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NRC Project Number 686

November 24, 1999

To: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: R.K. Anand, Project Manager
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Subject: Westinghouse Owners Group
Response to NRC Request for Additional Information on WOG Generic Technical Reports: WCAP-14577, "License Renewal Evaluation: Aging Management For Reactor Vessel Internals" (MUHP6110)

Reference: Request For Additional Information (Received from NRC, NRR - June 14, 1999)

Attached are the Westinghouse Owners Group responses to the NRC's Request for Additional Information on WOG Generic Technical Report WCAP-14577, "License Renewal Evaluation: Aging Management For Reactor Vessel Internals." Please distribute these responses to the appropriate people in your organization for their review.

If you have any questions regarding these responses, please contact Charlie Meyer, Westinghouse, at (412) 374-5027, or myself at Wisconsin Electric Power Company, (414) 221-2002.

Very truly yours,

Roger A. Newton

Roger A. Newton, Chairman
LCM/LR Working Group
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RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION ON WCAP-14577, "LICENSE RENEWAL EVALUATION: AGING MANAGEMENT FOR REACTOR VESSEL INTERNALS"

RAI #1 INDUSTRY PLANS

Since the submission of the topical report, the industry has consolidated efforts by the various owner's and other groups, e.g., the PWR Materials Reliability Project (MRP). What is the scope and nature of industry efforts addressing aging management issues related to RVI? What are the schedules for these activities, and how will the results of these industry efforts affect the conclusions and plans addressed in the topical report?

RESPONSE

The major part of the industry effort is the PWR Materials Reliability Project. The overview, mission and objectives of the MRP are:

I. Overview

The purpose of the PWR Materials Reliability Project is to provide a utility-directed oversight structure to proactively address and resolve, on a consistent industry-wide basis, existing and emerging PWR material-related issues. The focus will be on issue resolution and closure and there will be close coordination with and direct participation by the major US NSSS vendors, their associated Owners Groups, NEI and INPO.

II. Mission

The mission of the PWR MRP is to implement and maintain an industry wide program focused on resolving selected existing and emerging performance, safety, reliability, operational and regulatory PWR materials issues. The Executive Group of the PWR MRP will serve as the industry focal point for resolution of issues related to PWR materials degradation management.

III. Objectives

The objectives of the PWR MRP is to provide a utility-directed oversight structure to proactively address and resolve, on a consistent industry-wide basis, selected PWR materials issues. The specific objectives are to:

- Resolve existing and emerging performance, safety, reliability, operational, and regulatory PWR materials issues that meet specific screening criteria,
- With the direct involvement of NEI, serve as the focal point for industry-wide PWR materials-related regulatory issues,
- Fully integrate any work undertaken with OG activities and, where appropriate, ASME Code activities.

The reactor pressure vessel internals is an issue within the MRP and an issues technical group (ITG) has been formed to address this issue. The intent of this group is to ensure 60+ years of safe and reliable plant operation with no surprises or forced shutdowns due to degradation of reactor internals. The

ongoing EPRI Joint Baffle Bolt (JoBB) Program, established in 1996, has been incorporated into the ITG to provide plant inspection and research test data.

Within the WOG and the ITG programs on reactor internals some of the projects under consideration are:

A. Hot Cell Material Testing of Baffle/Former Bolts Removed from Two Lead Plants

B. Hot Cell Material Testing of Baffle/Former Bolts Removed from Ginna

C. Determination of Bolt Operating Parameters of Extracted Bolts

D. Characterization of European and US Baffle/Former Bolt Manufacture, Operation, and Performance

E. Evaluation of the Effects of Irradiation on the IASCC and Mechanical Properties of Core Shroud Materials (SA 316, SA 304, CW 304, and 308 welds)

F. Determination of the Occurrence and the Magnitude of Irradiation Induced Swelling of the Core Shroud

G. Irradiation Embrittlement of Reactor Vessel Internals in PWR's

H. Development of Enhanced Visual Inspection Requirements for Core Internal's Materials Subject to Aging Degradation

I. Preparation of a white paper summarizing available void swelling data and determining the effect on reactor vessel internals components

J. Additional Projects to be added, pending funding

The hot cell testing, item A, is in progress using baffle bolt and locking device materials from two US domestic plants. Additional testing on these materials and on material from a third plant, item B, is being performed separately by the Westinghouse Owners Group as part of the overall industry initiative. This overall program on US materials is currently scheduled to be completed during 2000 and 2001.

MODIFICATIONS TO THE TOPICAL REPORT

The final part of the RAI asks "how will the results of these industry efforts affect the conclusions and plans addressed in the topical report?"

Because this question is similar to the question in RAI #2, the proposed changes in the topical report are addressed in the response to RAI #2.

RAI #2 TECHNICAL PROGRESS

Since almost two years have elapsed from the date that the topical report was submitted, what changes would be made to the report considering technical progress during that time, with particular emphasis on the report sections addressing aging effects and aging management programs (AMP)?

RESPONSE

Since the topical report has been submitted there have been significant amounts of new information generated. The inspection and partial replacement of baffle bolts at four US plants coupled with the testing of bolts removed from three of these plants has provided extensive and important information.

The inspections at the Farley Units 1 and 2 plants revealed no indications in any of the strain hardened Type 316SS baffle bolts. Approximately 200 removed bolts from Farley Unit 1 were subsequently tensile tested. No testing was done on Farley Unit 2 bolts. No indication of pre-existing defects were found on any of the Farley Unit 1 bolts. This conclusion is based on the stress strain curves and on enhanced visual inspection of the bolts after testing. The enhanced visual system consisted of a TV system capable of seeing a 0.002 inch wire. The tensile testing of the bolts was, in effect, a functional test of the bolts' load-bearing capabilities. The data demonstrated the expected increase in yield strength of the bolts with an unexpectedly large elongation to fracture. As an example, yield strengths of >130ksi were measured with >25% elongation to fracture. This demonstrates that there is more than adequate ductility and that the bolts do not behave in a brittle manner.

The bolting material at the Point Beach plant is 347 SS. Nine bolts were found to be non-functional. The bolts at this plant have a significantly greater fluence than those at the Bugey plant when cracks were first observed. Subsequent fluorescent dye testing of the bolts with indications sent to the hot cells did not reveal any cracking, suggesting that the inspection criteria used resulted in a large number of false positives. Tensile testing of the removed bolts resulted in the similar tensile data information as described above for Farley, namely, that the bolts showed the expected increase in yield strength but with greater than expected ductility.

At the Ginna plant the findings were similar to those at Point Beach with a total of five non-functional bolts found. Fourteen of the bolts removed from Ginna were sent to the hot cells for material testing. Detailed microstructural and material property evaluations are currently being performed on the bolts and the 304 SS locking devices from the three plants.

Examination of a 347 SS bolt from a European plant with extensive exposure has shown voids in a small volume of the bolt where the gamma heating and fluence are optimum for the production of voids. A calculation of the degree of swelling showed approximately 0.2% increase in volume in this region. The bolts removed from the US plants will be similarly investigated. This will be considered and discussed further in the response to RAI #8.

MODIFICATIONS TO THE TOPICAL REPORT

The following subsections will be revised due to the industry efforts described above in the responses to RAI's #1 and #2. Although the RAI responses focus on baffle / former bolts, substantial replacement of guide tube split pins has taken place since the report was issued.

2.6.2 Fasteners - Threaded and Pinned

(no change except last paragraph modified as follows)

Specific inspections of baffle / former bolts at several domestic WOG plants has indicated a small degree of degradation (<1.2%). Several of these bolts were removed for subsequent hot cell testing. In addition, a PWR Materials Reliability Project has been implemented by the industry, with a specific Issue Technical Group (ITG) to address reactor vessel internals issues. The ITG and the WOG have implemented a series of tasks including the hot cell testing and characterization of the irradiated bolts removed from the WOG plants.

As new information becomes available from the MRP and WOG tasks, it will be factored into plant specific license renewal applications. This report provides a bounding set of aging mechanisms and effects and the on-going programs are not expected to identify any new issues.

2.6.7 Guide Tubes

2.6.7.1 Guide Tube Assembly

(no change)

2.6.7.2 Guide Tube Support Pins

(no change except last paragraph modified as follows)

Evaluations were subsequently performed by the WOG to investigate indications of degradation that were found on four foreign plants and one domestic plant that has Rev. A pin material. Currently, support pins at a number of WOG plants are being replaced. As noted above, pin degradation does not lead to a loss of intended function. Generally, pin replacement is considered to be a sound maintenance practice to preclude degradation when industry experience indicates that such degradation has been observed.

Changes to other sections of the WCAP regarding separate issues included in these RAI's (e.g., fatigue, thermal embrittlement, swelling, irradiation-assisted stress corrosion cracking, and aging management programs) will be described as part of the responses to those RAI's.

RAI #3 BAFFLE-FORMER BOLTS

In Sections 3.0 and 4.0 of the subject report, WOG, in part, addresses aging management review, aging effects evaluation, and proposed generic aging effects management activities and programs with regard to aging-related degradation of baffle bolts. Subsequent to the submittal of the subject report, WOG had periodic meetings and interactions with the staff from 1997 to the present regarding its ongoing programs and activities to resolve the baffle bolt cracking issues. The ongoing programs and activities include: (1) development and approval of a prescribed analytical methodology for evaluating the acceptability of baffle bolting distributions under faulted conditions; (2) assessment of the safety significance of potentially degraded baffle bolting; (3) performance of baffle bolting inspections/replacements and testing on lead plants; and (4) development of an inspection monitoring and aging management program.

The staff requests that WOG describe their plans and schedules for including the results of the above programs and activities in the aging management of baffle-former bolts.

Because this question is related to the topic of RAI #4, the response is provided together with that for RAI #4 below.

RAI #4 BAFFLE-FORMER BOLTS

In Section 4.2.2 of WCAP-14577, WOG describes the AMP for baffle-former bolts (AMP-4.6), which recommends continued use of the present surveillance techniques. The present surveillance techniques include: (1) visual (VT-3) examination; (2) loose parts detection monitoring; and (3) reactor coolant chemistry monitoring. AMP-4.6 provides options for correcting relevant conditions detected by the VT-3 examination. Further, WOG indicates that baffle-former bolt cracking has not been observed in Westinghouse domestic plants. However, baffle-former bolt cracking has been observed in French and Belgian plants and more recently in Westinghouse domestic plants using volumetric (UT) examination techniques.

Based on the recent experience of volumetric inspection of baffle-former bolts at Ginna and Point Beach Unit 2, the staff requests that WOG propose an alternative program. In lieu of the proposed VT-3 examination, the WOG should consider volumetric inspection.

RESPONSE

Reference [1] provided the safety evaluation (SER) report prepared by the NRC staff to address the acceptability of the Westinghouse methodology to determine number and distribution of intact and functional baffle bolts required to ensure safe plant operation. Application of this approved methodology to determine the number and distribution of required functional and intact bolts has been performed for both plant specific applications as well as for generic plant groups which have similar reactor internals designs. This grouping of plants with similar design features was utilized in order to reduce the number of required evaluations to cover the complete Westinghouse fleet. Plant specific applications of the Westinghouse methodology were performed in support of the inspection and replacement programs at Farley Units 1 and 2 and Point Beach Unit 2. These plant specific applications demonstrated that safe plant operation could be maintained with a reduced number of functional and intact baffle-former-barrel bolts. The results of the generic evaluations completed to date have also shown that in many cases only a small number of intact and functional baffle-former-barrel bolts are required to ensure safe plant operation.

The inspection at Point Beach Unit 2 was done using an angle beam transducer placed on the side of the internal hex socket. Comparing the results of the UT inspections and the mechanical (pull) testing has shown that the inspection techniques of these 347 stainless steel bolts resulted in a large number of false positive indications. However, nine bolts were determined to be non-functional based on observed (full or partial) cracking through the shank. In addition, the final review of the bolt mechanical test data identified two bolts that had full strength capability but had less total elongation and reduction in area than similar bolts. Visual examination of the fracture surface revealed a small in-service crack (~2%) on one bolt but could not confirm a likely similar small crack on the other bolt due to videotape limitations. Both of these bolts were considered to be functional but were judged to have very small in-service cracks on the shank surface at the bolt fillet.

The bolts at Farley Units 1 and 2 have a different head configuration than those at Point Beach Unit 2 and are fabricated of Type 316 CW stainless steel. The bolts inspected at Farley Units 1and 2 did not have any indications. This result was supported by the mechanical (pull) testing performed on the removed bolts. The bolt replacement approach at Point Beach Unit 2 and Farley Units 1 and 2 was to replace the bolts in a number and pattern that would provide acceptable safe plant operation using the replacement bolts alone with no credit taken for the remaining original bolts. As a result, for these plants (Point Beach Unit 2 and Farley Units 1 and 2), no further bolt inspections are planned at his time during the initial operating license periods. If these plants pursue license renewal, then further actions may be considered at that time.

Information on crack initiation rates is provided by the periodic European plant inspection data and the recent data provided by the inspection, testing and replacement at three U.S. plants. The data from the U.S. plants indicate that the cracking is significantly less than that experienced in France. For the plants that have experienced cracking in France, the cracking rate in the more resistant plants is of the order of 2 bolts per cycle. Analysis has demonstrated that between 32% and 80% (depending on the plant design) of the bolts in a plant, randomly distributed, can be cracked with no effect on safety. Thus, if the number of cracked bolts is low, based either on hours of operation or fluence, in comparison to the already inspected plants, then the rate of crack initiation could be sufficiently low that many years of continued operation could be justified without the need for further inspection.

In summary, the Westinghouse Owners Group concurs that degradation of the baffle former bolting is an aging management issue. The results of the recent inspections at Point Beach Unit 2, have shown that through approximately 182,000 hours of operation only a small number of 347 SS bolts have become nonfunctional. The inspection results at Farley Unit 1 have shown that through 144,000 hours of operation none of the 316 CW SS bolts had lost any functionality. The following table summarizes the results of bolt testing.

Plant	Number of Effective Full Power Hours (K)	Non-Functional Bolts (% of Bolts Verified to be Non-functional)
Farley Unit 1	144	0 (0.0%)
Point Beach Unit 2	182	9 (1.2%)
Ginna	195	5 (0.8%)

As a result of the inspections and replacements for Point Beach Unit 2 and Farley Units 1and 2, no further bolt inspections are planned at his time for any other WOG plants during the initial operating license periods. For those WOG plants considering license renewal, further actions will be developed as part of an overall industry program.

The topical report already includes an extensive discussion on different surveillance techniques (Section 4.2.2) including ultrasonic testing. The two Aging Management Programs provided for baffle/former and barrel/former bolts (AMP-4.6 and AMP-4.7) already include the option of "augmented inspections. A single paragraph will be added to the end of Section 4.2.2 as described below.

MODIFICATIONS TO THE TOPICAL REPORT

Section 4.2.2 in WCAP-14577 contains the "Aging Management Program for Baffle/Former and Barrel/Former Bolts (AMP-4.6 and AMP-4.7)."

4.2.2 Aging Management Program for Baffle/Former and Barrel/Former Bolts (AMP-4.6 and AMP-4.7)

(no change except the last paragraph is expanded as follows)

Based on the aging effects identified and current industry initiatives, it is recognized that enhanced inspections beyond present Section XI requirements may be required to manage the effects of aging on Baffle/Former and Barrel/Former bolts for extended periods of operation. Tables 4-7 and 4-8 provide a general path to manage the aging effects on the Baffle/Former and Barrel/Former bolts during the license renewal period. The details of these enhanced inspections will be provided in the aging management programs described in plant specific license renewal applications based on the best information available at that time from the industry programs.

RAI #5 FATIGUE - TIME-LIMITED AGING ANALYSIS

In Section 1.0 of WCAP-14577, WOG indicates that one objective of the report is to identify and evaluate time-limited aging analyses (TLAA). In Section 2.5 of the report, WOG identifies fatigue as the only TLAA related to the RVI, and that the results from current TLAAs have been projected to an extended period of operation. In Section 3.0, WOG provides a summary list (Table 3-3) of fatigue-sensitive RVI components that could reach the fatigue usage limit within a 40- to 60-year time period. WOG indicates that the listed components were identified based on a review of calculated fatigue usage factors for internals components designed to American Society of Mechanical Engineers (ASME) Section III, Subsection NG, hot functional test data, and a comparison of geometric and operating similarities.

The staff requests WOG to provide a list of the TLAAs and a brief summary description of each of the listed analyses used to identify the fatigue-sensitive reactor vessel components. The staff requests WOG to clarify whether the fatigue-sensitive components listed in Table 3-3 apply to all Westinghouse-designed RVI or only to those designed to ASME B&PV Code, Section III, Subsection NG as described in Section 2.5.1 of the report.

For RVI designed prior to the ASME Code adoption of Subsection NG in Section III, what requirements were used for fatigue analysis of the RVI components for the initial operating period? How were these analyses updated to account for the license renewal period?

RESPONSE

The only TLAA for the reactor internal components is fatigue. The fatigue sensitive determination for those reactor internal components described in Table 3-3 applies to the complete Westinghouse fleet. The criteria utilized by Westinghouse for pre-1974 plants was developed internally within Westinghouse and is similar to the subsection NG requirements since many of the Westinghouse designers were members of the ASME code committee that developed the NG subsection. At the present time, fatigue calculations have not been performed and/or have not been updated by Westinghouse to reflect operation in the license renewal period as presented in Figure 4-1 of the WCAP 14577.

The license renewal application will follow the flowchart in Figure 4-1 of WCAP-14577 to assess the acceptability of the RVI components relative to fatigue for the extended period of operation.

The preferred approach, shown in the first part of Figure 4-1, is to demonstrate that the fatigue effects anticipated for the license renewal term are bounded by the fatigue effects anticipated for the original service period. This includes an assessment of the number and severity of the RCS design transients anticipated during the extended period of operation relative to those assumed to occur in the Current Licensing Basis (CLB), and an assessment of the impact of changes to the reactor internals component material properties that may have occurred during the current licensing period.

MODIFICATIONS TO THE TOPICAL REPORT

In WCAP-14577, Section 2.5.2 describes "Industry and Regulatory Actions on Fatigue." This section will be modified to note that GSI-166, "Adequacy of Fatigue life of Metal Components," has been closed and that a separate item, GSI-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life," has been opened to carry forward the issue of metal fatigue during an extended period of operation.

Under Section 4.2, "Additional Activities and Program Attributes, Section 4.2.1, "Aging Management Program for Fatigue (AMP-4.5) includes an extensive discussion on the conservatisms included in the current analyses methods. These discussions will be moved to Section 3.2.10 "Time-Limited Aging Analysis (Fatigue) Evaluation to better reflect the current thinking relative to evaluating the TLAA. Revised Section 3.2.10 and 4.2.1 are modified to read as follows:

3.2.10 Time-Limited Aging Analyses (Fatigue) Evaluation

Section 2.5 identifies fatigue as the only TLAA related to the reactor internals. This section provides the overall approach that licensees will take in addressing the fatigue TLAA for the reactor vessel internals. If the TLAA cannot be dispositioned analytically, options are presented in Section 4.0 to manage the identified aging effects.

3.2.10.1 Mechanism Description

(No change)

3.2.10.2 Aging Effect Evaluation

American Society of Mechanical Engineers (ASME) Code Section III fatigue design procedures use a design fatigue curve that is a plot of alternating stress range (S_a) versus the number of cycles to failure (N). The design fatigue curve is based on the unnotched fatigue properties of the material, modified by reduction factors that account for various geometric and moderate environmental effects.
The fatigue usage factor (U) is defined by Miner's rule as the summation of the damage over the total number of design basis transient types (X), as given by the ratio of expected cycles of that type (n_i) to the allowable number of cycles (N_i) for the stress ranges associated with that transient:

$$U = \sum_{i=1}^{X} \frac{n_i}{N_i}$$

For ASME Code design acceptance, the cumulative usage factor (CUF) calculated in this manner cannot exceed unity (1.0) for the design lifetime of the component.

A recommended flowchart that provides guidance for the management of fatigue in the license renewal period is shown in Figure 4-1. Note that Figure 4-1 addresses the potential effects of the water reactor environment on fatigue through the determination of material property changes. The CLB for fatigue can be maintained in the license renewal period if it can be demonstrated that the nature and frequency of the license renewal period reactor coolant system (RCS) transients are bounded by those assumed in the CLB and that there has been no significant change in reactor internals components' material properties including environmental effects from those assumed in the CLB. However, if this is not possible, then an aging management program for fatigue for each component should be established.

Note that in Figure 4-1 some paths are reversible, that is, decisions can be reversed when another strategy selected, thus allowing a greater flexibility to include new or more complete information (e.g., test data, regulatory acceptability).

The purpose of the component fatigue evaluations is to verify that the component has a cumulative fatigue factor of less than 1.0. It is important to note that, depending on the plant specific application, there are usually several conservatisms included in these fatigue usage calculations. After a determination is made that the number and severity of the RCS transients for the license renewal term are not within the current design basis, then these conservatisms should be evaluated and/or analyzed to increase the present fatigue usage margins. In general, these conservatisms can be found in:

- Definition of RCS design transients
- Enveloping of design loadings
- Calculational methodologies

These conservatisms are discussed in the next subsections.

3.2.10.3 Conservatisms in the Design Transients

The conservatisms built into the RCS primary-side design transients consist of:

- RCS transients that are typically more severe than those experienced during service.
- RCS transients with a larger number of expected occurrences than could reasonably occur during the plant lifetime. For example, the unit loading and unloading between 15 and 100 percent transient has 13,200 to 18,600 postulated design cycles, depending on the plant. This means that a plant will be cycled through these loading and unloading cycles once every day for 40 years, which is unrealistic.

One way to address excess conservatisms in design transients is transient monitoring and cycle counting. It is important to also note that, in general, plants designed in the 1960s and 1970s have fewer RCS design transients defined than those plants designed in the 1980s. Transients that have occurred during operation or are postulated to occur in the licensee renewal term and are not bounded by the CLB transients require re-evaluation on a case-by-case basis.

3.2.10.4 Conservatisms in the Analysis

One of the conservatisms built into the analytical approach consists of performing bounding or enveloping analyses based on bounding RCS design transients and/or loadings. If it can be shown that the calculated design fatigue usage was less than 1.0 by performing a simplified bounding analysis, it is not always necessary to perform additional analysis to show that the fatigue analysis requirements can be met by a larger margin.

In addition, there are two sources of conservatisms inherent in the ASME code fatigue methodology. First, the design fatigue curves contain a factor of 2 on stress range and a factor of 20 on the number of cycles to failure. Second, a substantial margin exists because of conservatisms in the magnitude and frequency of occurrence assumed for the various design basis transients [Refs. 2, 43, and 44].

An additional source of conservatism with respect to high-cycle fatigue for internals components in operating plants is derived on the basis of the fatigue curves for typical internals materials. The stress range for cycle to failure beyond 10^6 cycles is approaching the endurance limit of the material. Typical PWR internals vibration frequencies are in the range of 5 to 10 Hz, so that an operating plant accumulates 10^9 to 10^{10} fatigue cycles in less than 32 full power years. In practical terms, this means that in the absence of changes in loading or configuration, internals components that have not experienced high-cycle fatigue damage during the original licensing period are unlikely to experience high-cycle fatigue damage during the license renewal term.

Conservative calculations use bounding design transients and subsequent design basis stresses to estimate low-cycle fatigue accumulation for the specified transients. High-cycle fatigue analysis is proof-tested by hot functional tests. The rationale for the latter is that a component with high-cycle fatigue susceptibility is identified during hot functional testing, which induces higher flow loads without the resistance of the fuel assemblies. Any high-cycle fatigue issues that have been identified by hot functional testing have

either required subsequent design or operational modifications or analyses to demonstrate the acceptability of the observed behavior.

As a result, the combined low-cycle and high-cycle fatigue usage estimates are conservatively high. These conservatisms are in addition to the ASME Code factor of 2 on stress range and 20 on cycles to failure inherent in the code fatigue curves.

Only those components which exceed a CUF of 1.0 during the license renewal period require aging management.

Table 3-3 summarizes the projected fatigue life of those reactor internals components that could reach a fatigue usage equal to the ASME design limit of 1.0 within the 40- to 60-year time period over the population of all WOG plants based on a conservative approach of extrapolating the number of design cycles by 150%. The projected fatigue life was determined assuming no changes in material properties or component loadings in the extended lifetime period. Projected Fatigue Service Life was based on a CUF=1. Table 3-3 does not represent the actual usage for the license renewal period but rather is a conservative method of screening the internals components that either are, or are not, fatigue-sensitive in the license renewal term. As a result, only those components that are fatigue-sensitive and whose failure would prevent the internals from performing their intended functions have been included in Table 3-3. Based on this screening method, those components not included in Table 3-3 are not considered to be fatigue-sensitive for any WOG plant.

Therefore, with the exception of those reactor internals components identified in Table 3-3 as fatigue-sensitive based on a review of calculated fatigue usage factors for internals components designed to ASME Section III, Subsection NG, a review of hot functional test data, and a comparison of geometric and operating similarities, the effects of fatigue are not significant for the reactor internals components covered in this report. Fatigue-sensitive reactor internals components that require further evaluation are discussed in Section 4.0. Note that those components which are included in Table 3-3 may, or may not, be fatigue sensitive for any specific plant, and should be evaluated on a plant specific basis.

The preferred approach, shown in the first part of Figure 4-1, is to demonstrate that the fatigue effects anticipated for the license renewal term are bounded by the fatigue effects anticipated for the original service period. This includes an assessment of the number and severity of the RCS design transients anticipated during the extended period of operation relative to those assumed to occur in the Current Licensing Basis (CLB), and an assessment of the impact of changes to the reactor internals component material properties that may have occurred during the current licensing period.

Under Section 3.3, "AGING EFFECT MANAGEMENT SUMMARY," Section 3.3.10 is modified as follows:

3.3.10 Fatigue

The effects of fatigue require an evaluation only for reactor internals components which would be projected to exceed a CUF of 1.0 during the extended period of operation. This determination of fatigue-sensitive components should be based on a review of calculated fatigue usage factors for internals components, a review of hot functional test data, and a comparison of geometric and operating similarities. For all other reactor internals components covered by this report, the effects of fatigue are not significant (see Subsection 3.2.10), and an evaluation or an aging management program for this effect will not be required for these components during an extended period of operation. For those components that would be projected to exceed a CUF of 1.0, this effect is discussed in Subsection 4.2.1, AMP-4.5.

Further evaluation of the baffle/former and core barrel/former bolts is discussed in Subsection 4.2.2, AMP-4.6 and AMP-4.7.

4.2 ADDITIONAL ACTIVITIES AND PROGRAM ATTRIBUTES

4.2.1 Aging Management Program for Fatigue (AMP-4.5)

The AMP attributes for fatigue are shown in Table 4-6.

Both main assemblies, upper internals, and lower internals can be removed for inspection. Examination Category B-N-3 of Section XI, Subsection IWB, provides requirements for the visual (VT-3) examination of accessible surfaces of core support structures that can be removed from the reactor vessel. These requirements refer to the relevant conditions defined in IWB-3520.2, which include "loose, missing, cracked, or fractured parts, bolting, or fasteners." Since manifestation of excessive fatigue damage is expected to be fatigue crack initiation on the surface of an affected item, the VT-3 examination is adequate for the detection of significant fatigue damage. If any relevant condition is identified, IWB-3142 provides options for the timely correction of the condition, such as: (1) acceptance by supplemental surface and/or volumetric examination to characterize the indication more accurately; (2) acceptance by analytical evaluation, which may include flaw evaluation and/or more frequent examination of the item; and (3) acceptance by corrective measures, repairs, or replacement.

For those cases when fatigue-sensitive components are essentially inaccessible to inservice examination, in accordance with Examination Category B-N-3, and where the cyclic loadings are sufficiently uncertain to preclude the effective use of detailed fatigue design analysis, alternatives for managing the effects of the age-related degradation are described in Section 4.2. Barrel/former and baffle/former assembly bolts are in this category (see Subsection 4.2.2).

To summarize, while aging management options for fatigue depend on the final United States Nuclear Regulatory Commission (U.S. NRC) position for license renewal, a flowchart, as outlined in Figure 4-1, provides guidance for the management of fatigue in the license renewal term.

The primary step is to demonstrate that the fatigue effects anticipated for the license renewal term are bounded by the fatigue effects anticipated for the original service period. Included in this step is the assessment of the number and severity of the RCS design transients anticipated during the extended period of operation relative to those assumed to occur in the CLB. Also included in this step is the assessment of the impact of changes to the reactor internals component material properties that may have occurred during the current licensing period.

Acceptable results from this step will indicate whether the component(s) can continue to operate during the extended period of operation in conjunction with the requirements of Examination Category B-N-3 of Section XI, Subsection IWB. Unacceptable results from this step would lead to the development of a fatigue license renewal strategy. Development of this fatigue strategy could include:

- Evaluation of conservatisms in the fatigue evaluations to increase fatigue margins
- Review of actual plant RCS primary-loop transient data to assess the actual number and severity of actual plant transients
- Re-evaluation of the cumulative fatigue usage factor for the license renewal term in accordance with the procedures of Section III, Subsection NG-3200
- Performance of a consequences of failure analysis
- Application of risk-based technology
- Application of fracture mechanics technology

- Monitoring, inspection, diagnostics, and testing

TABLE 4-6
AGING MANAGEMENT ACTIVITIES AND PROGRAM ATTRIBUTES FOR FATIGUE (AMP-4.5)

Attribute	Description
Scope	Fatigue effects on fatigue sensitive components
Surveillance Techniques	• Visual examination per ASME Section XI, Subsection IWB, Examination Category B-N-3, and Draft Subsection IWG • Loose parts monitoring • Neutron noise monitoring • Enhanced surveillance per fatigue management program
Frequency	ASME Section XI for visual examination, IWB-2410, -2420, -2430, Draft IWG-2410, -2420, -2430
Acceptance Criteria	Acceptable cumulative usage factor for license renewal term
Corrective Actions	Fatigue management program - see Subsection 4.2.1 and Figure 4-1
Confirmation	Meets ASME Code fatigue requirements

RAI #6 MANAGEMENT OF CRACKING AND NEUTRON IRRADIATION EMBRITTLEMENT

The topical report indicates that effects of cracking due to irradiation embrittlement and IASCC are managed by AMP-4.1 through visual examination, loose parts monitoring and supplemental examination. VT-3 visual examination as required by Examination Category B-N-2/B-N-3 of Subsection IWB of ASME Code Section XI is not adequate for detecting IASCC. The activities for managing IASCC and irradiation embrittlement should be revised to provide a more effective management program. One acceptable program for managing these aging effects is outlined in the draft SER for the Calvert Cliffs license renewal application (Ref. 2).

As an alternative AMP for IASCC and neutron irradiation embrittlement, the draft SER for the Calvert Cliffs license renewal application (Ref. 2) indicates that the applicant has committed to a two-part approach for managing IASCC and neutron embrittlement of RVI components. The first part of this approach is the use of supplemental (enhanced VT-1) examination of RVI components as part of the 10-year ISI program. This supplemental (enhanced VT-1) examination would be performed on the RVI components believed to be the limiting components for cracking, considering both the susceptibility of the component to the aging mechanism, as well as the material properties (in particular the fracture toughness) and the operating stresses on the component. These examinations would apply to all RVI components except for bolting.

The second part of this approach involves consideration of data and evaluations from industry research activities to determine the susceptibility of RVI components to IASCC and neutron embrittlement. Should these data or evaluations indicate that the supplemental (enhanced VT-1) examinations can be modified or possibly eliminated, the applicant would be required to provide plant specific justification to demonstrate the basis for the modification or elimination.

The topical report should be revised to provide a more effective aging management program for IASCC and neutron irradiation embrittlement. An acceptable alternative is the program committed to by the applicant for license renewal of the Calvert Cliffs plant.

RESPONSE

Susceptibility assessments will be conducted to identify limiting components based on the additional data from research activities currently underway. The need for supplemental examinations (VT-1) will be established as part of a 10-year ISI.

Augmented inspections and/or enhanced VT-1 examination of Westinghouse internals can be used on those components where access allows this to be conducted. Examples of the components that would be suitable for this type of inspection are baffle plates and baffle corner plates. Baffle bolts cannot be inspected in such a manner due to the limited access. A revised aging management program will be developed for baffle bolts. Such a program will be developed based on the current examinations of internals components, on the data being developed from the materials from these components, from the PWR Materials Reliability Project (see the responses to RAI's #1 and #2), and on an evaluation of the fluence and loading on these components.

MODIFICATIONS TO THE TOPICAL REPORT

4.1.1 Aging Management Program for Irradiation Embrittlement and Irradiation-Assisted Stress Corrosion Cracking (AMP-4.1)

(The last paragraph will be modified as follows)

Subsection 2.6.2 refers to baffle/former bolt cracking in French and Belgian reactors attributed possibly to IASCC. The fluence level at these bolts in 60 years' total service will exceed the threshold level given in Subsection 3.3.3. Moreover, the bolt stresses are high and ASME Section XI examinations cannot always detect cracking. Therefore, the effects of irradiation embrittlement and IASCC are potentially significant for baffle/former and barrel/former bolts. The barrel/former bolts have been included in this category because they exceed the fluence level threshold, are in the same assembly, and also have high tensile stresses. The aging management program for the baffle/former and barrel/former bolts is discussed in Section 4.2.

As a result of the current examinations of internals components, the data being developed from the materials from these components and from the PWR Materials Reliability Project, and on an evaluation of the fluence and loading on these components, modified guidance for managing the effects of irradiation embrittlement and IASCC may be developed by the industry. Any changes to the current programs will be reflected in plant specific license renewal applications.

RAI #7 AGING EFFECTS AND MANAGEMENT FOR CAST AUSTENITIC STAINLESS STEEL (CASS)

The RVI components fabricated from CASS are potentially subject to a synergistic loss of fracture toughness due to the combination of thermal and neutron irradiation embrittlement. This enhanced loss of fracture toughness is not accounted for within the topical report nor in guidance in revisions to EPRI TR-106092 (Ref. 3). Further, the topical report rules out consideration of thermal embrittlement of RVI CASS components based upon the lack of molybdenum in the materials. The NRC staff does not find this position of considering only thermal embrittlement to be acceptable. A modified screening approach should be used that is similar to that proposed in EPRI TR-106092 (Ref. 3), but also reflecting the potential synergistic effects of neutron irradiation and thermal embrittlement. One acceptable program is outlined below, consistent with the draft SER for the Calvert Cliffs license renewal application (Ref. 2).

The modified approach described in the draft SER for the Calvert Cliffs license renewal application (Ref. 1) consists of either a supplemental (enhanced VT-1) examination of the affected components as part of the applicant's 10-year ISI program during the license renewal term, or a component-specific evaluation to determine the susceptibility to loss of fracture toughness. The proposed evaluation will look first at the neutron fluence of the component. If the neutron fluence is greater than 1×10^{17} n/cm^2 (E > 1 MeV), a mechanical loading assessment would be conducted for the component. This assessment will determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough to preclude fracture of the component, then the component would not require supplemental inspection. Failure to meet this criterion would require continued use of the supplemental (enhanced VT-1) inspection. If the neutron fluence is less than 1×10^{17} n/cm^2 (E > 1 MeV), an assessment would be made to determine if the affected component(s) are bounded by the screening criteria in EPRI TR-106092 (Ref. 3), modified as described below. In order to demonstrate that the screening criteria in EPRI TR-106092 (Ref. 3) are applicable to RVI components, a flaw tolerance evaluation specific to the RVI would be performed. If the screening criteria are not satisfied, then a supplemental (enhanced VT-1) inspection will be performed on the component.

The CASS components should be evaluated to the criteria in EPRI TR-106092 (Ref. 3) with the following additional criteria:

- *Statically cast components with a molybdenum content meeting the requirements of SA-351 Grades CF3 and CF8 and with a delta ferrite content less than 10 percent will not need supplemental examination.*
- *Ferrite levels will be calculated using Hull's equivalent factors or a method producing an equivalent level of accuracy (±6 percent deviation between measured and calculated values).*
- *Cast austenitic stainless steel components containing niobium are subject to supplemental examination.*
- *Flaws in CASS with ferrite levels less than 25 percent and no niobium may be evaluated using ASME Code IWB-3640 procedures.*
- *Flaws in CASS with ferrite levels exceeding 25 percent or niobium will be evaluated using ASME Code IWB-3640 procedures. If this occurs, fracture toughness data will be provided on a case-by-case basis.*

Components that have delta ferrite levels below the screening criteria have adequate fracture toughness and do not require supplemental inspection. Components that have delta ferrite levels exceeding the

screening criteria may not have adequate fracture toughness, as a result of thermal embrittlement, and do require supplemental inspection.

The topical report should be revised to provide a more effective aging management program for cast austenitic stainless steel. An acceptable alternative is the program committed to by the applicant for license renewal of the Calvert Cliffs plant.

RESPONSE

The possible synergistic interaction of thermal and neutron embrittlement of CASS needs to be carefully considered with respect to the possible embrittling mechanisms involved and the available data for both types of embrittlement.

It appears that the fluence level of 10^{17}n/cm^2 was taken from the data for the onset of embrittlement in ferritic pressure vessel steels. This threshold is expected to be much higher in stainless steel weld and base metal materials. The primary mechanism for the embrittlement in these steels is the precipitation of a copper rich phase with possible contributions from nickel and phosphorous. The mechanism of thermal embrittlement below 500°C in the delta ferrite of CASS or of austenitic welds is primarily due to the spinodal decomposition of the chromium rich ferrite to produce variations in chromium content in the ferrite which is referred to as alpha prime embrittlement. There is no copper in these materials. In other words, the mechanisms of embrittlement are quite different for the two different types of ferrite. The data for many welds irradiated at intermediate temperatures (370°C to 430°C), even those with high molybdenum, show that "exposures up to 1 dpa have no significant effect on fracture resistance (Ref. 4)." For the PWR spectrum 1 dpa is 7 x 10^{20} n/cm^2. The literature data on castings is limited. (Ref. 5) That which is available indicates that the fracture toughness of an SA351 CF8 casting with 15% delta ferrite behaves in a similar manner to a 308SS weldment with 7% delta ferrite when irradiated (Ref. 6). Thus, any effects of thermal and neutron embrittlement are not expected until significant fluence at temperature is accumulated.

The major use of CASS in the internals of some of the Westinghouse PWR is the lower core support casting. The fluence at 32 EFPY for this component is typically less than 10^{19} n/cm^{2}and at 48 EFPY will still be less than the 1 dpa fluence cited above. Therefore, if the casting is acceptable based on the guidelines of EPRI TR-106092, no additional concerns should be addressed due to neutron fluence. The temperature of operation of the lower core support casting is expected to be close to the core inlet temperature (~520°F) which is significantly less than that at which the above referenced data was generated leading to a degree of conservatism in the argument. A rough calculation of the effect of lowering the temperature from 370°C where the reference data is cited to the conservatively expected temperature of 330°C (utilizing an estimated activation energy of 30 K cal/mol) on the embrittlement suggests that the susceptibility is lower at least by a factor of 4. The casting will be evaluated in accordance with the guidelines of TR-106092 as modified according to the additional criteria listed in RAI #7.

The only other place where CASS is used in some of the internals of a Westinghouse PWR is as a mixing vane device in the upper internals. It is expected that the loading on these components is sufficiently low that fracture will be precluded. These devices have been determined to not perform any intended function (see Table 2-1) and therefore aging management review is not required.

MODIFICATIONS TO THE TOPICAL REPORT

3.2.8 Thermal Aging

3.2.8.1 Mechanism Description

(no change)

3.2.8.2 Aging Effect Evaluation

The cast austenitic stainless steel lower core support forging is exposed to temperatures that could potentially lead to eventual thermal aging embrittlement, provided that the term of exposure is sufficiently long and that the other factors that control the extent of embrittlement (e.g., casting process, delta ferrite, and material chemistry) are unfavorable. The degradation of cast duplex stainless, if it occurs, is manifested by a decrease in fracture toughness, tearing modulus, and impact strength at room temperature. The fracture toughness, tearing modulus, and impact strength show only a moderate decrease at operating temperatures, 554°F to 617°F.

A review of thermal aging effects shows that cast austenitic stainless steel with ferrite contents as low as 10 percent are susceptible to thermal aging. Further, the structural welds in forged material could be susceptible to thermal aging. As stated above, all the cast duplex stainless steel reactor internals in the Westinghouse-designed NSSS are made from CF-8 or CF-8A.

While CF-8 material is susceptible to thermal aging at operating temperatures (354°F to 617°F), the remaining toughness is high with a Charpy value of 64 ft-lb and fracture toughness values of 750 in.-lb/in.^{2} for J_{Ic} and 3000 in.-lb/in.^{2} for J_{max} at room temperature for material with a high ferrite content (17 percent). Fracture mechanics evaluation of primary piping demonstrates structural integrity with Charpy impact energies as low as 2 ft-lb. Increasing the thermal aging temperature accelerates the thermal aging degradation of the fracture toughness of austenitic cast stainless steels. Using test results, higher temperature thermal aging data can be used to extrapolate to longer periods of time for thermal aging at lower thermal aging temperatures. Using an acceleration factor of 15 (which is conservative) for a thermal aging time for 752°F versus 617°F can project out to 450,000 hours of operation. CF-8 cast stainless steel is expected to have a Charpy value in excess of 28 ft-lb at the end of 60 calendar years or 48 effective full power years (EFPY).

Evaluations of cast internals components demonstrate that the effects of thermal aging for the reactor internals components are not significant and an evaluation or an aging management program for this effect will not be required during an extended period of operation.

RAI #8 SIGNIFICANCE OF VOID SWELLING

The topical report dismisses change of dimension of the RVI components due to void swelling as a significant aging effect due to (1) core management reducing neutron exposure levels such that the effects of swelling are either not significant or are limited to a small number of baffle-barrel region bolts, and (2) no degradation in ability of the structures in this region from meeting their intended functions. The NRC staff finds this evaluation of void swelling to be inadequate. EPRI TR-107521 (Ref. 7) cites one source which predicts swelling as great as 14 percent for PWR baffle-former assemblies over a 40-year plant lifetime. The issue of concern is the impact of change of dimension due to void swelling on the ability of the RVI to perform their intended function.

The WOG should address the following:

- *How much of a change in dimension would be required before the internals would not be able to meet their intended function?*
- *What programs are the WOG participating in that will evaluate the impact of the void swelling on the intended function of the internals?*
- *When will these programs provide data to determine whether void swelling could impact the intended function of the internals?*

Should it be determined that change of dimension by void swelling can impede the ability of the RVI to perform its intended function, then an appropriate aging management program would be required to assure that the need for corrective actions can be properly identified.

RESPONSE

Westinghouse conducted a program with French PWR units to make an assessment of the effect of dimensional changes of critical components on their functionality. Westinghouse believes that the swelling estimate of 14% in Ref. 7 is overly conservative for PWRs. The basis for this will be included in the topical report.

Since EPRI TR-107521 was written there have been significant new findings and re-evaluations of the data. One of the major items is the realization that the fluence levels on US plants are significantly less than first considered due to the use of low leakage core management strategies. The WOG is participating in the MRP programs and through the MRP is a major contributor and an active participator in the MRP task where void swelling is of major interest. At the last MRP meeting, it was reported that voids were observed in the center of the shank, just below the head of baffle bolts removed from a European plant. This is the region where the gamma heating raises the temperature of the bolt to that where it is possible that the fluence can cause voids. The swelling was calculated to be approximately 0.2% in this region of the bolt. Stresses within the bolt from this differential swelling effect are limited to relatively low levels by irradiation creep. The WOG continues to actively monitor new information in the area of swelling through this and other organizations, e.g., the ICG-EAC.

Also within the MRP there are programs to specifically evaluate the impact of void swelling on reactor components and an industry position is to be prepared for the MRP by the three owners groups collaborating with Pacific Northwest Laboratories. In addition, the WOG and the MRP are funding detailed metallographic examinations of the bolts and locking devices removed from three US plants with the search and recording of voids using transmission electron microscopy being a specific part of the project.

The data on swelling are being evaluated at the moment and more data are being generated as part of the previously listed WOG and MRP programs. At present there have been no indications from the different bolt removal programs or from any of the other inspection and functional "evaluations" (e.g., refueling) that there are any discernible effects attributable to swelling. The industry position to consider the accumulating microscopic data, the engineering evaluations of the ramifications of swelling and the field observations is presently scheduled to be complete in 2001.

MODIFICATIONS TO THE TOPICAL REPORT

The information currently in the report on swelling (Section 3.1) will be moved to the section on Aging Management Review. Current Section 3.2, AGING MANAGEMENT REVIEW, will become Section 3.1, and the section on swelling will become section 3.1.11. The previous section 3.3 AGING EFFECT MANAGEMENT SUMMARY, will become Section 3.2. A new section for swelling will be added as 3.2.11

The revised numbering scheme for the swelling information is shown here:

3.1.11 Swelling

In addition to the aging effects identified for the reactor internal components in Section 2.7, swelling has been postulated from laboratory testing for LMFBRs and is discussed in the following subsections.

3.1.11.1 Mechanism Description

Swelling, frequently referred to as cavity swelling or void swelling, is defined as a gradual increase in size (dimensions) of a given reactor internals component. Reactor internals components are fabricated from materials that contain nickel and a small amount of boron. Under reactor internals irradiation conditions, helium is generated in these materials by nuclear transmutation reactions. Cavity or bubble nucleation is accounted for by the helium-vacancy cluster evolution, while void formation occurs when helium bubbles grow beyond a critical size. Helium bubbles have diameters of 2 to 3 nm or less while voids have diameters larger than 4 nm. Helium helps to stabilize small vacancy clusters and promotes nucleation of voids. After helium bubble nucleation, if the temperature is high enough, the helium bubbles grow to a critical diameter. At the critical diameter, the helium bubbles convert to bias-driven voids. Void formation results in the swelling of the material.

3.1.11.2 Aging Effect Evaluation

The effect of irradiation on stainless steel has been extensively studied in programs directed toward their use in LMFBRs, also referred to as liquid metal reactors (LMRs). These studies identified three major materials problems: void swelling, irradiation creep, and radiation-induced embrittlement. The data for PWR applications are extremely limited, and the use of LMFBR data is complicated by the effects of irradiation temperature, displacement rate, and displacement effectiveness. LMFBRs operate at higher temperatures and high displacement rates relative to those for PWRs.

During the past 30 years, swelling of PWR internals components was not considered a significant age-related degradation mechanism. However, Garner, et al. [Ref. 18] concluded that, based on LMFBR data, end-of-life exposures of some PWR internals will lead to significant levels ($\geq$10 percent) of swelling. Foster, et al. [Ref. 19] concluded that at the approximate reactor internals end-of-life dose of

100 dpa, swelling would be less than 2 percent at irradiation temperatures between 572°F and 752°F. To date, field service experience in PWR plants has not shown any evidence of swelling.

Original core loading pattern strategies, known as "out-in" loading patterns, consisted of placing fresh fuel in all peripheral assembly core locations and burned fuel in all of the inboard assembly core locations. Peripheral assemblies are defined as those with one or two faces or one corner adjacent to the core baffle plates. Utility interest in reducing the rate of PWR vessel embrittlement by reducing the incident fast neutron flux to the reactor vessel through fuel management and core periphery modifications has grown in recent years. In addition, the fuel cycle cost advantages of reduced core neutron leakage coupled with higher permissible core power peaking limits have resulted in fuel management strategies with significantly lower power levels in the peripheral fuel assemblies than was the case with the traditional out-in fuel management. This low leakage loading pattern places burned fuel in some of the peripheral assembly locations and most of the fresh fuel assemblies in interior core positions.

Table 3-1 presents estimates of the representative ranges of neutron irradiation for the baffle plates for both types of loading patterns and at either the 40 or 60 year design life.

The relative vertical displacements of the baffle plates and the core barrel due to swelling will be defined by the average irradiation on the components. Therefore, bolt stresses from swelling due to the relative motion of the baffle plates and the core barrel can be described by the average irradiation values in Table 3-1. Using the data from Table 3-1 and Reference 19, the differential swelling could approach 1 percent at 60 years life for the out-in loading pattern and 0.5 percent for the low leakage loading pattern.
The maximum irradiation values in Table 3-1 will be the values that cause baffle/former bolt loadings due to local swelling. This localized effect results from the differential swelling between the 304 stainless steel baffle plate and the bolt materials, which exhibit much less swelling. Actual data to evaluate this are scarce but the available data suggest that this swelling could approach 3 percent at 60 years life for the out-in loading pattern and 1-2 percent for the low leakage loading pattern for a limited number of baffle/former bolts.

It is important to note that:

- Estimates using the available data indicate that the maximum swelling in PWR internals components is significantly less than the 10-percent value predicted by Garner, et al. [Ref. 18]
- The continued utilization of low leakage loading patterns will reduce the irradiation dose and hence the differential swelling and loadings on the bolts in the baffle/barrel region
- The magnitude of swelling will be mitigated by stress relaxation and irradiation creep within the bolt
- There exists a limited amount of data to estimate the swelling percentage as a function of dpa level

Plants now use some form of low leakage loading pattern for their core management strategy. Therefore, it is judged that swelling of the baffle plates, former plates, and core barrel will not prevent them from performing their intended function during the license renewal term.

Moreover, careful core management strategies can reduce the dpa dose levels in the baffle/barrel region structures to levels in which the effects of swelling on the loadings of the baffle/barrel bolts are either not significant or are limited to a small number of bolts. In either case, the intended functions of the baffle/former and barrel/former bolts would not be significantly degraded by swelling.

Industry data of swelling are currently being evaluated as part of WOG and MRP programs. At present, there have been no indications from the different bolt removal programs or from any of the other inspections and function evaluations that there are any discernible effects attributable to swelling, An

industry position to consider the accumulated data, engineering evaluations of the ramifications of swelling, and the field observations is presently scheduled to be complete in 2001.

The following New Section on aging Effect Management will be added to the topical report:

3.2.11 Swelling

The effects of swelling can be potentially significant for those components which experience significant neutron irradiation while operating at elevated temperatures. However, actual plant operations do not appear to produce the conditions necessary for significant swelling. Fuel management schemes to reduce neutron leakage from the core have reduced one of the major factors contributing to swelling, and mechanisms such as creep and stress relaxation serve to reduce some of its adverse effects. It is judged that any actual swelling of the baffle plates, former plates, and core barrel will not prevent them from performing their intended function during the license renewal period.

The data on swelling are currently being evaluated and more data are being generated as part of WOG and MRP programs. At present there have been no indications from the different bolt removal programs or from any of the other inspection and functional "evaluations" (e.g., refueling) that there are any discernible effects attributable to swelling. The industry position to consider the accumulating microscopic data, the engineering evaluations of the ramifications of swelling and the field observations is presently scheduled to be complete in 2001.

RAI #9 ASME CODE LIMITATIONS ON STRESSES OR DEFORMATIONS

Section 2.4.1.2 of the topical report describes ASME Code limitations on stresses or deformations required to ensure a safe and orderly reactor shutdown in the event of an earthquake and major loss-of-coolant incident loading conditions. Describe the specific current licensing basis limitations, and demonstrate that the material properties of the RVI components will continue to meet these limits under the neutron irradiation embrittlement conditions which will exist at the end of the license renewal period.

RESPONSE

Bolts removed as part of the inspection and bolt replacement program in the lead plants have been subjected to various examinations and testing. This testing has shown that there is considerable ductility remaining in the irradiated material. The percentage elongation in the irradiated bolts is 30 to 60 percent. Therefore, these bolts would not be expected to fail in a brittle manner. The yield and ultimate strength found in the removed bolts were found to be within the expected ranges for irradiated material thus demonstrating the continued acceptability of these materials in the license renewal term.

MODIFICATIONS TO THE TOPICAL REPORT

None

RAI #10 INTENDED FUNCTIONS OF THE REACTOR VESSEL INTERNALS

Section 2.2 of the topical report describes the intended functions of the reactor vessel internals on system level. The staff believes that the rule [10 CFR 54.21(a)(3)] requires that a renewal applicant demonstrate that the intended functions are maintained at the basic structure or component level. The report should, therefore, include RVI component-level intended functions which may include, but not be limited, to the following intended functions:

- *Provide support and orientation of the reactor core (i.e., the fuel assemblies).*
- *Provide support, orientation, guidance, and protection of the control rod assemblies.*
- *Provide a passageway for the distribution of the reactor coolant flow to the reactor core.*
- *Provide a passageway for support, guidance, and protection for incore instrumentation.*
- *Provide a secondary core support for limiting the core support structure downward displacement.*
- *Provide gamma and neutron shielding for the reactor pressure vessel.*

RESPONSE

Most of the identified functions are already identified in the topical report Executive Summary and will be incorporated into Section 2.2.

MODIFICATIONS TO THE TOPICAL REPORT

2.2 COMPONENTS OF THE REACTOR INTERNALS SUBJECT TO AN AGING MANAGEMENT REVIEW

The reactor internals support the following intended functions:

- Provide the capability to shut down the reactor and maintain it in a safe shutdown condition
- Prevent failure of all nonsafety-related systems, structures, and components whose failure could prevent any of these functions
- Ensuring the integrity of the reactor coolant pressure boundary (bottom-mounted instrumentation flux thimbles only)

These component intended functions support the same RCS intended functions. In addition, since the bottom-mounted flux thimbles have been included in the scope of this report, the flux thimbles must ensure that the integrity of the reactor coolant pressure boundary is maintained. (Note that the inclusion of the flux thimbles in the scope of this report is arbitrary. They are the only pressure boundary component included here, and on a plant specific basis, could also be evaluated together with other pressure boundary components).

Specific functions can also be defined for the individual subcomponents comprising the reactor vessel internals as follows:

1. Provide support and orientation of the reactor core (i.e., the fuel assemblies).
2. Provide support, orientation, guidance, and protection of the control rod assemblies.
3. Provide a passageway for the distribution of the reactor coolant flow to the reactor core.
4. Provide a passageway for support, guidance, and protection for incore instrumentation.

5. Provide a secondary core support for limiting the core support structure downward displacement.

6. Provide gamma and neutron shielding for the reactor pressure vessel.

Table 2-1 provides a matrix of the reactor vessel internals intended function (by number) for each of the reactor internals subcomponents that specifically support each intended function.

The reactor internals components listed in Table 2-1 that perform an intended function in a passive manner and which are long-lived are subject to an aging management review (see Table 2-2).

In order to provide a note of clarification to WOG utilities who will be using / referencing this report, the following note will be associated with the bottom-mounted incore instrumentation flux thimbles in Section 1.2 REACTOR INTERNALS SCOPE:

- Bottom-mounted incore instrumentation columns and flux thimbles *

* The inclusion of the flux thimbles in the scope of this report is arbitrary. They are the only pressure boundary component included here, and on a plant specific basis, could also be evaluated together with other pressure boundary components.

TABLE 2-1
SUMMARY OF REACTOR INTERNALS SUBCOMPONENTS
SUPPORTING IDENTIFIED INTENDED FUNCTIONS

Part or Subcomponent	Intended Function (see Section 2.2)					
	1	2	3	4	5	6
Lower core plate and fuel alignment pins	Y	N	Y	Y	Y	N
Lower support forging or casting	Y	N	Y	Y	Y	N
Lower support columns	Y	N	N	Y	Y	N
Core barrel and core barrel flange	Y	N	Y	N	N	Y
Radial support keys and clevis inserts	Y	N	N	N	N	N
Baffle and former plates	Y	N	Y	N	N	Y
Core barrel outlet nozzle	N	N	Y	N	N	N
Secondary core support	Y	N	Y	Y	Y	N
Diffuser plate	N	N	Y	N	N	N
Upper support plate assembly	N	Y	N	N	N	N
Upper core plate and fuel alignment pin	Y	N	Y	N	N	N
Upper support column	N	Y	N	Y	N	N
Guide tube and flow downcomers	N	Y	N	N	N	N
Upper core plate alignment pin	N	Y	N	N	N	N
Holddown spring	N	N	N	N	N	N
Head and vessel alignment pins	N	Y	N	N	N	N
Control rod	N	N/A	N	N	N	N
Drive rod	N	N/A	N	N	N	N
Neutron panels/thermal shield	N	N	N	N	N	Y
Irradiation specimen guide	N	N	N	N	N	N
BMI columns and flux thimbles	N	N	N	Y	N	N
Head cooling spray nozzles	N	N	Y	N	N	N
Upper instrumentation column, conduit, and supports	N	N	N	Y	N	N
Mixing device	N	N	N	N	N	N
Bolts and locking mechanisms	Y	Y	Y	Y	Y	N
Specimen plugs	N	N	N	N	N	N

TABLE 2-2
SUMMARY OF REACTOR INTERNALS SUBCOMPONENTS REQUIRING AGING MANAGEMENT REVIEW

Part or Subcomponent	Aging Management Review Required?
Lower core plate and fuel alignment pins	YES
Lower support forging or casting	YES
Lower support columns	YES
Core barrel and core barrel flange	YES
Radial support keys and clevis inserts	YES
Baffle and former plates	YES
Core barrel outlet nozzle	YES
Secondary core support	YES
Diffuser plate	YES
Upper support plate assembly	YES
Upper core plate and fuel alignment pin	YES
Upper support column	YES
Guide tube and flow downcomers	YES
Upper core plate alignment pin	YES
Holddown spring	NO
Head and vessel alignment pins	YES
Control rod	NO
Drive rod	NO
Neutron panels/thermal shield	YES
Irradiation specimen guide	NO
BMI columns and flux thimbles	YES
Head cooling spray nozzles	YES
Upper instrumentation column, conduit, and supports	YES
Mixing device	NO
Bolts and locking mechanisms	YES
Specimen plugs	NO

REFERENCES FOR THESE RAI RESPONSES

1. References Thomas H. Essig to Lou Liberatori, 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), November 10, 1998.
2. "Safety Evaluation Report Related to the License Renewal of Calvert Cliffs Nuclear Power Plant, Units 1 and 2," dated March 1999.
3. EPRI Technical Report TR-106092, "Evaluation of Thermal Aging Embrittlement for Cast Austenitic Stainless Steel Components in LWR Reactor Coolant Systems," Electric Power Research Institute, September 1997.
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