Former-Barrel Fluid-Structure Interaction Modeling

a,b,c

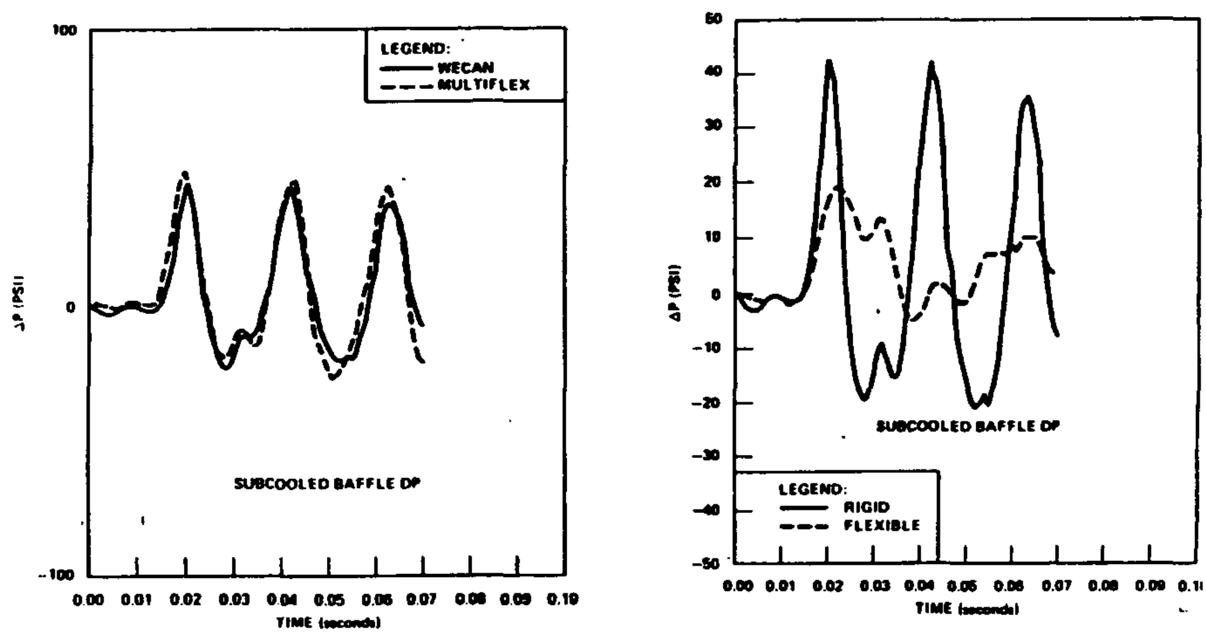
2.1.3.2.4 Typical Results

Figures 2.1.3.2.4-1 to 2.1.3.2.4-3 illustrate baffle plate and former plate pressure load-time histories calculated by the techniques described in this section. Figure 2.1.3.2.4-1 is from the Reference 2-8 study. In the left hand part of the figure (a), a comparison between baffle pressure loads calculated by the present technique with WECAN and baffle pressure loads calculated by MULTIFLEX 1.0 shows excellent agreement. In both these calculations, the baffle plates were assumed to be rigid; the comparison therefore demonstrates the correctness of the fluid acoustics as described in Section 2.1.3.2.3. Figures 2.1.3.2.4-2 and 2.1.3.2.4-3 similarly compare baffle and former pressure load-time histories as calculated by ANSYS using the techniques discussed in Section 2.1.3.2.3, and by MULTIFLEX 3.0. This comparison also assumes a rigid structure and therefore also demonstrates the correctness of the fluid acoustic modeling.

a,b,c

a,b,c

Figure 2.1.3.2.3-2 Fluid-Structure Interaction Volume Displacement Compatibility



(a) Comparison of MULTIFLEX and WECAN

(b) WECAN Results with and without Fluid-Structure Interaction

Figure 2.1.3.2.4-1 Comparison of Finite Element (WECAN) Calculations of Subcooled Baffle Pressure Difference vs Time

a,b,c

Figure 2.1.3.2.4-2 Comparison of (Rigid Structure) Baffle Pressure Differences Calculated by MULTIFLEX 3.0 and by ANSYS

a,b,c

Figure 2.1.3.2.4-3 Comparison of (Rigid Structure) Former Pressure Differences Calculated by MULTIFLEX 3.0 and by ANSYS

Figure 2.1.3.2.4-1(b) shows the pressure load reduction that can be achieved by incorporating fluid-structure interaction into the analysis models. Further verification of the fluid-structure interaction methodology against test data is provided in Appendix A.

2.1.3.3 Sub-Cooled LOCA Finite Element Model Output Parameters

The data generated by the baffle-former finite element analysis with sub-cooled LOCA input loads are of two types: 1) baffle-displacement time histories, and 2) baffle-barrel bolting loads. These are discussed below.

2.1.3.3.1 Baffle Displacement - Time History

For the calculation of fuel grid impact loads in Section 4.1.1, the displacement-time histories of the baffle-plates at the fuel grid elevations are needed. These are calculated by adding the calculated baffle displacements relative to the barrel (Section 2.1.3.2) to the calculated core barrel displacement-time histories (Section 2.1.3.1). The first of these (Section 2.1.3.2) is necessary only for the determination of acceptable bolting distributions, and is not therefore included in Reference 1-1. The second, core barrel displacements, is part of the methodology described in Section 3.3 of Reference 1-1 and should not therefore be considered new technology.

2.1.3.3.2 Baffle-Former-Barrel Bolt Loads

The bolt loads (stresses) that are calculated must be compared against allowables to assure that the assumed bolting distributions are valid candidates. For some plants or plant groups, it may be necessary to combine LOCA and seismic bolt loads (SRSS); for others, it may not be necessary. The stress allowables against which calculated stresses are to be compared are discussed in Section 4.3.

2.1.4 Two-Phase Baffle Displacement Analysis

As noted in Section 2.1.3, the time scale of the two-phase portion of the LOCA transient is sufficiently large (10 - 100 seconds) that asymmetric loads on the core barrel are not significant. The finite element analyses associated with the two-phase region of the transient, should it be needed, is outlined in the following sections.

2.1.4.1 Finite Element Modeling of the Baffle-Former-Region

This model is the same as described in Section 2.1.3.2.1 and Figure 2.1.3.2-1.

2.1.4.2 Forcing Function Input

The results of the WCOBRA/TRAC (Reference 1-7, 1-8) blowdown analysis consist of the pressures in the reactor vessel/internals. These pressures are used to calculate the differential pressures across the baffle and former plates. Note that no fluid-structure interaction is included in the WCOBRA/TRAC blowdown calculation.

2.1.4.3 Two-Phase LOCA Output Parameters

The data generated by the baffle-former finite element analysis with two-phase LOCA input loads are of two types: 1) maximum baffle displacements, and 2) baffle-barrel bolting loads. These are discussed below.

2.1.4.3.1 Baffle Displacements

For the calculation of fuel grid loads in Section 4.1.1, the maximum displacements at the fuel grid elevations are needed. Only maximum displacements are needed because the baffle-former-barrel structure responds quasi-statically to the slow, two-phase portion of the LOCA transient. This large time constant also renders negligible the asymmetric core barrel loads that can be significant during the sub-cooled part of the transient (Section 2.1.3.3.1). Calculations of maximum displacements resulting from two-phase pressure differences across the baffles and formers may also be performed as part of the determination of acceptable bolting.

2.1.4.3.2 Baffle-Former-Barrel Bolt Loads

The maximum bolt loads (stresses) calculated in the two-phase portion of the LOCA transient can be evaluated in the same way as are the bolt loads (stresses) generated in the sub-cooled portion of the transient (Section 2.1.3.3.2).

2.2 SAFE SHUTDOWN EARTHQUAKE (SSE) ANALYSIS

As with the LOCA event, the analysis of the seismic (SSE) transient involves several stages of analyses. The steps are similar to those of the LOCA analysis, because both are transient analyses and both involve movement of the core internals, vessel, and fuel relative to each other. A basic difference between the transient analyses of the LOCA and seismic events is that, in the latter, no external acoustic pressure waves of any significance generate pressure loads on the baffle plates; only "system" loads — those associated with core barrel motion — are important.

The various steps in a seismic (SSE) analysis for determining acceptable bolting distributions are outlined below.

2.2.1 Seismic Spectrum Determination

The plant or plant groups to be considered in the acceptable bolting analysis will determine the seismic spectrum or spectra to be used. This may be a limiting spectrum or a composite spectrum for the entire group. Determining the appropriate spectrum to be analyzed is not dependent on the baffle-former-barrel bolting, and is therefore covered by Reference 1-1, Section 3.2.

2.2.2 Seismic Time-History Determination

Once a spectrum has been defined it must be converted to a time-history for analysis. This synthesis is a part of the seismic analysis procedure outlined in Section 3.2 of Reference 1-1 and is the same for full or reduced baffle-former-barrel bolting.

2.2.3 Baffle Displacement-Time Analysis

The total baffle displacement at a given time is the sum of the core barrel displacement plus the displacement of the baffle plate relative to the core barrel. These are discussed below.

2.2.3.1 Core Barrel/Core Plate Displacement - Time Analysis

This analysis models the reactor internals, core, and reactor vessel to calculate core barrel/core plate displacement time histories resulting from the seismic accelerations (Section 2.2.2) applied at the reactor vessel supports. The manner in which this analysis is done is adequately described in Section 3.2 of Reference 1-1.

2.2.3.2 Baffle-Former Displacement - Time Analysis

The core barrel/core plate motions determined from the analysis described above (Section 2.2.3.1) are used as input to the baffle-former displacement - time analysis. The finite element model for this is the same one discussed in Section 2.1.3.2 above and illustrated in Figure 2.1.3.2.1-1. The manner in which the core barrel/core plate motions are applied to this model may be directly (displacement or acceleration time-histories applied to the appropriate barrel-side nodes) or by converting accelerations to equivalent pressure-time histories acting on the baffle plates.

This analysis is performed only in the assessment of acceptable bolting distributions.

2.2.3.3 Output Parameters

The output parameters from the seismic (SSE) analysis are the same as those for the LOCA analysis, as discussed in Section 2.1.3.3. These are comprised of: 1) maximum bolt stresses which, when compared to faulted allowables (Section 4.3), determine the integrity of candidate bolting distributions, and 2) baffle displacement-time histories, which are used in subsequent fuel assembly load evaluations (Section 4.1).

3. NORMAL/UPSET LOAD ANALYSIS

The faulted load analysis approaches described in Section 2.0 are the primary means for assessing the acceptability of revised baffle-barrel-bolting distributions. The methods used to evaluate N/U conditions are the same regardless of the numbers and distributions of baffle-barrel-bolting; therefore, the details of the N/U analysis methods used to assess the acceptability of reduced bolting distributions are not included here. However, it is possible that reduced bolting could lead to undesirable N/U performance. The N/U analyses are performed for this reason, and are described below.

3.1 THERMAL GROWTH AND STEADY-STATE PRESSURE LOADS

Thermal growth of the baffle plates relative to the core barrel and steady-state pressure loads have two potentially adverse effects: 1) Opening of the baffle/baffle gaps, and 2) thermal cycling and fatigue of the baffle-barrel-bolting. The purpose of performing this analysis is to determine the extent to which these effects are influenced by reduced bolting, particularly reduced edge bolting. These phenomena may lead to: 1) increases in the number of cracked bolts, and/or 2) opening the baffle-baffle gaps enough to increase bypass flow beyond design limits.

The thermal growth analysis is performed in two stages: 1) analysis of baffle-barrel temperature differences, and 2) analysis of former temperature distributions. The results of these analyses are then combined into a structural model, from which bolt stresses and baffle gaps are determined. The final steps in the analysis sequence are to take the structural analysis results and use them to perform baffle-jetting and bypass flow calculations as well as a bolt low cycle fatigue evaluations.

3.2 FLOW-INDUCED VIBRATION

Vibration of the baffle plates may be induced by the vibration of the core barrel acting through the formers and by creating pressure fluctuations in the baffle-former-barrel fluid. To analyze the response of the baffle plates for each candidate bolting distribution, the appropriate core barrel modal displacement amplitudes are applied to the formers and the fluid nodes of a finite element model such as the one shown in Figure 2.1.3.2.1-1. This is done for the significant core barrel beam and shell modes. The resulting bolt stress amplitudes are then combined by square-root-of-sum-of-squares (SRSS) and a high cycle fatigue evaluation performed.

4. BOLTING CONFIGURATION ACCEPTANCE CRITERIA

Fuel assembly dynamic and grid load analyses under LOCA and seismic conditions are discussed in Section 3.0 of Reference 1-1. The following summarizes the essential features of that analysis methodology, which has not changed since Reference 1-1 was issued.

4.1 FUEL ASSEMBLY LOAD EVALUATION

Per U.S. NRC Standard Review Plan 4.2 (Reference 4-1), the objectives of the fuel system safety analyses are to provide assurances that: (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained.

The structural adequacy of the fuel assembly design is assessed using U.S. NRC requirements for combined seismic and LOCA loads per Appendix A of the NRC SRP 4.2. Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Force," is the basic document for the fuel assembly structural integrity evaluation, Reference 4-1. This document outlines the detailed review of the analysis methodology, strength requirements, and acceptance criteria. The methodology and analysis procedures, including analytical method, fuel assembly and reactor core modeling, grid strength determination, and computer code WEGAP have been reviewed and approved by the U.S. NRC, Reference 1-1. U.S. NRC approved code, methodology, and testing methods will be used to assess the structural design of the fuel assemblies.

The two criteria for seismic loading design are: (1) fuel rod fragmentation must not occur as a result of the seismic loads and (2) control rod insertability must be maintained. The principal acceptance criteria for a LOCA event are: (1) fuel rod fragmentation must not occur as a direct result of the blowdown loads, (2) CFR Part 50 temperature and oxidation limits must not be exceeded, and (3) control rod insertability must be used in the analysis. Therefore, the core coolable geometry and control rod insertability functional requirements must be satisfied when combined seismic and LOCA loads exceed the fuel assembly grid's strength.

The faulted condition loads considered in the design evaluation are derived for the maximum responses obtained from the lateral safe shutdown earthquake, SSE, and the loss-of-coolant blowdown accident, LOCA. Section 2.0 describes how displacement-time histories for the baffle plates, core barrel and core plates are calculated for faulted conditions. For sufficiently large displacements of these components, impacting between the fuel and the baffles, and between adjacent fuel assemblies, may occur. Since the reactor core exhibits a geometrically non-linear behavior due to core component discontinuities, a time history method is used to obtain the fuel assembly responses. The grid loads are obtained by combining the maximum seismic and LOCA loads using the SRSS methodology, when applicable. The stresses in the fuel assembly components resulting from the seismic and LOCA induced deflections are used to evaluate the structural integrity of the guide thimbles and fuel rods.

4.1.1 Fuel Grid Loads

To comply with the requirements in Appendix A to U.S. NRC SRP 4.2, the maximum grid responses from the seismic and LOCA accident analyses are combined using the square-root-sum-of-squares (SRSS) method, if applicable. The combined SRSS grid loads are compared to the allowable grid structural strength load (P(crit)) obtained through elevated temperature dynamic tests. The allowable grid strength load P(crit) is the lower 95% confidence level on the true mean as taken from the distribution of measurements on unirradiated production grids at (or corrected to) operating temperature.

Maintaining structural integrity of the grid assembly will insure coolable geometry and guide thimble pattern geometry for control rod insertability. However, if the combined SRSS grid loads exceed the allowable grid structural load (P(crit)), an evaluation of the grid failure locations and conditions must be assessed to insure control rod insertability and coolable geometry.

4.1.2 Control Rod Insertability

To insure the rod control cluster assembly (RCCA) control rod insertability, the structural integrity of the guide thimbles tubes under combined faulted condition must be maintained. The stress acting on the thimble tubes resulting from the combined LOCA and seismic loading conditions must not exceed the allowable stress limit. In addition, when a cylinder shell is subjected to a compressive force or a bending moment, it may axially buckle or deform to become oval. This leads to further distortion on the guide tube and eventually to lose its stable geometrical configuration and flatten. The induced stresses may be within the elastic stress limit. The thimble tube combined compressive and bending stress levels must not exceed the onset stress levels to initiate guide tube buckling due to an external load.

As stated in Section 4.1.1, if the grid loads exceed P(crit), an evaluation of the grid failure location and conditions must be assessed for control rod insertability.

4.1.3 Fuel Rod Integrity

The fuel rod integrity analyses is performed to assure that fuel rods do not fail due to specific causes during normal operation and anticipated operational occurrences. Fuel rod failure due to mechanical fracture means that the fuel rod leaks and that the fission barrier (the cladding) has been breached. A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force such as hydraulic load or a load derived from core plate motion. Fuel rod cladding integrity is assured if the applied stress is less than 90% of the irradiated yield stress at the appropriate temperature.

The stresses for the fuel rods are appropriately combined for the vertical LOCA stresses, seismic stresses, and operational stresses. The combined stress is compared to the allowable limits to assess the integrity of the fuel rod cladding.

4.2 CORE COOLING

Core cooling requirements for LOCA are codified in 10 CFR 50.46. Intact baffle barrel bolting configurations will only influence LOCA PCT calculations to the extent that they affect the amount of fuel grid deformation predicted to result from LOCA loads.

Baffle plate deflections resulting from LOCA loads under reduced bolting configurations which are of sufficient magnitude to cause loss of fuel rod geometric integrity (fuel rods are ruptured to the extent that they can lose fuel pellets) are deemed to be unacceptable without regard to core cooling capability. Baffle plate deflections resulting from LOCA loads under reduced bolting configurations which are of sufficient magnitude to cause loss of control rod insertability for accident analyses which take credit for control rod insertion are deemed to be unacceptable. No analytical remedy currently exists to deal with predicted loss of fuel rod integrity with respect to 50.46 criteria. Accidents which credit control rod insertion might be reanalyzed without control rods, however that is not a recommended approach for determining the acceptability of reduced baffle-barrel bolting configurations although it may be acceptable or required for other purposes. As a result, the acceptability analysis paths defined below presume fuel rod integrity is predicted, and control insertability is predicted for any potentially acceptable baffle-barrel bolting distribution considered.

4.2.1 All Fuel Grid Loads Below Allowables

Where calculated fuel grid loads with reduced baffle-barrel bolting distributions remain below the allowable grid crush strength, there is no impact on the fuel geometry. As a result, there is no change to the predicted accident analysis consequences with respect to meeting the requirements of 10 CFR 50.46. A bolting distribution with this result is automatically considered a success. The margins available in this bolting configuration consist of both the additional load required to cause grid crush, plus the available core cooling margin between the current analysis and the acceptable limits of 10 CFR 50.46. Most Westinghouse plants currently have all fuel grid loads below allowable grid crush strength limits, although there are some plants with calculated grid crush prior to consideration of reduced baffle-barrel bolting distributions.

4.2.2 Some Grid Loads in Peripheral Assemblies Greater than Allowables

In some cases, the baffle plate impact loads on the peripheral assemblies may be calculated to exceed the grid crush strength on the peripheral assemblies, and yet the loads will remain below the grid crush strength for interior assemblies at least one assembly removed from the baffle plates. Under these circumstances, the LOCA analyses are required to address the fuel grid deformation for any potential impact on 10 CFR 50.46 limits. In Westinghouse plants, the peripheral assemblies tend to be at lower power than the remainder of the core. In particular, the adoption of low leakage loading patterns have ensured this to be the case. As a result, the power reduction in peripheral assemblies tends to completely offset any penalty from grid deformation. The accepted methods for demonstrating that this remains true differs from appendix K LOCA analysis tools to best-estimate LOCA analysis tools.

4.2.2.1 Appendix K LOCA Analysis Methodology for Peripheral Assembly Fuel Grid Crush

Appendix K LOCA analysis tools in general tend to be simpler and less sophisticated than best-estimate LOCA analysis tools. As a result, it is more difficult to demonstrate success with appendix K LOCA analysis tools than with best-estimate analysis tools. In the case of Westinghouse 1981 EM + BASH (SATAN, COCO/BASH, LOCBART), Reference 4-2, the hot rod heat up calculation is performed in a single assembly hot channel calculation. As a result, no separate peripheral assembly model exists in the appendix K BASH EM. To treat peripheral assembly grid crush, a separate analysis is performed for a peripheral assembly being modeled as the hot assembly. The fuel assembly flow area is reduced and flow resistance increased in accordance with the predicted grid crush. However, the peaking factor limits applied are those that are confirmed by core design as being applicable to the peripheral assemblies. Since these are substantially lower than the peaking factor limits for the interior assemblies of the core, the reduction in peaking factors offsets the penalty from reduced flow area (grid crush). In this way, peaking factor limits for the peripheral assemblies are established, and the potentially crushed peripheral assemblies are confirmed to be non-limiting when compared to the PCT established for the core. These peripheral assembly peaking factor limits are then confirmed on a cycle specific basis to ensure the safety of each reload.

4.2.2.2 Best Estimate LOCA Analysis Treatment of Peripheral Assembly Fuel Grid Crush

Best-Estimate LOCA (BELOCA) analysis tools are far more sophisticated than appendix K analysis tools. In the case of the WCOBRA/TRAC code approved for BELOCA analysis of Westinghouse plant designs, the model explicitly contains both the hot assembly and peripheral assemblies (among four core regions explicitly modeled). The response to additional comment 30. b) regarding the answers to question 53. b) in Reference 4-3 defines the approved Best-Estimate LOCA treatment of grid crush. "If assemblies on the core periphery are the only assemblies to experience grid crushing, it is concluded that additional analyses are not warranted, and no PCT penalty should be applied. This conclusion is based on taking credit for the low power generation in the peripheral assemblies, and the observation that any flow redistribution which may occur would tend to benefit the in-board assemblies." Therefore, plants analyzed with BELOCA will see no penalty from peripheral assembly grid crush. No credit for grid crush diversion of flow from the periphery to the in-board assemblies would be taken.

4.2.3 Some Fuel Grid Loads in Core Interior Greater than Allowables

At some point, the calculated fuel grid loads may be sufficient to crush both peripheral assembly grids and interior assembly grids. With interior grid crush, a PCT penalty is anticipated using appendix K methodology, and possibly using BELOCA methodology.

4.2.3.1 Appendix K LOCA Analysis Methodology for Interior Assembly Fuel Grid Crush

In using appendix K LOCA methodology to analyze plants with interior assembly grid crush, the hot assembly is assumed to be the one with crushed grids. The analysis is performed using a reduced assembly flow area and increased flow resistance corresponding to the crushed grids.

4.2.3.2 Best Estimate LOCA Analysis Methodology for Interior Assembly Fuel Grid Crush

In accordance with the approved BELOCA methodology (Reference 4-3), the following approach is to be taken for fuel grid crushing which affects in-board assemblies. "If in-board assemblies are also affected, a specific calculation would be performed to assess the effects. The flow resistance of the channel representing the assemblies on the periphery would be increased to reflect the calculated extent of crushing, at the elevations where crushing is calculated to occur. It will be conservatively assumed that the hot assembly is one of the affected in-board assemblies, and the flow resistance for the hot assembly would be modified in a similar manner. Finally, the same treatment would be applied to the average channels, with an assumption that the number of affected assemblies in each average channel is proportional to the number of assemblies in the channel. The calculation would be performed using the reference case transient (power shape 10), and the resulting change in PCT would be applied to the $PCT^{50\%}$ and $PCT^{95\%}$ calculated without grid deformation." If a PCT benefit is calculated, it will not be credited.

4.3 INTACT BAFFLE-BARREL BOLT LOAD ALLOWABLES

4.3.1 Replacement Bolts

The mechanical properties at operating temperatures for SA-193 B8M grade (type 316 CW SS) of the replacement bolts for the baffle-former assembly are provided in Tables A, B, and C of the ASME Code Case N-60-4. Also the mechanical properties for the type 347 SS replacement bolts are provided in Tables I-1.2, I-2.2, and I-3.2 of the 1989 Edition Section III, Division 1 of the ASME Code (Reference 4-4). The ASME code allowables for the replacement bolts are defined as:

<u>Allowable Stress Limits for Normal and Upset Conditions</u>

a. Primary Membrane Stress, P_m

$$P_m = S_m$$

b. Primary Membrane Plus Secondary Membrane, $P_m + Q_m$

$$P_m + Q_m = \text{Lesser of } 0.9\, S_y \text{ or } 2/3\, S_u$$

c. Shearing Stress for Threads, τ

$$\tau = 6.0\, S_y$$

d. Bearing Stress Under Thread, σ

$$\sigma = 2.7\, S_y$$

e. Primary Membrane and Bending Plus Secondary Membrane and Bending, $P_m+Q_m+P_b+Q_b$

$$P_m+Q_m+P_b+Q_b = \text{Lesser of } 1.2\ S_y \text{ or } 8/9\ S_u$$

Faulted Conditions

In accordance with F-1331 of Appendix F of the ASME Code:

a. Primary Membrane, P_m

$$P_m = \text{Lesser of } 2.4\ S_m \text{ or } 0.7\ S_u$$

b. Primary Membrane Plus Primary Bending, $P_m + P_b$

Allowable primary membrane plus primary bending, P_m+P_b, shall not exceed 150% of P_m limit.

4.3.2 Irradiated Bolts

The ASME Code does not provide allowable stress limits for irradiated materials. Using the interpretation of the ASME code, the allowable limits for the irradiated 316 CW SS and the 347 SS bolting materials are established. The test data which was considered for the 316 CW SS and 347 SS bolting materials had significantly lower fluence levels than those for the bolts in the baffle region (5 to 75.0 E+21 n/cm^2 versus 1.6 E+23 n/cm^2) thereby making them conservative limits since the material yield and ultimate strengths increase with increasing fluence levels.

Based on the ASME Code stress limits, it is seen that the S_m value for 347 SS is approximately 90% of the yield strength at operating temperatures (650 degree F), and approximately 50% of the yield for the 316 CW SS. Since, the ASME Code does not specify S_m value for the irradiated materials, it is conservative to consider that for these irradiated bolting materials $S_m = 0.4\ S_y$. The yield stress, S_y, and the ultimate stress, S_u, for these irradiated materials is obtained from the test data.

Having determined the values of S_m, S_y, and S_u, the allowable stress limits described in Section 4.3.1 can be defined and used for normal/upset conditions. For faulted conditions, the stress allowable for primary membrane and bending of irradiated bolt material is taken to be 0.9 S_y. This is consistent with the practice for evaluating fuel rod cladding integrity discussed in Section 4.1.3.

5. REFERENCES

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