A Probabilistic Approach to Baffle Bolt IASCC Predictions

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Abstract. A methodology for evaluating the probability of baffle-former bolt cracking was developed for applicability to PWRs. The methodology is based upon IASCC test results for the stainless steels most commonly used for baffle-former bolts (e.g., Type 304 SA, Type 347 SA, and Type 316 CW) and predictions of the representative stress patterns in those bolts that were developed as inputs to the model. The predictive methodology for IASCC is based on a single parameter that was developed to incorporate the combined effects of dose and stress ratio (applied stress divided by yield strength, where the yield strength includes irradiation hardening) plus a Weibull statistical distribution that is defined in terms of that parameter.

Baffle-former bolt cracking has been observed in a number of PWRs, and these incidents have raised concerns about the likelihood of future cracking or failures. In this damage model, IASCC “failure” is defined when the component becomes fully susceptible to stress corrosion cracking; that is, after a certain level of irradiation and sustained stress. The length of time required for the material to become fully susceptible was determined from IASCC test data. IASCC crack initiation, which is defined to be equivalent to failure as noted above, is defined by an IASCC susceptibility curve from the test data that relates applied stress and cumulative neutron dose. The dose duration under constant stress is interpreted as the incubation time needed to make the material susceptible to stress corrosion cracking, after which crack initiation and propagation to full rupture under constant stress will occur within a relatively short time (hundreds of hours). The IASCC failure model uses the calculated stress in a material such as the baffle-former bolt and calculates a damage index as the ratio of the current applied stress to the allowable stress as a function of irradiation dose. IASCC initiation data for various irradiated bolting materials was obtained from test results reported in the literature, and plotted as percent of irradiated yield strength versus irradiation dose. This method was used to determine the probability of IASCC occurring at various stress levels, using a Weibull fit of the cumulative failure probability vs. IASCC ratio.

To benchmark the model, bolt-by-bolt stresses in a typical PWR were estimated and the accumulated fluence or dose level were used with the model to predict the probabilities (or numbers) of baffle-former bolt failures due to IASCC over time (i.e., at various Effective Full Power Years). The resulting predictions were compared to actual field experience with bolt cracking in several operating PWRs. The model provides a probabilistic estimate of the
number of cracked bolts that might be expected to be found during any future refueling outage with inspections of the baffle-former bolts. Such a priori knowledge is important because the plan for inspection of the baffle-former bolts may require additional contingencies depending on the likely outcome.

1. Introduction
Failures of baffle-former bolts in pressurized water reactors (PWRs) have occurred in a number of plants in France, Belgium, Japan, and the United States. Those failures have most commonly been attributed to IASCC of the bolts after extended exposures (more than 20 effective full power years, EFPY). However, methods for planning inspections to avoid failures of those bolts are lacking.

The objectives of this work were to:

• Use probabilistic methods and a statistical approach to evaluate cracking in Type 304/347 SA and Type 316 CW baffle-former bolts
• Determine failure probabilities due to IASCC as a function of stress ratio and fluence
• Apply the model to several PWRs to correlate with plant data
• Use the model to predict trends and estimate numbers of baffle-former bolt failures for future inspection planning.

A methodology based upon IASCC test results for the stainless steels most commonly used for baffle-former bolts (Type 304 SA, Type 347 SA, and Type 316 CW) and approximations of the development of stress and irradiation dose patterns in those bolts was developed. That methodology developed a single parameter that incorporates dose and stress ratio (applied stress divided by yield strength, where the yield strength includes irradiation hardening), then defined a statistical distribution of IASCC susceptibility versus that parameter using a Weibull distribution. Predicted stress results from existing prior analyses, adjusted to fit the specific geometry of the reactor internals and fluence histories of three US plants were then evaluated for one octant of the core. Those stress results were used to define the IASCC parameter for each bolt as a function of time. The probability of failure of each bolt (where failure is defined as IASCC initiation) was then determined from the Weibull distribution. The overall numbers of cracked bolts predicted in an octant were then computed from those individual probabilities.

1. Background
1.1 Reactor Design Characteristics
In PWRs, the reactor core is surrounded by a series of vertical baffle plates that form a boundary for coolant flow to the core. These plates also provide lateral support for the fuel assemblies. The baffle plates are supported by horizontal plates (typically seven or eight) called formers or former plates that are bolted to the outside of the cylindrical core barrel. The baffle plates are in turn bolted to the former plates. The bolts that connect the baffle plates to the formers are called the baffle-former bolts. The overall arrangement of the baffle-former bolt configuration at the plane of a former plate is shown in Figure 1. In one of the PWR examples evaluated in this paper, there are a total of 832 baffle-former bolts, 104 bolts around the periphery of the baffle plates in each octant; 13 bolts at each of the 8 former levels in each octant as noted by the designations A through M Figure 1.

Some of the reactor coolant flow is diverted through the cavity between the inside of the core barrel and the outside of the baffle plates. In some units, the flow is downward (i.e.,
downflow). For “downflow” plants, the differential pressure acting across the baffle plate varies from a large value near the top of the core to a small value near the bottom of the core and, in some cases, can increase the tendency for jetting, which could have adverse effects on the peripheral fuel rods. To minimize the potential for gaps occurring in the “downflow” plants, edge bolts were added between adjacent baffle plates as shown in Figure 1.

![Diagram of Baffle Former Configuration](image)

**Figure 1. Baffle Former Configuration (slightly more than one octant)**

### 1.2 Materials
The baffle-former bolts in older US PWRs, for example, early Westinghouse-designed plants, are made from Type 347 stainless steel in the solution annealed condition (SA). Most of the French-designed plants and the later Westinghouse-designed plants used Type 316 SS in the cold worked condition (CW) for the bolting material. Type 347 (SA) was also used for baffle-former bolts in a few Japanese plants, and Type 304 (SA) was also used in a few plants.

Type 316 (CW) exhibits higher yield strength than Type 347 (SA) or Type 304 (SA) in the unirradiated condition. A comparison of the unirradiated and irradiated yield strengths between Type 316 CW) and Type 304 (SA) is shown in Figure 2 [1]. The room temperature yield strength properties and irradiation hardening response for Types 347 (SA) and 304 (SA) materials are very similar (green triangle and blue diamonds, respectively, in Figure 2). Figure 2 also shows that all three irradiated materials exhibit essentially identical yield strengths.
1.3 **Loads**
Baffle-former bolt loads will change over time, essentially following the sequence listed below:

- Pre-load (near unirradiated yield strength)
- Initial thermal relaxation (a slight, one-time decrease from the pre-load)
- Pressure differential (a slight increase from the relaxed pre-load during the entire extent of plant operation)
- Differential thermal expansion due to gamma heating and row to row variations (an additional increase in load during the entire extent of at-power operation)
- Irradiation relaxation (decreasing the applied load over the life of the bolt)
- Void swelling of baffles (increasing the bolt load at high dose)

Baffle-former bolts begin their service history with a design preload that induces stresses in the bolts in the elastic range of the unirradiated material. For the sample cases discussed in this paper, the bolt material is Type 347 SA with similar properties to Type 304 SA (SS-304), which has unirradiated yield strength of \( \sim 200 \text{ MPa (29 ksi)} \) at 330\(^\circ\)C (626\(^\circ\)F). Irradiation hardening raises this value of yield strength to \( \sim 585 \text{ MPa (85 ksi)} \) at an irradiation dose of only 5 dpa. At 10 dpa, the yield strength is raised to \( \sim 690 \text{ MPa (100 ksi)} \), and to a fully saturated value of \( \sim 800 \text{ MPa (116 ksi)} \) within an irradiation dose of 12 dpa. To put these values in perspective, the most highly exposed bolt in a typical PWR experiences a 12-dpa dose within about 8 equivalent full power years (EFPY) of operation. Thus, the bolt material properties become independent of irradiation dose after a service life of 8 EFPY for a high-leakage loading pattern, and about 30-40% longer for low-leakage loading patterns. For plants that generally begin their service life and continue to operate in the high-leakage mode for some number of cycles, and then switch to a low-leakage mode, the bolt material properties will not be affected by the reduced dpa dose rate during the low-leakage operation.

1.4 **Baffle-Former Bolt Cracking**
Cracked baffle-former bolts have been detected in French, Belgian, Japanese, and U.S. plants. In the late 1990s there were a number of failed baffle-former bolts found at Point Beach Unit
2 and Ginna [2]. A number of the bolts from these plants were evaluated in hot cells to determine the cause of the failures. Several possible failure mechanisms were identified for the bolts. The examinations of the extracted bolts (Figure 3) confirmed that the primary cause of failures in these baffle-former bolts was most likely Irradiation Assisted Stress Corrosion Cracking (IASCC) and applied cyclic stress loading [2]. The observation of failed bolts at a number of PWRs has raised the question of whether baffle-former bolts in a particular reactor are likely to fail, when susceptibility to failure becomes a major concern, and when and how to appropriately inspect to characterize the condition of baffle-former bolts.

Figure 3. Examples of Baffle-Former Bolt Cracking

2. Methodology
The methodology presented here was based upon the following steps:
- Review of IASCC test data of typical alloys from the literature
- Development of a generic IASCC trend curve (i.e., Stress Ratio versus Dose)
- Calculation of that stress ratio for each data point and comparison to the trend curve
- Fit of all data point ratios to a Weibull distribution
- Application and comparison of predictions to baffle-former bolt inspection results at Ginna/Point Beach 2 (ca. 1999) and to an additional plant example
In order to develop the probability distribution, a single IASCC susceptibility parameter is needed that reflects both dose and stress. Taking guidance from prior work on the subject [3], the ratio of applied stress to an IASCC susceptibility curve was chosen for this parameter. However, if the dashed IASCC susceptibility curve from Figure 4 were used for this purpose, it would minimize the effect of increasing dose, since that curve saturates at 10 dpa. Since the purpose of this exercise is to develop a probability curve rather than a conservative go/no-go criterion, a trend curve was constructed from the same data, as shown in Figure 5, which effectively separates the majority of the failure and non-failure data, but which doesn’t necessarily bound all of the failure data. Data points that fall on that curve are expected to have a small, but non-zero, probability of failure.

3. Data Sources
A number of experimental programs have generated a significant amount of IASCC data. For example, data from O-Ring and constant load tests on Type 304 (SA) and Type 316 (CW) stainless steels irradiated in operating plants are reported in Fyfitch, et al. [3]. Although data for Type 347 SS were not included specifically, slow strain rate test (SSRT) data indicate that the IASCC susceptibility of Type 347 is similar to that of Type 304. All stresses were normalized to the irradiated yield strength, and stress levels from all tests were assumed to be nominal (P/A) stresses (i.e., on smooth specimens) with no stress concentration factors.

![Figure 4. IASCC Test Data [3]](image)

An IASCC Trend Curve, Figure 5, was developed from the data in Figure 4. That trend curve is similar to the threshold curve in Figure 4, but was modified to bound most (but not all) failure points. The intent is not a Go-NoGo threshold, but rather a tool to estimate the probability of IASCC. An IASCC “Susceptibility Ratio” was computed for each data point as the ratio of (%YS)/(curve %YS) at the applicable dose. That ratio provides a single
parameter that reflects both stress and dose. A ratio of 1.0 indicates a small (but non-zero) probability of failure.

![IASCC Trend Curve](image)

Figure 5. IASCC Trend Curve

The “failure” points from Figure 5 were sorted by IASCC Ratio and plotted on Weibull paper with non-failure points treated as “test suspensions”, in accordance with the standard methodology recommended in [4]. Finally, a straight line was fit to the data as shown in Figure 6. An excellent fit results, with a correlation coefficient of 0.9755, indicating that the data are well represented by a Weibull distribution. The Weibull distribution in Figure 6 was used in subsequent trials of the method to estimate the probability of cracking and number of cracked bolts that may be expected in any reactor vessel where the material and dose/age (in EPFY) were known. The predictions were then compared.

The very steep slope of the best fit line also indicates that the variance of the IASCC test points in this large sample is very small. That suggests that IASCC, as described by the IASCC ratio developed in this evaluation, is well behaved and readily predictable.
4. Trial Application

The method was first applied to 1999 vintage inspections from Ginna and Point Beach 2. Geometry and inspection results were obtained from EPRI TR-114779 (Feb. 2000).

- Bolt stresses and dose for these 2-loop plants were approximated based on generic 4-loop plant analysis. Those approximations assumed that bolt stress/temperature/dose distributions are comparable.
- The 4-loop plant bolt pattern (1088 bolts/48 baffle plates/8 former levels) was adapted to a 2-loop plant geometry (728 bolts/36 baffle plates/7 former levels).
- Further refinements to stress and dose were estimated based on the approximate location relative to the core.
- The adaptation of the 4-loop to 2-loop fluence and stress patterns provided the greatest source of uncertainty in analysis.

The results of that comparison (Figure 7) showed a reasonable agreement with inspection results. Note that Figure 7b includes inspection results from Point Beach-2, Ginna, and our example plant.
The method was also applied to different materials (e.g., solution annealed Type 304 or 347 stainless steel vs. 20% cold worked Type 316 stainless steel), as shown in Figure 8.

Figure 8 shows that the number of failures for the higher initial strength Type 316 (CW) would be less than that for Type 304 (or 347) in the solution annealed condition.
Figure 8. Material Sensitivity Comparison

This method was also used to make predictions of the spatial distribution of failures (also vs. time) for baffle-former bolts. As shown in Figure 9 (upper), the predictions from this methodology were in good agreement with observations of bolt failures and bolts with UT indications from Point Beach.
The predicted numbers of cracked bolts are considered reasonable, that is, they are on the order of the number of observed failures from industry OE, and they exceed the number of observed failures, as would be expected for a situation such as this where inspections rely exclusively or nearly exclusively on visual examinations that only reveal bolts that have cracked completed through. The numbers of predicted failures is large, considerably larger than the number of observed failures from the more recent example plant (circles on Figure 7b), but that large number is considered to be consistent with operation with a high leakage core design for a full 30 years (as computed). Most plants, including our example, changed their fuel design philosophy to incorporate a low leakage core relatively early in life.

All of the inputs for fluence, temperature, stresses, etc., including all predictions from the EPRI reports and other papers, assume that the baffle-former bolts in all eight octants would behave similarly to one another. Those assumptions would imply that the total number of cracked bolts would be eight times what is reported in Reference 2 and either plot a) or b) from Figure 7, which is obviously far greater than the number of failures that have been observed. That factor would be expected to decrease the number of predicted bolts with cracking, however, those differences would still not explain why all of the reported failures at some plants, including our example plant, have been in a single quadrant.
The results from the predictions and comparisons with the three examples discussed above suggest that those simplifying assumptions on loads and stresses are certainly reasonable, but that variability of loads and stresses around the circumference is also a powerful determinant of IASCC behavior. The good fits and very steep slope of the Weibull plot suggests that the variability in IASCC response to well-characterized loads (i.e., in the tests, loads were simple) and dose (also well characterized) is small. Further, the reasonable agreement between the predictions done per octant or per quadrant and service experience also shows that the simplifying assumptions appear to “fit” for one or two quadrants of a vessel, but do not necessarily agree well with loads, dose, material variability, etc. for a full circumference. The much broader distribution of actual failures around the circumference, especially its non-symmetric nature, suggests that the major unknowns are the actual loads and stresses for baffle-former bolts around a vessel. Variations in dose around the circumference, while expected to be predicted well by 4-fold or 8-fold symmetry, may also have an influence.

5. Summary and Conclusions
A methodology was developed for predicting the probability of IASCC of baffle-former bolts that incorporated effects of the material of construction, stresses, evolution of stress parameters during exposure, temperature, and accumulated dose. Based upon test data from the literature, a single IASCC susceptibility parameter was derived that incorporated the effects of stress and dose. Those data were well-fit by a Weibull distribution.

That methodology, based upon the Weibull prediction for IASCC failure was applied and compared to baffle-former bolt inspection results at Ginna at Point Beach 2 (circa 1999) and a third example. Bolt stress and dose vs. EFPY was approximated based upon the bolt location relative to a generic 4-loop plant analysis. There was significant uncertainty in this approximation, however, the results of the overall method compared reasonably well with experience.

Those results predict a steeply increasing cracking trend at 15~20 EFPY. The results also show the expected benefits of the use of higher strength cold worked bolts.

6. References
1. H.T. Tang, J.D. Gilreath, Aging Management of PWR Internals Components, 18th International Conference on Structural Mechanics in Reactor Technology (SMiRT 18) Beijing, China, August 7-12, 2005 SMiRT18-D06-1