# BWRVIP-140NP: BWR Vessel and Internals Project Fracture Toughness and Crack Growth Program on Irradiated Austenitic Stainless Steel

This report describes research sponsored by EPRI and the U.S. Department of Energy (Award No. DE-FC07-03ID14536).

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*Technical Report*

# BWRVIP-140NP: BWR Vessel and Internals Project
# Fracture Toughness and Crack Growth Program on Irradiated Austenitic Stainless Steel

**1008189NP**

Final Report, March 2005

Cosponsor
U.S. Department of Energy
Washington, DC

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This report describes research sponsored by EPRI, its participating BWRVIP members, and the U.S. Department of Energy under the Nuclear Energy Plant Optimization (NEPO) Program, Task 9, Year 2003.

Any opinions, findings, and conclusions or recommendations expressed in this material are those of the author(s) and do not necessarily reflect the views of the Department of Energy.

The report is a corporate document that should be cited in the literature in the following manner:

*BWRVIP-140NP: BWR Vessel and Internals Project, Fracture Toughness and Crack Growth Program on Irradiated Austenitic Stainless Steel,* EPRI, Palo Alto, CA, and U.S. Department of Energy, Washington, D.C.:  2005. 1008189NP.

# REPORT SUMMARY


As nuclear power plants age, the effects of neutron fluence become increasingly important to the reliability and performance of structural materials. It is known that austenitic materials, both stainless steels and nickel alloys, suffer significant fracture toughness reductions and increased susceptibility to intergranular stress corrosion cracking (IGSCC) at elevated fluence levels. This report describes a BWRVIP program plan to obtain additional information on the performance of these austenitic stainless steel alloys at the fluences and fluxes of interest for BWR and PWR internals.

### Background
The BWR Vessel and Internals Project has initiated a long-term strategy and project plan to obtain and develop additional crack growth rate and fracture toughness data for stainless steels exposed to high radiation fluences. Additional data are expected to lead to more accurate evaluations of component serviceability, avoiding unnecessary and costly repairs, replacements, or inspections.

### Objective
To describe the BWRVIP program plan called Fracture Toughness and Crack Growth Program on Irradiated Austenitic Stainless Steel and to summarize its technical basis.

### Approach
To prepare for this project, EPRI and BWRVIP conducted a workshop at Ponte Vedra Beach, Florida during February 19-21, 2003 (EPRI report 1007822). Attendees were invited to exchange relevant information on the effects of irradiation on austenitic materials in light water reactors and to produce recommendations for further work. EPRI reviewed the data, recommendations, and conclusions derived from the workshop and developed prioritized test matrices defining new data needs. Proposals were solicited, and selected proposals are the basis for the program described in this report.

### Results
The planned test matrix for fracture toughness testing includes 21 tests on 5 materials. Fluence levels as high as $8 \times 10^{21}$ n/cm$^2$ will be investigated. The planned test matrix for crack growth testing includes 13 tests, each on a different material. Again, fluence levels as high as $8 \times 10^{21}$ n/cm$^2$ are included in the crack growth test matrix. This inventory of highly irradiated materials apparently constitutes all samples that are available from BWR components removed from service.

## EPRI Perspective

This testing program makes good use of available highly irradiated materials removed from BWR service. The range of irradiation fluences represented here appears to be adequate to cover extended (60year) BWR service life of core structures. It may become necessary to expand the test matrix for weld metals and weld heat affected zone (HAZ) materials by conducting further irradiations in test reactors. This decision will depend in part on results of the present program and comparison of base metal properties to weld and HAZ properties.


## Keywords

Boiling water reactor
Irradiation damage
Stress corrosion cracking
Fracture toughness

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# 1 INTRODUCTION

As nuclear power plants age, the effects of neutron fluence become increasingly important to the reliability and performance of structural materials. The austenitic stainless steel alloys in BWR core structures experience significant fracture toughness reductions and increased susceptibility to stress corrosion cracking at elevated fluence levels. There are insufficient data on the performance of these alloys over the full range of radiation service exposures anticipated for BWR internals. The current database for irradiated materials contains certain gaps at neutron fluences that will become relevant to evaluation of serviceability of BWR core components. Without new data, these gaps would necessarily be spanned with conservative interpretations that might underestimate remaining service life. Thus, there is incentive to fill the gaps with supplemental data. The need for data is increasing as BWR plants pursue license renewal and 60-year plant operating lifetimes.

The serviceability of BWR core structures and components is verified periodically through a prescribed program of inspection and evaluation. Evaluations of current serviceability and projections of future serviceability rely on fracture toughness and crack growth rate data. Results of these evaluations support run/repair decisions.

To address the issue, the BWR Vessel and Internals Project has decided to develop a long-term strategy and project plan to both obtain and develop additional crack growth rate and fracture toughness data for stainless steels exposed to high radiation fluences. Additional data are expected to lead to more accurate evaluations of component serviceability, avoiding unnecessary and costly repairs, replacements or inspections. The BWRVIP program is entitled “Fracture Toughness and Crack Growth Program on Irradiated Austenitic Stainless Steel”. This report describes the program plan and summarizes its technical basis.

# 2
# BACKGROUND

To prepare for this project, EPRI conducted a workshop on irradiated stainless steel at Ponte Vedra Beach, Florida during February 19-21, 2003. Goals of the workshop were to identify existing relevant data; to identify plans of various organizations to develop new data; to identify needed data that is neither available nor currently planned; and to solicit suggestions for obtaining new data through cooperative programs or through collaboration or funding of new work.

Attendees were invited to exchange relevant information, to discuss and begin to resolve the major BWR (and PWR) issues related to effects of irradiation on austenitic materials in light water reactors, and to produce recommendations for further work. Conclusions from the workshop are contained in a summary document, "Fracture Toughness and Crack Growth Rates in Irradiated BWR Internals Components: Conclusions from an EPRI Workshop" published as BWRVIP-119 [Reference 9-1]. During and after the workshop, EPRI assembled a database on fracture toughness and crack growth rate test data relevant to BWR component assessments.

EPRI reviewed the data and conclusions derived from the workshop and developed prioritized test matrices defining new data needs. A Request for Proposals was issued in May 2004. In this first round of proposals, organizations were asked to provide testing services or test data responsive to identified needs and using irradiated materials already available to them. Three proposals were received. Two of these were joint proposals involving several organizations.

EPRI and BWRVIP appointed a technical review panel and convened a meeting on July 27, 2004 to form recommendations on the proposals. All three proposals offered both materials and testing services. No proposal offering test data was received. The review panel believed that all bidders were capable of performing satisfactory tests, and that it would be impractical to seek the transfer of irradiated materials from one bidder to another. Thus, the relative merits of each proposal depended in large part on the materials available to each bidder. The selected proposals and selected tests reflect the high value placed on high-fluence BWR materials removed from BWR service.

The review panel recommended specific tests selected from two of the three proposals. Revised proposals were requested to reflect the review panel's recommendations. The two revised proposals were received in September 2004. The proposed tests were closely responsive to the panel's recommendations. These revised proposals are the basis for the program described in this report. Workshop presentations found in Reference 9-1 are the basis for much of the technical discussion in this report.

The testing described here is scheduled to be completed by the end of 2008 (See Section 7). The review panel noted that, in typical plant evaluations, the more urgent need is for fracture toughness data. Most fracture toughness test results in this program will be available by the end of 2006. The review panel believed that satisfactory crack growth evaluations could be made, in the interim, using conservative assumptions.

The longer-term BWRVIP plan calls for new irradiations in test reactors, in the event that available irradiated materials do not meet all identified needs. New irradiations, if required, will be solicited through another round of proposals. It is not yet clear whether new irradiations will be necessary.

# 3
# DATA REVIEW

## 3.1 Data Applications

The irradiated materials of interest in BWR plant component evaluations are 300-series austenitic stainless steels, and especially Types 304, 304L, 316, 316L and 316NG. Data are needed for base materials, weldments, and associated weld filler metals. Data must be applicable to BWR core shrouds and top guides. The BWR core shroud is a large welded cylinder typically 1.5 - 2 inches thick (38 – 50 mm). The goal is a capability to evaluate the current and future load-carrying capacity and the load-deflection behavior of a core shroud containing surface cracks in highly irradiated locations.

Inspection indications in plant components may be evaluated using elastic-plastic fracture mechanics (EPFM), based on the so-called J-R curve and a J/T plot, provided that the material exhibits stable ductile crack growth prior to fracture. Alternatively, combinations of limit load analysis and linear elastic fracture mechanics analysis (LEFM) may be applied to evaluate serviceability and allowable flaw size. BWRVIP-100 (Reference 9-12) prescribes methodology appropriate for each of four ranges of fluence. Data from this project will lead to confirmation or refinement of BWRVIP-100 criteria for applications of EPFM and LEFM analyses.

When the maximum allowable flaw size has been determined, crack growth rate data are used to estimate remaining service life. BWRVIP-99 (Reference 9-15) prescribes crack growth rate curves for BWR environments and for a limited range of irradiation fluences. Data from this project are expected to confirm BWRVIP-99 methodology and extend it to higher fluences.

A summary of relevant and available data was prepared and provided to bidders on this project. Trends, gaps and data needs were identified. Current applications of the data were reviewed. Other planned or ongoing testing programs with possible relevance to BWRVIP needs were described. This discussion is summarized below.

## 3.2 Review of Fracture Toughness Test Data

Fracture toughness data relevant to this project are available from several sources, References 9-2 through 9-11. A database has been compiled and the existing data have been studied to identify needs for new data.

Fracture toughness testing laboratories usually report a period of stable crack extension with increasing load associated with these data. This is the J-R curve, J vs. Δa, from which $J_{IC}$ is derived (Figure 3-1). $J_{IC}$ is the value of the loading parameter J associated with the onset of ductile crack extension.

The test data of greatest value for evaluation of ductile materials is not $J_{IC}$ alone, but the J-R curve. Applications of nonlinear analysis to components, utilizing the J-R curve and the associated J/T curve, lead to a more realistic assessment of load-carrying capability and fracture margins. (The tearing modulus T is a measure of tearing resistance after $J_{IC}$ is exceeded and is proportional to the slope of the J-R curve after $J_{IC}$ is exceeded.) In practice, Δa (an increment of crack length) may be derived from measurements of crack opening displacement and load compliance.

BWRVIP-100 [Reference 9-12] is an evaluation of fracture toughness data from the above-listed sources. Data are presented as a family of J-R curves vs. fluence. Data analysis leads to an idealized, self-consistent family of curves suitable for application to component evaluations, and shown to be bounding in its representation of J-R and J/T from test results. Figure 3-1 illustrates the idealized family of J-R curves derived from test data [Carter and Gamble, Reference 9-1].

BWRVIP-35 (Reference 9-2) reports fracture toughness data for materials irradiated to a high BWR fluence, 8 x $10^{21}$ n/cm$^2$. Data were obtained from a single component, a welded control rod handle removed from BWR service. At this fluence relatively low $J_{IC}$ values were obtained from a base metal test at 25C, and from weld metal tests at all test temperatures. The material behavior associated with the lower values of $J_{IC}$ was described by the testing laboratory as ductile with little or no strain hardening capability. The amount of stable ductile crack extension was in some cases very small, but the test specimens were also small and could not support large values of stable crack extension. Surface examination revealed a different fracture mode associated with low toughness.

Data from the control rod handle are at least qualitatively consistent with high temperature (fast reactor) data showing toughness decreasing to a minimum beyond about 10 dpa (~7 x $10^{21}$ n/cm$^2$), with weld metal toughness decreasing perhaps more rapidly but approaching a similar threshold value [Tang, Xu and Fyfitch, Reference 9-1]. The fracture surface associated with this low toughness has been described as channel fracture. There is a need for more test data at BWR conditions to characterize the material dependence, and the possible temperature dependence, of the fracture toughness transition. The fast reactor data and the BWR data suggest that weld metal and base metal exhibit different transition behavior.

The weld heat-affected zone (HAZ) may exhibit different behavior than base metal or weld metal. Cracking in BWR core structures is found more often in the weld HAZ than in the weld metal. Weld sensitization may influence fracture toughness at intermediate fluences. While fracture behavior at very high fluences is thought to be independent of initial material condition, the BWR operates in the transition region where initial condition, and particularly sentization, could prove to be significant.

One laboratory reported fracture toughness data at an intermediate fluence as $K_{IC}$ rather than $J_{IC}$. $K_{IC}$ is the fracture toughness parameter associated with the onset of brittle fracture. No evidence of ductile crack extension was observed in these CT tests [Reference 9-3]. Apparently no other observations of brittle fracture have been reported in J-R fracture mechanics testing of similar alloys and radiation exposures. Because of these observations of non-ductile crack extension, BWRVIP-100 limits the application of elastic-plastic fracture mechanics to fluences below 3 x $10^{21}$ n/cm$^2$. It has been suggested that the difference between ductile and non-ductile behavior may be a result of crack orientation with respect to the plate rolling direction. For this reason the BWRVIP program plan includes examination of orientation effects on fracture toughness.

Several organizations are planning to conduct fracture toughness tests with probable application to BWRs. Of these, the Joint Owners Baffle Bolt (JOBB) test plan, to be conducted under the PWR Materials Reliability Program, is relatively well defined [Gilreath and Tang, Reference 9-1]. Some specimens will be irradiated in the Boris-60 test reactor; others will be cut from harvested PWR components. Testing will be conducted at 320°C. The PWR fluence range of interest is generally higher than the BWR core shroud end-of-life fluence. While the JOBB data will not be directly applicable to BWR components, results may offer insights that aid in interpretation of the BWR data.

Argonne National Laboratory (ANL) has been conducting a program over the past several years to develop fracture toughness test data at BWR conditions. The program involves irradiation of specimens in the Halden reactor and testing at the ANL facilities [Chopra, Gruber and Shack, Reference 9-1]. Results are reported in NUREG/CR-6826 [Reference 9-13].

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**Figure 3-1**
**Idealized J-R Curves as a Function of Neutron Fluence [Carter and Gamble, Reference 9-1]**

## 3.3 Review of Crack Growth Rate Data

Crack growth due to irradiation-assisted stress corrosion cracking (IASCC) is distinct from the intergranular stress corrosion cracking (IGSCC) that may occur in unirradiated, sensitized materials in the BWR environment. Both mechanisms are relevant to evaluations of BWR core shroud service life. In unirradiated materials, IGSCC is observed typically in weld-sensitized heat-affected zones or occasionally in weld metal. IASCC is not confined to sensitized material zones. Data are needed to characterize IASCC behavior in base materials and in associated weldments, in both normal and hydrogen water chemistry environments (NWC and HWC respectively).

Most of the available crack growth data for irradiated stainless steels are for irradiation fluence between 0.8 and 3 x $10^{21}$ n/cm$^2$. The crack growth rate as compared to that for unirradiated stainless steel [NUREG 0313 Rev 2, Reference 9-14] is elevated by a factor of 5 or more. A proposed disposition curve for irradiated stainless steels, based on these data, was presented in BWRVIP-99 [Reference 9-15]. Application of the elevated disposition curve is recommended when irradiation fluence exceeds 0.5 x $10^{21}$ n/cm$^2$ and is less than 3 x $10^{21}$ n/cm$^2$. A few data, discussed below, indicate that BWRVIP-99 may be unconservative and that HWC may become ineffective at some fluence well above 3 x $10^{21}$ n/cm$^2$.

The BWRVIP-99 disposition curve is based on a screened data set, from which invalid or questionable data have been excluded. Since the publication of BWRVIP-99, some data have been corrected and rescreened. Figure 3-2 shows a recent data update [Reference 9-16] and the BWRVIP-99 disposition curve for NWC data. The overall data assessment is unchanged.

Reference 9-16 shows that there are insufficient data to quantify the effect of fluence on crack growth rate. Similarly there are insufficient data to support quantification of effects of conductivity, temperature and ECP as was done for unirradiated data in BWRVIP-14.

In unirradiated materials, modification of the NWC environment by hydrogen addition suppresses or retards IGSCC in the BWR. This is commonly referred to as hydrogen water chemistry (HWC). For irradiated materials in the intermediate range of fluence, 0.8 to 3 x $10^{21}$ n/cm$^2$, data show that hydrogen water chemistry reduces the crack growth rate. BWRVIP-99 presents a proposed disposition curve for irradiated stainless steels in the HWC environment, based on these data. The HWC crack growth rate is lower than the NWC crack growth rate by a factor of 3.

Figure 3-3 shows a recent data update and the BWRVIP-99 disposition curve for HWC data. The BWRVIP-99 HWC disposition curve is quite conservative with respect to this updated data set. As for NWC, there are insufficient HWC data to quantify the effect of fluence on crack growth rate.

The slope of the disposition curve in Figures 3-2 and 3-3 is 2.5. This slope is based in part on data studies for unirradiated materials. The slope of 2.5 fits the NWC irradiated data very well (Figure 3-2), but there are insufficient HWC data (Figure 3-3) to confirm the slope independently of the other data. Earlier data correlations for unirradiated stainless steels (NUREG 0313 Revision 2 and BWRVIP-14) arrived at slopes close to 2.2. BWRVIP-99 cites evidence that the slope is higher than 3 in HWC environments, where the corrosion potential is low.

Data in Figures 3-2 and 3-3 are all within the fluence range of 0.8 to 3 x $10^{21}$ n/cm$^2$. A few data suggest that the mitigating effect of hydrogen water chemistry has largely vanished at a higher fluence [Jenssen et. al., Reference 9-1]. There appears to be a threshold fluence above which HWC becomes ineffective in slowing stress corrosion cracking. Crack growth data in the HWC environment are needed to characterize the transition.

Several organizations are planning or continuing to conduct crack growth tests with probable application to BWRs. The Halden Reactor Project (OECD) is conducting in-reactor crack growth testing on stainless steels in BWR and PWR operating environments [Karlsen, Reference 9-1]. The test matrix includes Types 304, 316NG, and 347 in BWR NWC and HWC environments, at fluences below $10^{22}$ n/cm$^2$. Some Halden data are shown in Figures 3-2 and 3-3.

Argonne National Laboratory is conducting a program to develop crack growth test data at temperatures and fluences of interest to the BWR. The program involves irradiation of compact tension specimens in the Halden reactor and testing at the ANL facilities. Data are reported to be consistent with results from the Halden crack growth studies. The effectiveness of oxygen reduction in mitigating IASCC was demonstrated, although one heat of material appears to be unresponsive [Chopra, Gruber and Shack, Reference 9-1].

Studsvik Nuclear, Ringhals AB and SKI have designed a program to develop crack growth rate data on irradiated Type 304L stainless steel in a BWR environment [Jenssen et. al., Reference 9-1]. The overall objective is to produce quantitative data on IASCC, comparing autoclave and in-pile results. The irradiated material is a control blade handle removed from service. Hydrogen water chemistry did not mitigate crack growth at this high fluence, 9 x $10^{21}$ n/cm$^2$.

A comprehensive Japanese national project is in progress to develop both IASCC and fracture toughness data and a methodology for its application to evaluation of BWR internal components. Materials include Types 304, 304L, 316L and 316NG (the latter used in core plates). Some data from this project are shown in Figures 3-2 and 3-3. Intended applications to the core shroud contemplate fluences up to 3 x $10^{21}$ n/cm$^2$ after 60 years of service. Neutron irradiation tests in the Japanese Material Test Reactor (JMTR) commenced in 2001 and are scheduled to continue into early 2006. Post-irradiation examinations will be completed in 2008 [Murakami, Reference 9-1].

The international Cooperative IASCC Research (CIR) Program is designed to quantify critical parameters that influence IASCC and to develop predictive methodology and countermeasures. The first phase has been completed (CIR I, 1995-2000). A second phase (CIR II, 2000-2004) has conducted irradiations in a fast reactor and plans tests in BWR environments [Pathania and Gott, Reference 9-1].

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**Figure 3-2**
**NWC Proposed Disposition Curve with Irradiated NWC Data Irradiation Fluence 0.8 to 3 x $10^{21}$ n/$cm^2$ (Reference 9-16)**

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**Figure 3-3**
**HWC Proposed Disposition Curve with Irradiated HWC Data Irradiation Fluence 0.8 to 3 x $10^{21}$ n/$cm^2$ (Reference 9-16)**

## 3.4 Data Needs.

Based on the review of available data, the following table was included in the Request for Proposals as a solicitation of fracture toughness data.

**Table 3-1**
**Proposal Solicitation of Fracture Toughness Data**

| Objectives | Priority |
|---|---|
| • Define the transition to lower fracture toughness at fluences between 2 or 3 and 10 x $10^{21}$ n/cm$^2$.<br>• Examine material and orientation dependence at BWR operating conditions.<br>• Identify possible transitions from J-R to $K_{IC}$ fracture behavior.<br>• Identify possible transitions associated with lower temperatures. | A. Plate, longitudinal. Type 304 or similar. Three fluence levels; 288°C. |
| | B. Plate, transverse. Same material and heat as A above. Three fluence levels; 288°C. |
| | C. Weld heat-affected zone. Three fluence levels; 288°C. |
| | D. Plate of a different type or heat than A and B, in the least tough orientation indicated by those results. Three fluence levels; 288°C. |
| | E. Weld metal. (Flux weld) Three fluence levels; 288°C. |
| | F. Selected repeats of A through E at test temperatures under 100°C. Three tests. |

Similarly, the following table was included in the Request for Proposals as a solicitation of crack growth rate data.

**Table 3-2**
**Proposal Solicitation of Crack Growth Data**

| Objectives | Priority |
|---|---|
| • Quantify the effectiveness of HWC in mitigating NWC crack growth rates at fluences between 1 and 3 x $10^{21}$ n/cm$^2$.<br>• Define the trend toward convergence of HWC and NWC behavior above 3 x $10^{21}$ n/cm$^2$.<br>• Provide a basis for flaw disposition criteria in this range of radiation exposures. | A. (1) Select a material and heat number which has been exposed to fluences between 1 and 3 x $10^{21}$ n/cm$^2$, and to higher fluences. Measure crack growth rates in the lower range of fluence for three or more levels of applied K. Alternate between HWC and NWC water chemistry environments at each K level. [6 or more data points]<br>(2) Test this same heat at one or two higher fluences between 5 and 10 x $10^{21}$ n/cm$^2$. Obtain data for three or more levels of applied K at each fluence. Alternate between HWC and NWC water chemistry environments at each K level. [12 or more data points] |
| | B. Select at least two other materials irradiated to fluences between 5 and 10 x $10^{21}$ n/cm$^2$. Obtain data for three or more levels of applied K at each fluence. Alternate between HWC and NWC water chemistry environments at each stress level. [12 or more data points] |
| • Quantify the effectiveness of HWC in mitigating NWC crack growth rates at fluences below $10^{21}$ n/cm$^2$.<br>• Provide a basis for flaw disposition criteria for exposures of 0.5 to 1 x $10^{21}$ n/cm$^2$. | A. Select a material irradiated to fluence in the range of 0.5 to 1 x $10^{21}$ n/cm$^2$, for which unirradiated crack growth rate data are available or can be obtained. Measure crack growth rates at 3 to 5 levels of K. Alternate between HWC and NWC water chemistry environments at each stress level. [approximately 10 unirradiated and 10 irradiated data points] |
| | B. Identify materials as described below and exposed to fluences in the range of 0.5 to 1 x $10^{21}$ n/cm$^2$. Measure crack growth rates in each at 3 to 5 levels of K. Alternate between HWC and NWC water chemistry environments at each stress level.<br>a. Weld HAZ [6 to 10 or more data points]<br>b. Weld metal [6 to 10 or more data points]<br>c. Other materials with radiation exposure in this range, for which unirradiated crack growth behavior has been characterized. [6 to 10 or more data points] |

# 4

# BWRVIP PROGRAM PLAN

Responses to the RFP were reviewed and revised as discussed above, leading to planned tests described here. Selected tests are to be performed by Studsvik Nuclear (Sweden), working with Nippon Nuclear Fuel Development (NFD) (Japan); and by General Electric's Vallecitos Nuclear Center laboratory, supported by the GE Research Center, by Battelle, Northwest Laboratories, and by the University of Michigan.

## 4.1 Program Plan: Fracture Toughness Testing

All of the fracture toughness tests are to be performed by Studsvik and NFD, in the respective laboratory where materials are available. Fracture toughness tests and materials are described by the table below:

**Table 4-1**
**Materials for Studsvik/NFD Fracture Toughness Testing**

| Mat'l | Source | Fluence dpa | Fluence (estimated) $10^{21}$ n/cm$^2$ | Orientation | No. of tests | CT specimen size, mm |
|---|---|---|---|---|---|---|
| 304 | Top Guide /Forsmark | 2 | 1.3 | Longitudinal | 3 | 1/2T, B=8 |
| 304 | Top Guide /Forsmark | 2 | 1.3 | Transverse | 3 | 1/2T, B=8 |
| 316L | CR/Oskarshamn | 5-7 | 3.3-4.7 | Longitudinal | 3 | W=16, B=8 |
| 316L | CR/Oskarshamn | 5-7 | 3.3-4.7 | Transverse | 3 | W=16, B=8 |
| 316L | CR/Oskarshamn | 5-7 | 3.3-4.7 | Weld | 1* | W=16, B=8 |
| 304L | CR/Halden | 12 | 8.0 | Longitudinal | 2 | W=16, B=8 |
| 304L | CR/Halden | 12 | 8.0 | Transverse | 2 | W=16, B=8 |
| 304 | CR/NFD | 8 | 5.3 | Longitudinal | 2 | 1/2T, B=8 |
| 304 | CR/NFD | 8 | 5.3 | Transverse | 2 | 1/2T, B=8 |

* Two if a smaller specimen size is used

Table notes:

In every case the source of material is a top guide or a control rod (CR) removed from an operating BWR. Materials at the Halden test reactor were removed from Swedish BWRs. The NFD materials were removed from TEPCO BWRs.

Fluence values given in the table are to be refined after final selection of specimen locations. Results from neutron fluence calculations are available that can be used to assess fluence at the locations where the specimens will be machined. For the Forsmark top guide material, fairly comprehensive calculations have been performed. Studsvik and NFD present fluence in terms of dpa, or displacements per atom. The fluence equivalents in terms of neutrons per cm2 are rough estimates produced by EPRI.

All testing is to be performed with CT (compact tension) specimens for which B is the thickness and W is the distance from the load line to the back face. The CT specimens will use side grooves with a depth of 5% of thickness. The number of similar specimens to be tested is three or less, depending on the amount of material available. The intention is to acquire as much information as practical from the available materials through duplicate or triplicate testing.

## 4.2 Program Plan: Crack Growth Rate Testing

Crack growth tests are to be performed by Studsvik and by GE-VNC, as described by the two tables below. See the preceding notes in 4.1 for an explanation of information on the Studsvik/NFD materials. NFD materials will be shipped to the Studsvik laboratory for crack growth testing.

**Table 4-2**
**Materials for Studsvik Crack Growth Testing**

| Phase | Material | Source | Fluence dpa | Fluence (estimated) $10^{21}$ n/cm$^2$ | CT specimen size, mm | Test time, months |
|---|---|---|---|---|---|---|
| 1 | 304L | CR /Halden a | 3.5 | 2.3 | W=16, B=8 | 5 |
| | 304L | CR /CirII/Barseb | ~5 | ~3 | W=16, B=8 | 5 |
| | 304L | CR /CirII/Barseb | ~10 | ~7 | W=16, B=8 | 3 |
| | 316L | CR /Oskarshamn | 5-7 | 3.3-4.7 | W=16, B=8 | 4 |
| | 304L | CR /Halden b | 12 | 8 | W=16, B=8 | 2 |
| 2 | 304 HAZ | Core Shroud /NFD | 0.8 | 0.5 | 1/2T, B=12.7 | 6 |
| | 304 weld | Core Shroud /NFD | 0.8 | 0.5 | 1/2T, B=12.7 | 6 |
| 3 | 316 | Top Guide /NFD | 0.7 or 1.4 | 0.5 or 0.9 | 1/2T, B=9.1 | 6 |

**Table 4-3**
**Materials for GE-VNC Crack Growth Testing**

| Phase | Material | Source | Fluence $10^{21}$ n/cm$^2$ | CT specimen size, mm* | Test time, months |
|---|---|---|---|---|---|
| 1 | 316NG | CR /TaiPower | 1 | B=7.5 | ~6 |
| | 316NG | CR /TaiPower | 3 | B=7.5 | ~6 |
| | 316NG | CR /TaiPower | 4.3 | B=7.5 | ~6 |
| 2 | 304 | CR /Millstone | 3 | B=7.5 | ~6 |
| | 304NG | CR /TaiPower | 4.3 | B=7.5 | ~6 |

*Standard 0.5T x 0.3-inch thick, 0.24 thick at side grooves, W = 1 inch or 25mm.

Crack growth tests have been prioritized and grouped into phases for costing and contracting purposes. It is anticipated that the second phase of testing will be initiated within a year after the first phase, and the third phase within a year after the second. The tests will be conducted in BWR normal water chemistry and hydrogen water chemistry environments at several values of stress intensity (K) to assess the effect of electrochemical potential (ECP) and K on crack growth.

Supplemental work described in Section 6.2 will be performed at Battelle, Pacific Northwest Division, to characterize materials to be tested by GE-VNC.

In view of responses to the RFP, the chosen approach for the crack growth portion of the program plan is to test virtually all of the available and suitable highly irradiated materials.

An alternative approach would be to identify the variables affecting irradiated material performance and to address each variable systematically in the test matrix. Mathematical models of crack growth rate as a function of salient variables could then in principle be constructed from test results. This approach was taken in the analysis of crack growth data for unirradiated materials in BWRVIP-14 [Reference 9-15]. Many variables are thought to influence irradiated material performance, and the limited number of available test materials is not expected to support comprehensive model development. The test data produced in this program are expected to characterize crack growth rate using a conservative bounding approach.

# 5
# TESTING REQUIREMENTS

## 5.1 Requirements for Fracture Toughness Testing

The Florida workshop developed recommendations for optimum fracture toughness testing and data recording [Gamble, Reference 9-1]. High quality data will reflect at least the following:

- Specimen dimensions
- Specimen configuration (side grooves)
- Load vs. displacement (tabular form)
- J vs. $\Delta a$ (tabular form)
- Specification used for J tests
- Specimen orientation
- Yield + Ultimate, RA (tabular form)
- Material specification (chemical composition)
- Heat treatment
- Weld process / heat treatment
- Fluence (0.1MeV), (>1 MeV), dpa
- Dose
- Extent of data collection (recorded data is not to be limited to valid data only)
- Offset line (if different from standard)
- Fracture surface description

The RFP calls for preparation of test specifications. Studsvik and NFD propose to develop a specification for fracture toughness testing and submit it to EPRI for review before testing commences. A pretest analysis will be provided to define the maximum J and $\Delta a$ values (per Section A8.3 of ASTM E 1820) and the expected test range for the specimen sizes and materials proposed for testing.

Studsvik and NFD propose to conduct fracture toughness testing in accordance with ASTM E1820, using the resistance curve procedure (single specimen technique). Actual crack extension will be determined using post-test photographs of the fracture surface, described in 6.1. Tensile tests will be conducted as required to obtain properties for the resistance curve procedure. The test specification will take into account the above recommendations from the workshop. In particular, the tests will be run to acquire as much data as possible per specimen for generating J-R curves, even if there are data outside the region of J-controlled crack extension. All data generated for the J-R curves will be evaluated after testing to assess if these data are appropriate or not.

Fracture toughness testing presents a specimen size validity issue similar to the K-validity issue discussed in 5.2.4 below in relation to crack growth testing. The fracture toughness issue is less restrictive, because validity limits are likely to be exceeded only when fracture toughness is high. These high but inaccurate data are not limiting in component evaluations. When fracture toughness is low, the test is likely to be valid. This can be confirmed case-by-case when test results are in hand.

## 5.2 Requirements for Crack Growth Rate Testing

The Florida workshop developed a list of key factors for consideration in crack growth rate testing and data recording [Hickling, Reference 9-1]. High quality data will reflect at least the following:

- Material within specifications, including composition/condition/heat treatment
- Mechanical strength properties
- ASTM specimen size criteria and degree of plastic constraint
- Pre-cracking technique (including straightness criteria, plastic zone size, crack morphology)
- Special requirements for testing welds (such as pre-crack location, residual stresses/strains)
- Environment (chemistry, temperature, ECP, flow rate at specimen, neutron/gamma flux)
- Loop configuration (once-through, refreshed, static autoclave)
- Water chemistry confirmation by analysis (Cl, SO4, O2, Cr, conductivity, etc.)
- Active constant or cyclic loading versus constant displacement loading (for example, using wedge)
- On-line measurement of crack length versus time during test (including precision)
- Actual crack length confirmed by destructive examination (assessment method/mapping)
- Appropriateness of crack characteristics (fraction SCC along crack front, uniformity, adequate SCC increment, transgranular portions within IGSCC fracture surface, etc.)
- Possible effects of changes in loading or chemistry conditions during a test (including heat up and cool down)
- Calculation and reporting of K or $\Delta K$ values
- Reporting of raw a vs. t data and derivation of da/dt values
- Reproducibility of data under nominally identical test conditions

The RFP calls for preparation of test specifications. Studsvik proposes to develop specifications for crack growth testing and submit them to EPRI for review before testing commences. The GE proposal describes test specifications and procedures in some detail. The following discussion is based on information in the proposals. Issues are identified which should be reviewed with GE and Studsvik before testing begins.

### *5.2.1 Water Environment*

In discussion of earlier data, the terms NWC and HWC have been applied broadly to water chemistry environments of high and low corrosion potential which do not necessarily meet operating plant standards for normal and hydrogen water chemistry respectively. A standard for current testing is proposed by GE. The NWC condition is defined as a dissolved oxygen concentration of 2000 ppb and a corrosion potential of +160mV (SHE). The HWC condition is defined as a hydrogen concentration of 100 ppb, and a corrosion potential of -520mV (SHE). An outlet conductivity not exceeding 0.1µS/cm, and a temperature of 288°C, are prescribed for both NWC and HWC testing.

Studsvik defines HWC as 100 ppb of hydrogen, and NWC as 2000 ppb of oxygen, consistent with the GE definition but less complete. The Studsvik proposal demonstrates capability to monitor potential and conductivity but defers specification of criteria for these variables.

Specifications should describe means for monitoring and control of dissolved hydrogen and oxygen; monitoring of electrochemical corrosion potential; monitoring and control of water conductivity; and identification of species of impurities contributing to conductivity. The flow rate should be such that water chemistry conditions at the test specimen can be unambiguously determined from monitored parameters. Studsvik describes a capability to monitor conductivity and to analyze water chemistry and identify species contributing to conductivity using grab samples. GE proposes to monitor conductivity.

### *5.2.2 Precracking*

Both GE and Studsvik propose to precrack CT specimens in a simulated BWR normal water chemistry environment by transitioning slowly from fatigue loading to nearly constant load. Studsvik prescribes a detailed sequence of steps, used successfully in the past, to effect the transition from transgranular to intergranular cracking. Studsvik adheres to the prescribed sequence to avoid uneven biasing of subsequent test results. GE is less prescriptive but highlights use of reversing DC potential probes to monitor crack growth during the precracking phase and during subsequent testing. The two procedures appear to be similar and could be made identical, but it may be unwise to modify the details of successful laboratory practice for the sake of standardization.

### *5.2.3 Testing sequence*

Studsvik and GE propose similar but not identical testing sequences for crack growth measurements. The two are compared in the table below (11 MPa√m is very nearly equal to 10 ksi√in.).

**Table 5-1**
**Testing Sequences for Crack Growth Measurements**

| Test condition | Studsvik | | GE-VNC | |
|---|---|---|---|---|
| | Stress Intensity K | Crack Increment, mm | Stress Intensity K | Crack Increment, mm |
| IG transitioning | 11 MPa√m | 1.2 | 10 ksi√in | 1 to 3 |
| NWC | 11 MPa√m | 0.4 | 10 ksi√in | 0.5 |
| HWC | 11 MPa√m | 0.2 | 10 ksi√in | 0.1 |
| NWC | 11 MPa√m | 0.1 | | |
| NWC | Cyclic, ramp up | 0.2 | | |
| NWC | 14 MPa√m | 0.4 | 16 ksi√in | 0.5 |
| HWC | 14 MPa√m | 0.2 | 16 ksi√in | 0.1 |
| NWC | 14 MPa√m | 0.1 | | |
| NWC | Cyclic, ramp up | 0.2 | | |
| NWC | 18 MPa√m | 0.4 | 22 ksi√in | 0.5 |
| HWC | 18 MPa√m | 0.2 | 22 ksi√in | 0.1 |
| NWC | 18 MPa√m | 0.1 | | |
| HWC | | | FT test | < 2 |

Note in the table above that GE-VNC proposes to test at a higher value of K than Studsvik (22 vs about 16.4 ksi√in). Like Studsvik, GE proposes to use slow cycling, not specified here, to effect the transitions to higher K-levels. Cumulative crack growth is held to a smaller value in the proposed GE test sequence. Both GE and Studsvik caution that additional test specimens may be needed to acquire valid data at the highest stress intensity. Potential advantages of a second test specimen (of each material) are (a) investigation of a greater range of K, (b) better accuracy deriving from a larger increment of crack growth, and (c) larger margins against an open question of K-validity discussed below.

### *5.2.4 K-Validity*

The validity of test data has become a significant issue in the interpretation of stress corrosion crack growth data in irradiated materials. In any test based on linear elastic fracture mechanics (LEFM), when crack length or applied stress intensity become too large, the extent of plastic deformation in the test specimen becomes excessive and the theoretical underpinnings of LEFM test data interpretation and application become invalid. In short, the test is no longer similar to the application.

Specimen thickness and side grooves also affect the extent of plastic deformation. Consensus criteria have been developed over the years by ASTM to address this issue of similitude and K-validity for various types of fatigue and fracture and corrosion-assisted cracking tests.

The application of ASTM validity criteria to SCC testing of irradiated materials presents unique issues. Highly irradiated materials with elevated yield stress do not exhibit the normal strain hardening characteristics of austenitic stainless steels. The material may even exhibit strain softening at higher fluences, meaning that the yield stress decreases when initial yielding occurs. The extent of plastic deformation in irradiated materials can be larger than would be predicted for other materials of similar yield strength. Consensus criteria defined for other kinds of testing do not appear to be applicable to SCC growth rate in irradiated stainless steels. The issue is important, in that excessively high crack growth rates have been reported where K-size criteria are clearly violated in crack growth testing of irradiated stainless steels.

This issue was reviewed by Andresen [References 9-18, 9-15] and has been discussed also by Jenssen, et al [Reference 9-19] and by Chopra, Gruber and Shack [Reference 9-13]. An emerging consensus agrees that an effective yield stress lower than the measured post-irradiation yield stress is appropriate for application of ASTM validity criteria to irradiated stainless steels. It is not yet clear how much reduction is needed or how the effective yield stress should be quantified. An effective yield stress equal to the average of irradiated and unirradiated yield stresses has been recommended and applied with caveats. Andresen advises that this (average value) may be non-conservative or borderline for highly irradiated materials, and suggests use of a lower value. Jenssen et al found a better fit to data using an effective yield stress lower than the average of irradiated and unirradiated yield stresses. Further study has been recommended.

The preferred form of a more conservative validity criterion is that suggested by Andresen: the effective yield stress is taken as the unirradiated yield stress, plus one-third (or some fraction) of the difference between irradiated and unirradiated yield stresses. The form used by Jenssen et al does not ensure that the effective yield stress will be higher than the unirradiated yield stress.

A second issue discussed by Andresen is in the fact that the SCC crack advance mechanism is not fracture alone, as standard LEFM theory supposes. Cracking is primarily due to very localized corrosion resulting from crack tip deformation. The crack advance mechanism may become intermittent or unsteady at low K, leading occasionally to crack arrest and contributing to the observed lack of reproducibility in the data. It is not clear that this is a K-similitude issue but it is another complicating factor in the interpretation of data.

The lack of consensus criteria for K-validity presents risks to this testing program. The validity of new and existing data may be called into question. A conservative approach probably entails added costs for additional test specimens. Data at higher values of K, where applications may require it, could be out of reach when test specimens are small and conservative K-validity criteria are applied.

In this context the word conservative is used by experimentalists to mean assured validity of test data. But invalid crack growth rate data obtained at high K are apparently always on the high side of valid data, meaning that invalid test data are overly conservative in application to structural evaluations. What is at issue is not the safety of reactor structures, it is the quality and ultimate utility of expensive test data. The challenge is to extract as much value as possible from a limited supply of materials and funding.

BWRVIP program management can take actions to add value to crack growth test data.

- Promote and support the timely development of consensus standards appropriate for crack growth rate testing in this class of materials. Related materials would include cold-worked and precipitation hardened alloys that exhibit little or no strain hardening [Reference 9-18].
- Ensure that appropriate and consistent alternative validity criteria are actually applied in test specifications and when data are reported. Studsvik states that the test at K=18 MPa√m will likely be close to the alternative (average) validity criteria discussed above. GE proposes to test at a 34% higher value of K, but the GE proposal does not make clear what validity criteria will be applied. Studsvik uses a smaller specimen than GE for most crack growth rate tests (W=16mm vs 25mm; see 4.2 above). Both GE and Studsvik note that higher K can be achieved with an additional test specimen, by testing at high K while the crack is shorter.
- Ensure that the highest stress intensity levels to be investigated in the testing program are reasonably consistent with the highest stress intensity levels encountered in plant applications.
- Prepare to supplement the testing program with additional data from additional crack growth specimens if necessary.

### 5.2.5 Crack growth monitoring

GE will instrument the CT specimens with platinum wires for reversing direct current potential drop (DCPD) data acquisition to monitor crack length. Duplicate probes will be installed on each test specimen to provide redundancy. Crack growth will be estimated from the most stable set of probes and subsequently adjusted to the actual final crack length as determined by post-test examinations.

Studsvik will also use a DCPD system to monitor crack growth on-line and conduct post-test fracture surface examination to adjust the DCPD measurements.

### 5.2.6 Corrosion Potential

GE offers to provide a description of the type and design of the reference electrode, its location relative to the specimen, example data, and its long-term reliability and stability. Studsvik will monitor corrosion potentials on the specimen using a copper/copper oxide membrane electrode. Studsvik advises that, because the specimen is not insulated from the autoclave at the loading pins, the measured corrosion potential will be that of the specimen and the Nimonic 90 clevises. Studsvik provides data demonstrating capability to control this corrosion potential in the required range.

# 6 POST-TEST EXAMINATIONS

## 6.1 Examination of Fracture Toughness Test Specimens

Fractographic examinations will be conducted at both Studsvik and NFD to determine the shape of the fatigue precrack, the actual crack length and the fracture mode, and to document all features of interest. TEM (transmitting electron microscopy) observations and grain boundary analyses will be performed at NFD to obtain microstructure and grain boundary chemistry data for all tested materials. The technical specification of these post-test examinations is found in the Studsvik-NFD proposal and reproduced in the Appendix.. Portions of fractured specimens will be shipped from Studsvik to NFD for these examinations.

## 6.2 Examination of Crack Growth Rate Specimens

Examinations of Studsvik crack growth rate specimens will be essentially the same as post-test examinations described in Section 6.1 for fracture toughness test specimens.

Post-test examination of the GE crack growth test specimens will be conducted on samples at the University of Michigan using the scanning electron microscope (SEM). Initial examinations will verify and quantify the fatigue precrack region, the region of transgranular cracking, and the transition to intergranular cracking. Fracture surfaces will be examined to verify the straightness of the crack front and to verify that the crack growth mode was indeed intergranular. Fracture surfaces will be examined to validate the DC potential drop results for each step of the testing sequence. High resolution methods will quantify the amount of intergranular cracking and identify the specific fracture planes participating in the fracture process.

Additional characterization tasks will be performed at Battelle, Pacific Northwest Division, on samples of the GE crack growth specimens. Microstructural characterizations, using SEM and TEM, will provide information on mechanisms leading to changes in fracture toughness and on mechanical aspects of IASCC. Microchemical analysis will provide information on radiation-induced grain boundary segregation of elements. Gas analysis and retrospective dosimetry will reduce uncertainty on irradiation exposures and gas generation, and support extrapolation to higher fluence of probable degradation.

# 7
# PROGRAM MANAGEMENT

## 7.1 Schedule

For fracture toughness testing, 50% of the Studsvik tests are scheduled to be completed by the end of 2005. The other 50% of Studsvik tests are scheduled to be completed by the end of 2006. The NFD tests are scheduled to be completed by the end of 2006. Post-test examinations and the final report are to be complete by 2007.

The schedule for crack growth testing is presented below, based on timely commitments for each phase of the work.

**Table 7-1**
**Studsvik/NFD (Final Report February 2009)**

| Phase | Test | Date | Material | Fluence $x10^{21}n/cm^2$ |
|---|---|---|---|---|
| Phase 1 | Test #1 | 03/2006 | 304L, 3.5dpa | ~2.3 |
| | Test #2 | 08/2006 | 304L, ~5dpa | ~3 |
| | Test #3 | 12/2006 | 304L, ~10dpa | ~7 |
| | Test #4 | 04/2007 | 316L, 5-7dpa | ~3.3-4.7 |
| | Test #5 | 07/2007 | 304L, 12dpa | ~8 |
| Phase 2 | Test #6 | 01/2008 | 304 HAZ, 0.8dpa | ~0.5 |
| | Test #7 | 02/2008 | 304 weld, 0.8dpa | ~0.5 |
| Phase 3 | Test #8 | 07/2008 | 316, 0.7 or 1.4dpa | ~0.5 or 0.9 |

**Table 7-2**
**GENE (Final Report January 2008)**

| Phase | Test | Date | Material | Fluence $x10^{21} n/cm^2$ |
|---|---|---|---|---|
| Phase 1 | Test #1 | 12/2005 | 316NG | 1 |
| | Test #2 | 06/2006 | 316NG | 3 |
| | Test #3 | 12/2006 | 316NG | 4.3 |
| Phase 2 | Test #4 | 06/2007 | 304 | 3 |
| | Test #5 | 12/2007 | 304NG | 4.3* |

*Optionally 4.9 at extra retrieval cost

## 7.2 Reporting

Both GE and Studsvik/NFD will provide an initial report documenting available information on the materials used in the testing program, including irradiation histories. Studsvik will provide a test specification for EPRI review, and GE will report on CGR testing procedures. All contractors will provide monthly and quarterly progress reports, annual interim technical reports, and final reports.

## 7.3 Technical Steering Committee

EPRI and the BWR Vessel and Internals Project (BWRVIP) will form a project steering committee having representation from testing organizations, end users, technical experts, and perhaps others in the nuclear industry. The steering committee will identify and seek resolution of issues which could impact the value of the product. Issues might involve technical adequacy and consistency, regulatory acceptance, or applicability of results to BWR plants. The K-validity issue discussed in 5.2.4 is an example.

# 8
# CONCLUSIONS

- A testing program has been defined to address needs for data on irradiated materials by making best use of available materials.
- Data needs were defined following a workshop, well attended and well documented, to solicit relevant views and information.
- Available materials were identified through a request for proposals.
- Testing objectives were defined and prioritized in the RFP.
- Test matrices have been defined, based on responses to the RFP.
- Fracture toughness tests are planned at fluences as high as $8 \times 10^{21}$ n/cm$^2$ in base material. A very limited amount of irradiated weld metal is available for fracture toughness testing, at a fluence near $4 \times 10^{21}$ n/cm$^2$.
- Crack growth rate tests are planned at fluences as high as $8 \times 10^{21}$ n/cm$^2$ in base material. Weld and HAZ materials are available for crack growth rate testing only at a low fluence, $0.5 \times 10^{21}$ n/cm$^2$.
- Crack growth data is difficult to obtain at high values of "K" (crack-tip stress intensity) without exceeding validity criteria. Recommendations are offered to address this issue.

# 9
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