**UNITED STATES**
**NUCLEAR REGULATORY COMMISSION**

**ATOMIC SAFETY AND LICENSING BOARD**

| | |
|---|---|
| In re: | Docket Nos. 50-247-LR; 50-286-LR |
| License Renewal Application Submitted by | ASLBP No. 07-858-03-LR-BD01 |
| Entergy Nuclear Indian Point 2, LLC,<br>Entergy Nuclear Indian Point 3, LLC, and<br>Entergy Nuclear Operations, Inc. | DPR-26, DPR-64<br><br>June 18, 2015 |

**DECLARATION BRIAN LUSIGNAN**

Pursuant to 28 U.S.C. § 1746, Brian Lusignan hereby declares as follows:

1. I serve as an Assistant Attorney General for the State of New York, counsel for petitioner-intervenor State of New York in this proceeding. I submit this declaration and accompanying attachments a part of the State of New York's reply to the June 4, 2015 Joint Brief of Entergy and Westinghouse Regarding Proprietary Documents (Joint Industry Brief).

2. Attached to this declaration as Attachment 1 is a true and correct copy of the publicly available, full-text copyrighted document authored by Christopher Kupper and Mark Gray, entitled *License Renewal Environmental Fatigue Screening Application*, Paper No. PVP2014-29093, Proceedings of the ASME 2014 Pressure Vessels and Piping Conference, Anaheim, California (July 20-24, 2014). This paper is publicly available at http://proceedings.asmedigitalcollection.asme.org/proceeding.aspx?articleID=1937745.

3. Attached to this declaration as Attachment 2 is a true and correct copy of EPRI Report 1024995 entitled *EAF Screening: Process and Technical Basis for Identifying EAF Limiting Locations*. It is publicly available at http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001024995.

4. Attached to this declaration as Attachment 3 is a true and correct copy of the

September 19, 2013 NRC Staff's License Renewal Team Inspection Report 05000247/2013010 (ML13263A020). It is publicly available at http://pbadupws.nrc.gov/docs/ML1326/ML13263A020.pdf. NRC Staff disclosed this document in September 2013 as NRC document 56-0012. Entergy disclosed this document in October 2013 as Entergy document 1652.

5. On behalf of the State of New York, I attended the February 19, 2015 public meeting convened by NRC Staff and industry representatives at Three White Flint North at the NRC Headquarters in Rockville, Maryland, to discuss reactor pressure vessel issues. To my knowledge, no transcript was made of the meeting. The meeting was attended by many industry representatives and NRC Staff members. Individuals attempted to listen to the meeting via telephone, and at certain times those individuals also attempted to ask questions or make comments. A variety of technical issues appeared to substantially impair the public's ability to participate by telephone.

6. During the February 19, 2015, public meeting, NRC Staff presented a PowerPoint presentation entitled "Part II: Assessment of Impact on Plants Using BTP 5-3 to Estimate $RT_{NDT(u)}$." This slideshow is publicly available on ADAMS at Accession No. ML15061A075. The slideshow evaluated the potential impact of BTP 5-3 non-conservatisms on 19 pressurized water reactors (PWRs), and in particular on the Pressurized Thermal Shock (PTS) screening criteria. Slide 2 indicated that adjustment of the PTS evaluation to account for BTP 5-3 non-conservatism "may cause one of these plants to exceed the PTS screen criteria during the license renewal (LR) period." In response to this presentation, a person who had been listening over the telephone asked NRC Staff to identify which plant would exceed the PTS screening limit. NRC Staff refused to identify that plant, although a staff member did state that the plant was not in Michigan. A person on the telephone also asked NRC Staff to identify the 19 PWRs that had

been evaluated as part of this study. Again, NRC Staff refused to identify the plants that had been studied.

7. Attached to this declaration as Attachment 4 is a summary of the February 19, 2015 public meeting in Rockville, Maryland to discuss reactor pressure vessel issues, including a list of attendees (April 6, 2015) (ML15096A128).

8. Attached to this declaration as Attachment 5 is an e-mail I received from Hearingdocket@nrc.gov on June 4, 2015, notifying me that Entergy and Westinghouse had filed their Joint Brief using the non-public Electronic Information Exchange (EIE).

9. Attached to this declaration as Attachment 6 is an e-mail I sent to attorneys for Westinghouse, Entergy and NRC Staff on June 15, 2015 regarding the apparent failure of Westinghouse and Entergy to properly file the Joint Brief publicly.

10. I declare under penalty of perjury that the foregoing is true and correct.

Executed June 18, 2015

***Signed (electronically) by***

Brian Lusignan
Assistant Attorney General
Office of the Attorney General
of the State of New York
The Capitol
Albany, New York 12224
(518) 776-2399
Brian.Lusignan@ag.ny.gov

**Attachment 1**

**License Renewal Environmental Fatigue Screening Application**

**Authored by Christopher Kupper and Mark Gray**

**Paper No. PVP2014-29093**

**Proceedings of the ASME 2014 Pressure Vessels and Piping Conference**

**Anaheim, California (July 20-24, 2014)**

**Publicly Available**

http://proceedings.asmedigitalcollection.asme.org/proceeding.aspx?articleID=1937745

Proceedings of the ASME 2014 Pressure Vessels & Piping Conference
PVP2014
July 20-24, 2014, Anaheim, California, USA

PVP2014-29093

# LICENSE RENEWAL ENVIRONMENTAL FATIGUE SCREENING APPLICATION

**Christopher T. Kupper**
Westinghouse Electric Company
Cranberry Township, PA, USA

**Mark A. Gray**
Westinghouse Electric Company
Cranberry Township, PA, USA

## ABSTRACT

In NUREG-1801 (GALL) Revision 0 and Revision 1, the US Nuclear Regulatory Commission (NRC) defined the locations evaluated in NUREG/CR-6260 as a minimum acceptable set for evaluation of environmentally assisted fatigue (EAF), in addressing license renewal for nuclear plant components. Within GALL Revision 2, the NRC revised the expectation, so that plants also investigate the possibility of other locations being more limiting. To address GALL Revision 2 and NUREG-1800 Revision 2, an EAF screening methodology was developed that considers all Safety Class 1 reactor coolant pressure boundary components in major equipment and piping systems that are susceptible to EAF, including those locations listed in NUREG/CR-6260. While the overall screening process steps are similar to those published by EPRI, elements of the detailed application of some steps were performed using alternative techniques. The screening process utilized the comprehensive database of plant component fatigue qualifications available in NSSS vendor documentation, and yielded a comprehensive list of lead indicator locations for EAF consideration. This paper describes the overall process and alternate methods in the context of a specific plant license renewal application.

## INTRODUCTION

Nuclear power plants were originally given a license to operate for 40 years. To extend the license to allow for continued operation through 60 years, evaluations are required to ensure that the 40-year current licensing basis remains applicable. The evaluations to justify operation through the license renewal period include the evaluation of Time Limited Aging Analyses (TLAA). TLAAs are plant-specific safety analyses that are based on explicitly assumed 40-year plant life. The evaluation of metal fatigue is included as a TLAA. Metal fatigue is the loss of structural integrity due to fluctuating stresses. NRC Generic Safety Issue (GSI) 190 [1] was established to address residual concerns regarding the environmental effects of fatigue on pressure boundary components for 60-years of plant operation. The scope of GSI-190 included design-basis fatigue transients, studying the probability of fatigue failure of selected metal components for 60-year plant life. Studies related to the resolution of GSI-190 showed that some components have cumulative probabilities of crack initiation and through-wall growth that approach unity within the 40- to 60-year period. Therefore, it was concluded that environmentally assisted fatigue degradation should be addressed in aging management programs developed for license renewal as stated in the Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (NUREG-1800) [2].

Fatigue evaluations for a sample set of components in the reactor coolant pressure boundary were performed in NUREG/CR-6260 [3], including the effects of the reactor water environment. The sample set consisted of components from facilities designed by each of the four United States nuclear steam supply system vendors. For each facility, six locations were studied. Revisions 0 and 1 of the Generic Aging Lessons Learned (GALL) Report (NUREG-1801) [4] defined the sample set of components from NUREG/CR-6260 as the minimum acceptable set for evaluation of environmentally assisted fatigue. However, within GALL Rev. 2, the NRC revised the expectation so that operating plants address the concern of other locations that may be more limiting than the sample set. GALL Rev. 2 explicitly states in Section: X.M1 FATIGUE MONITORING, under the “Evaluation and Technical Basis” item 1. Scope of Program:

> *For purposes of monitoring and tracking, applicants should include, for a set of sample reactor coolant system components, fatigue usage calculations that consider the effects of the reactor water environment. This sample set should include the locations identified in NUREG/CR-6260 and additional plant-specific component locations in the reactor coolant pressure boundary if they may be more limiting than those considered in NUREG/CR-6260.*

 

NUREG-1800 Rev. 2 also states in Section 4.3.2.1.3 "Environmental Fatigue Calculations for Code Class 1 Components":

*Applicants should consider adding additional component locations if they are considered to be more limiting than those considered in NUREG/CR-6260.*

To determine if there were any possible locations that may be more limiting than the NUREG/CR-6260 locations, an environmental fatigue screening evaluation was performed. The EAF screening evaluation is a detailed review of the current licensing basis (CLB) fatigue evaluations for all Safety Class 1 reactor coolant pressure boundary components in major equipment and piping systems, including the NUREG/CR-6260 locations, to determine the lead indicator locations in terms of EAF. A report was published by EPRI (1024995) [5] to document a methodology associated with performing a plant-specific EAF screening evaluation. The overall methodology described in this paper is similar to the EPRI report but differs in that it utilizes a comprehensive database of plant component fatigue qualifications available in NSSS vendor documentation.

## NOMENCLATURE

NRC – Nuclear Regulatory Commission
GALL – Generic Aging Lessons Learned
EAF – Environmentally Assisted Fatigue
EPRI – Electric Power Research Institute
NSSS – Nuclear Steam Supply System
TLAA – Time Limited Aging Analysis
GSI – Generic Safety Issue
$F_{en}$ – Environmental Fatigue Correction Factor
AOR – Analysis of Record
CUF – Cumulative Usage Factor
$CUF_{en}$ – CUF with Environmental Effects
P&ID – Piping and Instrumentation Diagram
PWR – Pressurized Water Reactor
DO – Dissolved Oxygen
CLB – Current Licensing Basis
FE – Finite Element
$T^*$ – Transformed Temperature [6-8]
$O^*$ – Transformed Dissolved Oxygen Content [6-8]
$\dot{\varepsilon}^*$ – Transformed Strain Rate [6-8]
$S^*$ – Transformed Sulfur Content [6-8]

## METHOD OVERIEW

To perform the EAF screening process for a particular plant, all of the Safety Class 1 reactor coolant pressure boundary components that are susceptible to EAF were reviewed and categorized into common groups for the purpose of identifying leading locations for EAF consideration. These leading locations supplement those identified in NUREG/CR-6260, resulting in a comprehensive list of plant-specific lead indicator locations for EAF consideration. The process developed is outlined below, and then described in more detail in the subsequent sections.

A methodology was also proposed in EPRI Report 1024995 to perform an EAF screening evaluation. The overall method is similar to that described here. The only fundamental difference between the proposed EPRI method and the method described in this paper is in the application of the step to compare component fatigue usage on a common basis with respect to the stress analysis methods used for qualification. The approach described herein utilizes a large database of component fatigue evaluations and related experience to establish the analysis method basis of comparison, as opposed to a more rudimentary approach utilizing a combination of new analysis and approximations.

The process elements of the overall screening method are summarized below:

1. Data Collection
   a. All of the pertinent inputs must be collected from the utility. This includes all of the Safety Class 1 reactor coolant pressure boundary component drawings, materials, and current licensing basis fatigue evaluations.
2. Transient Section Considerations
   a. Determine the transient sections for all piping systems and major equipment included in the screening evaluation. Components within a common transient section are evaluated initially as a group before they are compared against components in other transient sections within the same system/equipment.
3. Screening $F_{en}$ Application
   a. Determine the maximum environmental fatigue correction factor ($F_{en}$) for each transient section based on material (Material $F_{en}$) and calculate a screening cumulative usage with EAF for each component (Material $CUF_{en}$). Components with a Material $CUF_{en}$ < 1.0 can be eliminated from consideration at this point.
   b. As needed, calculate refined estimated $F_{en}$ factors for each component in the transient section based on maximum section temperature (Temperature $F_{en}$) to reduce the $CUF_{en}$ (Temperature $CUF_{en}$) to a value below 1.0. Components with a Temperature $CUF_{en}$ < 1.0 can be eliminated from consideration at this point.
4. Stress Basis Comparison
   a. Determine the level of technical rigor and qualification criteria used in the stress and fatigue evaluation for each component within the transient section.
   b. Using a consistent stress analysis method ranking basis for comparison, assign a rank to each limiting component within a transient section based on the information developed in step 4a.

5. Leading Location Identification

   a. Those components with a Material (step 3a) or Temperature (step 3b) screening $CUF_{en}$ less than 1.0 are removed from the final leading location list.

   b. Identify the location(s) with the maximum screening $CUF_{en}$ and least conservative method of stress analysis in each transient section.

   c. Compare components of different transient sections within common systems/equipment. This may require additional stress basis comparisons to determine one or two leading locations per system/equipment.

   d. Compare candidate leading locations against any NUREG/CR-6260 locations within the system. Those components with a screening $CUF_{en}$ less than the NUREG/CR-6260 location and with the same or lower analysis method ranking are removed from the final set of leading locations.

Each of these process elements is expanded in the subsequent sections of this paper.

## DATA COLLECTION

All of the required data necessary to perform an EAF screening evaluation must be obtained. This information may include the following:

1. Component geometries
2. Component material properties
3. Component fatigue analysis of record (AOR)
4. Plant transient characteristics and/or specifications
5. Drawings:
   a. P & IDs
   b. Isometrics
   c. Detailed component drawings
6. Plant water chemistry requirements

## TRANSIENT SECTIONS

A transient section is defined as a group of sub-components/locations that experience the same transients. Components that reside in the same transient section can be compared with each other to determine the most limiting component (or leading location). The differences in stresses experienced by each component in a transient section are generally the result of the material and geometry differences, and possibly differing stress analysis methods.

## SCREENING FEN APPLICATION

When performing an EAF evaluation, GALL Revision 2 recommends either guidance from NUREG/CR-5704 [6] for austenitic stainless steels, NUREG/CR-6583 [7] for carbon and low alloy steels, and NUREG/CR-6909 [8] for nickel alloy steels, or guidance from NUREG/CR-6909 for all materials. Note that if NUREG/CR-6909 is used, its corresponding fatigue curves must be considered in the EAF screening process. For the methodology described in this paper, NUREG/CR-5704 is used for austenitic stainless steels, NUREG/CR-6583 is used for carbon and low alloy steels, and NUREG/CR-6909 is used for nickel alloy steels.

For screening, the maximum Material $F_{en}$ is first calculated using bounding assumptions regarding the parameters used in the $F_{en}$ equations from the NUREG reports (temperature, dissolved oxygen, sulfur content, strain rate). Generally, parameters are chosen such that the maximum $F_{en}$ penalty factor for the material is calculated, with the exception of dissolved oxygen content. A value of 0.005 ppm is used for the dissolved oxygen (DO) content, which is typical of the PWR environment. For PWRs, the DO content is generally well below 0.05 ppm, except for short periods during heatup/cooldown operations. However, elevated DO content usually only occurs when reactor coolant temperature is low. During these periods of operation, fluid temperatures are in the range where T* = 0 and the T*O* term in the applicable $F_{en}$ equation still reduces to zero.

The Material $F_{en}$ is applied to the current licensing basis (CLB) CUF to calculate a screening cumulative usage with EAF for each component (Material $CUF_{en}$). If required, CLB fatigue curve adjustments are considered. Components with a Material $CUF_{en} < 1.0$ are eliminated from consideration at this point. The basis for elimination is that a detailed EAF evaluation of these components would result in a lower $CUF_{en}$ than that obtained using the bounding maximum material penalty, and therefore would remain below 1.0. Eliminating such locations allows the screening comparisons to focus on the remaining locations that are more limiting.

For the locations still remaining in the transient section, refined estimated $F_{en}$ penalty factors are calculated for each component within the transient section based on temperature in an effort to reduce the $CUF_{en}$ to a value below 1.0 (Temperature $F_{en}$ and Temperature $CUF_{en}$). The maximum Temperature $F_{en}$ is calculated using the same assumptions as the maximum Material $F_{en}$, except that the maximum temperature for each transient section is input to the applicable NUREG equation. Components with a Temperature $CUF_{en} <$ 1.0 are eliminated from consideration at this point. The basis for elimination is that a detailed EAF evaluation of these components would result in a lower $CUF_{en}$ than that obtained using the bounding maximum temperature penalty, and therefore would remain below 1.0. Eliminating such locations allows the screening comparisons to focus on the remaining locations that are more limiting. If the Temperature $CUF_{en}$ is greater than 1.0, this location is retained for the next step. The Temperature $CUF_{en}$ is the final screening $CUF_{en}$.

The applicable NUREG $F_{en}$ penalty factor equations, as well as the assumptions used for screening, are shown below for each material.

<u>Austenitic Stainless Steels (NUREG/CR-5704):</u>

$$F_{en} = \exp(\,0.935 - T^* * O^* * \dot{\varepsilon}^*)$$

Max $F_{en}$ = 15.348* when:

Service Temperature, T = 200°C (392°F) or higher*

Dissolved Oxygen, DO = 0.005 ppm
Strain Rate, $\dot{\varepsilon}$ = 0.0004%/sec or lower
**below 392°F the Max $F_{en}$ = 2.547*

Carbon Steels (NUREG/CR-6583):
$$F_{en} = \exp(\,0.554 - 0.101 * S^* * T^* * O^* * \dot{\varepsilon}^*)$$
Max $F_{en}$ = 1.740 for Carbon when:
Sulfur Content, S = 0.015 weight percent or higher
Dissolved Oxygen, DO = 0.005 ppm
Service Temperature, T = 350°C (662°F) or higher*
Strain Rate, $\dot{\varepsilon}$ = 0.001%/sec or lower*
**since O* is 0.0 for PWR conditions, S*, $\dot{\varepsilon}^*$, & T* are irrelevant for Carbon and LAS*

Low Alloy Steels (NUREG/CR-6583):
$$F_{en} = \exp(\,0.898 - 0.101 * S^* * T^* * O^* * \dot{\varepsilon}^*)$$
Max $F_{en}$ = 2.455 for LAS when:
Sulfur Content, S = 0.015 weight percent or higher
Dissolved Oxygen, DO = 0.005 ppm
Service Temperature, T = 350°C (662°F) or higher*
Strain Rate, $\dot{\varepsilon}$ = 0.001%/sec or lower*
**since O* is 0.0 for PWR conditions, S*, $\dot{\varepsilon}^*$, & T* are irrelevant for Carbon and LAS*

Ni-Cr-Fe- Alloys (NUREG/CR-6909):
$$F_{en} = \exp(\,-T^* * O^* * \dot{\varepsilon}^*)$$
Max $F_{en}$ = 4.524* when:
Service Temperature, T = 325°C (617°F) or higher*
Dissolved Oxygen, DO = 0.005 ppm
Strain Rate, $\dot{\varepsilon}$ = 0.0004%/sec or lower
**Below 617°F, T* is a function, and maximum $F_{en}$ is calculated as a function of the maximum temperature of the transient section.*

## STRESS BASIS COMPARISON

A major consideration in the comparison process for a plant-wide screening of this nature is the fact that different stress analysis techniques may have been used for each component CLB usage factor calculation. For example, presume there is a piping component that was analyzed using NB-3600 analysis methods and yielded a usage factor of 0.9. Also, presume there is another component in the same transient section that had a usage factor of 0.9, but was qualified using NB-3200 plastic analysis methods. Although both of these locations have the same usage factor, the amount of technical rigor that was applied to the second component far exceeds that of the first component. Therefore, the screening method must consider the various stress analysis methods and techniques that were used in the usage factor evaluation.

The limiting locations for each of the transient sections within a system/equipment are initially compared against each other only and are not yet compared against other components of different piping systems or equipment. This reduces the need to perform a plant-wide stress basis comparison of all components at the outset.

When performing such an assessment, the following stress analysis characteristics are considered in determining the limiting locations within a given transient section:

1. Qualification Criteria (NB-3200, NB-3600, etc.)
2. Stress Analysis Method
   a. Interaction Analysis
   b. Simplified or One-Dimensional Analysis (e.g., NB-3600 formula, etc.)
   c. Finite Element Analysis
      i. Thermal
      ii. Mechanical
   d. Elastic/Plastic Analysis

To perform these stress basis comparisons, a hierarchy of stress analysis methods was developed based on fatigue analysis experience to define the relative complexity of the various methods. In general, fatigue analysis performed to NB-3200 criteria are regarded as more complex than those performed to NB-3600 criteria. The hierarchy used is presented below, ordered from the least complex to the most complex methods within typical NB-3200 and NB-3600 analyses. Note that combinations of the various methods are assessed on a case by case basis.

1) Standard NB-3600 analysis
2) NB-3600 with <u>mechanical</u> FE stress quantities substituted in stress formulas
3) NB-3600 with <u>thermal</u> FE stress quantities substituted in stress formulas
4) Combination of 2) and 3)
5) NB-3200 Fatigue Analysis:
   a. NB-3200 with interaction analysis
   b. NB-3200 with elastic FE analysis
   c. NB-3228 Plastic analysis

Note that this hierarchical list is used primarily for the stress-based comparison of the piping components. Since the majority of the components associated with the equipment are performed using an NB-3200 analysis, the equipment components are ranked using an independent ranking system, similar to category 5) above, within each piece of equipment, based on the amount of conservatism in the analysis.

In executing the stress basis comparison, elimination of the location with the lower final screening $CUF_{en}$ value and analysis method ranking is justified, since, if it were analyzed with the same rigor as the retained location, its CUF would be even lower, and result in an even lower $CUF_{en}$. For cases where a clear qualitative assessment cannot be completed, it may be required to perform additional analyses to determine which component is actually the most limiting for a given transient section or system. However, it is not expected that many of these cases will exist during the screening process.

Generally, within a transient section there are only one or two components at most where more advanced stress analysis methods were used relative to the other components in the same

transient section. Therefore, determining the most limiting location within a transient section is usually not a difficult task, since the component where the most technical rigor was applied is often the limiting location within a transient section. Note that components in a common transient section will have similar screening $F_{en}$ correction factors applied since they experience the same transients.

However, comparisons made between components in the same system/equipment, but different transient sections, must consider the potential differences in the $F_{en}$ correction factors. These comparisons must also consider the previously discussed stress basis comparison characteristics in determining the ultimate leading location for each system/equipment.

## LEADING LOCATION IDENTIFICATION

Stage 1: Determine Transient Section Leading Locations

Stage 1 (Steps 5a and 5b from Method Overview Section) has two aspects: Screening $F_{en}$ application and transient section stress basis comparisons. The ultimate goal of Stage 1 is to determine the leading locations for each transient section.

Once the appropriate screening $F_{en}$ factors are calculated for the components in each transient section, they can be applied to those components. Those components with a screening $CUF_{en}$ of less than 1.0 can be removed from the list as explained previously.

A stress basis comparison must be performed on remaining transient section components. Typically, a stress basis comparison will yield one or two distinct leading locations. If a further reduction in the number of leading locations is required, then a more detailed quantitative analysis can be performed. The remaining components represent the leading locations for EAF consideration for each transient section.

Stage 2: Determine System/Equipment Level Leading Locations

Stage 2 (Step 5c from Method Overview Section) involves components that reside in different transient sections, but are within a common system or piece of major equipment. These can also be compared to determine one or two leading locations to represent their respective system/equipment. Often, it is the transients themselves that control which components have the highest usage factors in a given system. So within a particular system, those transient sections with the most severe system transients will usually have components with the highest usage factors. However, the comparison of components in different transient sections must be done after the appropriate $F_{en}$ correction factor is applied to the component usage factor, and stress basis ranking is applied. This is because the $F_{en}$ correction factor is dependent on temperature and strain rate and therefore can vary for each transient section.

The ultimate goal is to compare those components within a common system against each other to determine the leading locations on a system/equipment level. Similarly to Stage 1, when comparing components of different transient sections, but of common systems/equipment, it is necessary to consider the stress analysis characteristics outlined previously.

Stage 3: Comparison of System/Equipment Level Leading Locations with NUREG/CR-6260 locations

Stage 3 (Step 5d from Method Overview Section) involves comparing the leading locations to the NUREG/CR-6260 locations. For those systems where the NUREG/CR-6260 locations have the highest screening $CUF_{en}$, no additional locations need to be considered for EAF. For those systems where the NUREG/CR-6260 locations do not have the highest screening $CUF_{en}$ or a NUREG/CR-6260 location does not exist, the locations within that system that have the highest analysis ranking and screening $CUF_{en}$ should be considered as leading locations. In cases where multiple locations could be the leading location, additional analyses may need to be performed (as stated in Stages 1 and 2) to determine the most limiting location, or each of the locations should be considered for further EAF management.

The screening process yields a comprehensive and plant-specific list of locations that supplement those evaluated in NUREG/CR-6260 as lead indicator locations for EAF consideration. The final set of leading locations should then be included in the plant's fatigue management program to ensure that the $CUF_{en}$ for each component remains below 1.0 for the duration of plant operation.

## APPLICATION OF METHODOLOGY

The EAF screening methodology described above was applied to a plant with major equipment and piping designed to ASME Code Section III. An example of the results of applying the process for one piping system is provided below. The example system is the cold leg safety injection/accumulator piping in a Westinghouse PWR, as shown in Figure 1.

Upon gathering the CLB data, the system transient sections were defined consistent with the CLB transient definitions and component fatigue evaluations.

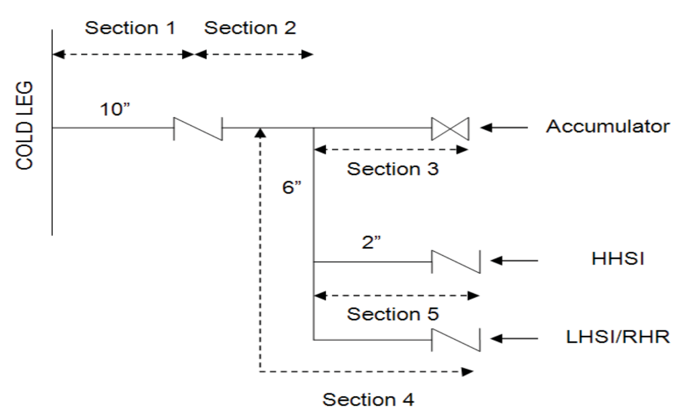


**FIGURE 1: SAFETY INJECTION/ACCUMULATOR - TRANSIENT SECTIONS**

Tables 1 through 5 show the components in each section and illustrate the results of applying Steps 3 through 5b from the Method Overview Section. The Analysis Ranking values correspond to the hierarchical list as described in the Stress Basis Comparison Section.

**TABLE 1: SAFETY INJECTION/ACCUMULATOR EAF SCREENING - TRANSIENT SECTION 1**

| Section | Component | Design CUF | Material (1=SS, 2=CS, 3=LAS, 4=Ni Alloy) | Material $F_{en}$ | Material $CUF_{en}$ | Max Temp > 392°F (1=yes, 2=no) | Temp $F_{en}$ | Temp $CUF_{en}$ | Final EAF Screening $CUF_{en}$ | Analysis Ranking |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | **RCL Nozzle** | **0.95** | **1** | **15.35** | **14.58** | **1** | **15.35** | **14.58** | **14.58** | **4** |
| | Elbow | 0.0875 | 1 | 15.35 | 1.34 | 1 | 15.35 | 1.34 | 1.34 | 1 |
| | Butt Weld | 0.099 | 1 | 15.35 | 1.52 | 1 | 15.35 | 1.52 | 1.52 | 1 |
| | Small Branch/Plug | 0.099 | 1 | 15.35 | 1.52 | 1 | 15.35 | 1.52 | 1.52 | 1 |
| | Valve Butt Weld | 0.5368 | 1 | 15.35 | 8.24 | 1 | 15.35 | 8.24 | 8.24 | 3 |
| | Valve | 0.48 | 1 | 15.35 | 7.37 | 1 | 15.35 | 7.37 | 7.37 | 1 |

As shown in Table 1, all components have a screening $CUF_{en}$ > 1.0 after the maximum Material and Temperature specific $F_{en}$ penalty factors were applied. All components were analyzed using NB-3600 equations (analysis ranking = 1) except for the RCL Nozzle and the valve butt weld. The section 1 RCL nozzle was qualified to NB-3600 but using Finite Element Analysis for thermal and mechanical stress quantities (analysis ranking = 4). The section 1 valve butt weld was qualified to NB-3600 but using Finite Element Analysis only for thermal stress quantities (analysis ranking = 3). The final screening $CUF_{en}$ for the section 1 valve butt weld is less than that of the section 1 RCL Nozzle, and the section 1 valve butt weld was qualified using a less rigorous analysis methodology than the section 1 RCL Nozzle. Therefore, it is concluded that the section 1 RCL Nozzle is more limiting than the section 1 valve butt weld. The RCL Nozzle is chosen as the limiting location from section 1.

**TABLE 2: SAFETY INJECTION/ACCUMULATOR EAF SCREENING - TRANSIENT SECTION 2**

| Section | Component | Design CUF | Material (1=SS, 2=CS, 3=LAS, 4=Ni Alloy) | Material $F_{en}$ | Material $CUF_{en}$ | Max Temp > 392°F (1=yes, 2=no) | Temp $F_{en}$ | Temp $CUF_{en}$ | Final EAF Screening $CUF_{en}$ | Analysis Ranking |
|---|---|---|---|---|---|---|---|---|---|---|
| 2 | Valve Butt Weld | 0.091 | 1 | 15.35 | 1.4 | 2 | 2.547 | 0.23 | | |
| | Elbow | 0.061 | 1 | 15.35 | 0.94 | | | | | |
| | 10" x 10" x 6" Tee | 0.093 | 1 | 15.35 | 1.43 | 2 | 2.547 | 0.24 | | |
| | Valve | 0.48 | 1 | 15.35 | 7.37 | 2 | 2.547 | 1.22 | 1.22 | 1 |

In Table 2 for section 2, with the exception of the valve, all components have a screening $CUF_{en}$ < 1.0 after maximum Material and Temperature specific $F_{en}$ penalty factors were applied. The valve was qualified to NB-3545 (analysis ranking = 1). Because it is the only location remaining, the valve is chosen as the limiting location from section 2.

**TABLE 3: SAFETY INJECTION/ACCUMULATOR EAF SCREENING - TRANSIENT SECTION 3**

| Section | Component | Design CUF | Material (1=SS, 2=CS, 3=LAS, 4=Ni Alloy) | Material $F_{en}$ | Material $CUF_{en}$ | Max Temp > 392°F (1=yes, 2=no) | Temp $F_{en}$ | Temp $CUF_{en}$ | Final EAF Screening $CUF_{en}$ | Analysis Ranking |
|---|---|---|---|---|---|---|---|---|---|---|
| 3 | Elbow | 0.002 | 1 | 15.35 | 0.03 | | | | | |
| | 10" x 3/4" Branch | 0.086 | 1 | 15.35 | 1.33 | 2 | 2.547 | 0.22 | | |
| | Valve Butt Weld | 0.001 | 1 | 15.35 | 0.01 | | | | | |
| | Valve | 0.19 | 1 | 15.35 | 2.92 | 2 | 2.547 | 0.48 | | |

**TABLE 4: SAFETY INJECTION/ACCUMULATOR EAF SCREENING - TRANSIENT SECTION 4**

| Section | Component | Design CUF | Material (1=SS, 2=CS, 3=LAS, 4=Ni Alloy) | Material $F_{en}$ | Material $CUF_{en}$ | Max Temp > 392°F (1=yes, 2=no) | Temp $F_{en}$ | Temp $CUF_{en}$ | Final EAF Screening $CUF_{en}$ | Analysis Ranking |
|---|---|---|---|---|---|---|---|---|---|---|
| 4 | Elbow | 0.007 | 1 | 15.35 | 0.11 | | | | | |
| | 6" x 2" Branch | 0.243 | 1 | 15.35 | 3.73 | 2 | 2.547 | 0.62 | | |
| | Valve Butt Weld | 0.013 | 1 | 15.35 | 0.2 | | | | | |
| | Valve | 0.29 | 1 | 15.35 | 4.45 | 2 | 2.547 | 0.74 | | |

**TABLE 5: SAFETY INJECTION/ACCUMULATOR EAF SCREENING - TRANSIENT SECTION 5**

| Section | Component | Design CUF | Material (1=SS, 2=CS, 3=LAS, 4=Ni Alloy) | Material $F_{en}$ | Material $CUF_{en}$ | Max Temp > 392°F (1=yes, 2=no) | Temp $F_{en}$ | Temp $CUF_{en}$ | Final EAF Screening $CUF_{en}$ | Analysis Ranking |
|---|---|---|---|---|---|---|---|---|---|---|
| 5 | Socket Weld | 0.1 | 1 | 15.35 | 1.54 | 2 | 2.547 | 0.25 | | |
| | Pipe | 0.016 | 1 | 15.35 | 0.25 | | | | | |
| | Elbow | 0.1 | 1 | 15.35 | 1.54 | 2 | 2.547 | 0.25 | | |

As shown in Tables 3, 4, and 5, all components in transient sections 3, 4, and 5 have a screening $CUF_{en}$ < 1.0 after maximum material and temperature specific $F_{en}$ penalty factors were applied. Therefore, all components in transient sections 3, 4, and 5 can be eliminated from consideration. The basis for elimination is that a detailed EAF evaluation of these components would result in a lower $CUF_{en}$ than that obtained using the bounding maximum material penalty, and therefore would remain below 1.0.

Applying Step 5c for the five transient sections in the cold leg safety injection accumulator system, only two potential leading locations remain. The final screening $CUF_{en}$ for the section 2 valve is less than that of the section 1 RCL Nozzle, and the section 2 valve was qualified using a less rigorous analysis methodology (lower analysis ranking) than the section 1 RCL Nozzle. Therefore, it is concluded that the section 1 RCL Nozzle is more limiting than the section 2 valve. The RCL Nozzle is chosen as the limiting location for the system.

The final step (5d) applies for this system, since the RCL Nozzle is a NUREG/CR-6260 location. The step is trivial in this example since the nozzle is the final leading location for the system.

Similar methodology would be used for performing an EAF screening evaluation with piping not designed to ASME Code Section III. The difference is in the data collection step. Because the current licensing basis for the Safety Class 1 piping does not include an ASME Code Section III fatigue evaluation, plant-specific usage factors are not available for each piping component. Instead, the comprehensive database of plant component fatigue qualifications available in NSSS vendor documentation is utilized. Comparisons are made between the plant-specific piping components and those which are available in the database to justify the applicability and make relative comparisons for screening purposes. Such comparisons include materials, geometry, and transients.

The final result of the process applied to all plant Class 1 systems and components is a list of locations requiring further evaluation and/or inclusion in the fatigue management program. This list includes the NUREG/CR-6260 locations plus any additional screened-in locations.

## PHASE 2 EAF SCREENING

The EAF screening process described above can be considered a first-pass evaluation (Phase 1) designed to identify the potential locations of concern for further EAF analysis. The next step of the EAF screening process (Phase 2) is to perform a more refined EAF evaluation for the identified leading locations if required.

The goal of Phase 2 is to minimize the number of detailed analyses (e.g., finite element or integrated $CUF_{en}$ calculations) required for the identified leading locations. For this step, the screening $CUF_{en}$ values can potentially be reduced below 1.0 by using one (or a combination) of the following methods:

1. Application of projected transient cycles for 60 years
2. Reduction of conservatism in existing stress calculations (without performing any new detailed analyses)
3. Comparison with similar component detailed analyses $F_{en}$ refinement

## CONCLUSIONS

Revision 2 of the NRC GALL report required plants applying for license renewal to consider the effects of reactor water environment on fatigue for the sample set of components defined in NUREG/CR-6260 as well as any other reactor coolant pressure boundary component(s) that may be more limiting. To determine if there are any locations more limiting than the NUREG/CR-6260 sample set, a plant-wide environmental fatigue screening evaluation has been described.

The ultimate goal of the EAF screening evaluation was to compare all Safety Class 1 reactor coolant pressure boundary components within common systems to determine the leading locations with respect to EAF. Components within each system were compared on the bases of common transients and common stress analysis methods to determine the most limiting locations. A systematic method for comparison of components and stress analysis methods has been described. The comparisons are based on common data and experience from plant component fatigue evaluation methods and results. The approach can also be supplemented by the NSSS vendor fatigue evaluation database for similar components if required. The list of leading locations can be further reduced without detailed analysis by removing conservatism from the fatigue analysis of record (including application of 60-year projected cycles) or further refinement of the $F_{en}$ penalty factors. The final list of leading locations should be included in the aging management program for the plant.

## REFERENCES


[1] Thadani, A. C., 1999, "GENERIC SAFETY ISSUE 190, FATIGUE EVALUATION OF METAL COMPONENTS FOR 60-YEAR PLANT LIFE," GSI-190, NRC Office of Nuclear Regulatory Research, Washington, DC.

[2] U.S. Nuclear Regulatory Commission, 2010, "STANDARD REVIEW PLAN FOR REVIEW OF LICENSE RENEWAL APPLICATIONS FOR NUCLEAR POWER PLANTS," NUREG-1800, Division of License Renewal, Office of Nuclear Reactor Regulation, Washington, DC.

[3] Ware, A. G., Morton, D. K., and Nitzel, M. E., 1995, "APPLICATION OF NUREG/CR-5999 INTERIM FATIGUE CURVES TO SELECTED NUCLEAR POWER PLANT COMPONENTS," NUREG/CR/6260, Idaho National Engineering Laboratory, Idaho Falls.

[4] U.S. Nuclear Regulatory Commission, "GENERIC AGING LESSONS LEARNED (GALL) REPORT," NUREG-1801 All Revisions, Division of License Renewal, Office of Nuclear Reactor Regulation, Washington, DC.

[5] Electric Power Research Institute, 2012, "ENVIRONMENTALLY ASSISTED FATIGUE SCREENING PROCESS AND TECHNICAL BASIS FOR IDENTIFYING EAF LIMITING LOCATIONS," EPRI Report No. 1024995, Structural Integrity Associates, Inc, San Jose.

[6] Chopra, O. K., 1999, "EFFECTS OF LWR COOLANT ENVIRONMENTS ON FATIGUE DESIGN CURVES OF AUSTENITIC STAINLESS STEELS," NUREG/CR-5704, Argonne National Laboratory, Argonne.

[7] Chopra, O. K. and Shack, W. J., 1998, "EFFECTS OF LWR COOLANT ENVIRONMENTS ON FATIGUE DESIGN CURVES OF CARBON AND LOW-ALLOY STEELS," NUREG/CR-6583, Argonne National Laboratory, Argonne.

[8] Chopra, O. K. and Shack, W. J., 2007, "EFFECT OF LWR COOLANT ENVIRONMENTS ON THE FATIGUE LIFE OF REACTOR MATERIALS," NUREG/CR-6909, Argonne National Laboratory, Argonne.

**Attachment 2**

***EAF Screening: Process and Technical Basis for Identifying EAF Limiting Locations***

**EPRI Report 1024995**

**August 2012**

**Publicly Available at:**

**http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001024995**

ELECTRIC POWER RESEARCH INSTITUTE

# Environmentally Assisted Fatigue Screening

## Process and Technical Basis for Identifying EAF Limiting Locations

2012 TECHNICAL REPORT

# Environmentally Assisted Fatigue Screening

## *Process and Technical Basis for Identifying EAF Limiting Locations*

This document does NOT meet the requirements of 10CFR50 Appendix B, 10CFR Part 21, ANSI N45.2-1977 and/or the intent of ISO-9001 (1994).


EPRI Project Manager
S. Chu

ELECTRIC POWER RESEARCH INSTITUTE

3420 Hillview Avenue
Palo Alto, CA 94304-1338
USA

PO Box 10412
Palo Alto, CA 94303-0813
USA

800.313.3774
650.855.2121

askepri@epri.com

www.epri.com



**1024995**
Final Report, August 2012

THE FOLLOWING ORGANIZATION, UNDER CONTRACT TO EPRI, PREPARED THIS REPORT:

**Structural Integrity Associates, Inc.**



**NOTE**

For further information about EPRI, call the EPRI Customer Assistance Center at 800.313.3774 or
e-mail askepri@epri.com.

# Acknowledgments

The following organization, under contract to the Electric Power Research Institute (EPRI), prepared this report:

Structural Integrity Associates, Inc.
5215 Hellyer Avenue, Suite 210
San Jose, CA  95138

Principal Investigators
D. Gerber
C. Carney
T. Gilman
T. Herrmann

This report describes research sponsored by EPRI.

The work for this report was completed with the support and input of industry reviewers.

This publication is a corporate document that should be cited in the literature in the following manner:

*Environmentally Assisted Fatigue Screening: Process and Technical Basis for Identifying EAF Limiting Locations.*
EPRI, Palo Alto, CA: 2012. 1024995.

# Abstract

This report provides the technical basis and process for a screening evaluation of a nuclear power plant. This screening will identify appropriate limiting locations for systematic monitoring of the environmentally assisted fatigue (EAF) effects in a Class 1 reactor on the reactor coolant pressure boundary components that are wetted with primary coolant. Use of this process will ensure that the most limiting locations for EAF are determined on a consistent basis.

The process developed in this report provides guidance for the evaluation and relative ranking of estimated $U_{en}$ values for locations in components and systems where EAF is a concern in order to minimize the possibility of the need for a formal fatigue evaluation. The estimated values are compared to $U_{en}$ values for locations in each given system/component that have been specifically identified in regulatory guidance as being of concern for EAF. Locations from previous guidance are to be monitored, and if estimated values for other locations are higher or as high, the other locations should also be monitored. For components or systems where there are no locations included in previous regulatory guidelines, the recommendation is to monitor up to three locations with the highest estimated $U_{en}$.

This report is a public document available for reference by license renewal applicants.

**Keywords**
Design basis
Environmentally assisted fatigue
Fatigue monitoring
Fatigue usage
License renewal

# Acronyms

The following acronyms are used in this report.

| | |
|---|---|
| ASME | American Society of Mechanical Engineers |
| BWR | Boiling Water Reactor |
| CS | Carbon Steel |
| CVCS | Chemical and Volume Control System |
| CUF | Cumulative Usage Factor |
| DSR | Design Stress Report |
| DO | Dissolved Oxygen |
| EAF | Environmentally Assisted Fatigue |
| EPRI | Electric Power Research Institute |
| $F_{en}$ | Environmental Fatigue Correction Factor |
| FMP | Fatigue Management Program |
| FSRF | Fatigue Strength Reduction Factor |
| FSAR | Final Safety Analysis Report |
| HWC | Hydrogen Water Chemistry |
| ID | Inside Diameter |
| LRA | License Renewal Application |
| LWR | Light Water Reactor |
| LAS | Low-Alloy Steel |
| NRC | Nuclear Regulatory Commission |
| NWC | Normal Water Chemistry |
| OD | Outside Diameter |
| P&ID | Piping and Instrumentation Drawing |
| PWR | Pressurized Water Reactor |
| RCS | Reactor Coolant System |
| RHX | Regenerative Heat Exchanger |
| RAI | Request for Additional Information |
| SS | Stainless Steel |
| $S_m$ | Design Stress Intensity Value |
| SCF | Stress Concentration Factor |
| U | ASME Code Cumulative Usage Factor |
| $U_{en}$ | EAF Cumulative Usage Factor |

# Table of Contents

# List of Figures

# List of Tables

# Section 1: Introduction

## Background

The NRC requires license renewal applicants to assess the fatigue usage effects from a reactor water environment and demonstrate acceptable fatigue cumulative usage factors (CUF) with the effects of a reactor water environment considered for Class 1 components for the entire period of extended operation. This is commonly referred to as environmentally assisted fatigue (EAF).

This demonstration requires an EAF evaluation and screening of Class 1 components, some of which lack a CUF calculation. Evaluating Class 1 components with a CUF calculation is straightforward and may require a relatively simple evaluation process to estimate and apply environmental fatigue correction factors ($F_{en}$) to the existing CUF values. Class 1 components without CUF calculations may require a more extensive evaluation process to evaluate plant locations on a similar stress basis and apply $F_{en}$ factors.

## Purpose of this Report

The purpose of this report is to describe the technical basis of and define a process that may be used for EAF screening and ranking of components in nuclear power plant Class 1 systems. This example process is documented to allow for consistent application, but it is not the only acceptable way to identify limiting locations. This process must be effective for PWRs and BWRs, both with ASME Section III [1] / B31.7 [2] piping and B31.1 [3] piping. This process can be used to screen plant locations in order to rank them on the basis of EAF values. These ranked locations can then be compared to the NUREG/CR-6260 [4] sample locations and may augment a plant's Fatigue Management Program (FMP).

The desired outcome of this process is to determine plant locations which can be demonstrated to bound other locations of like materials and can serve as limiting EAF locations for the plant.

## Criteria for Process

The procedure developed for this report has the following properties:

- No need for new formal stress or fatigue analysis

- Includes procedures that are practical to use, with readily available design input
- Provides appropriate ***relative*** EAF rankings of components
- Allows the use of either NUREG/CR-5704 [5] (stainless steel)/ NUREG/CR-6583 [6] (carbon and low alloy steel)/ NUREG/CR-6909 [7] (Ni-Cr-Fe) or just NUREG/CR-6909 for all materials

## Benefits to Plant Owners and NRC

### *License Renewal*

- This process will enable plant owners to demonstrate knowledge of the locations in their plant that can serve as limiting locations for EAF evaluations as the plants enter the period of extended operation. This process provides the rationale for selecting these bounding locations. Plant owners will minimize costs by avoiding the necessity of formal fatigue analyses, while meeting the regulatory requirements of determining the bounding EAF locations in the plant.
- The process will provide the NRC with a uniform approach to determination of limiting locations for EAF evaluations in license renewal applications.

# Section 2: Fatigue and EAF Basics

This section provides a definition of terms used in this report and a background on fatigue analysis and license renewal requirements.

Current plants are qualified via a design process defined in ASME Code, Section III [1] or B31.7 [2] or B31.1 [3]. For components subject to cyclic loadings, this generally includes a fatigue analysis. The ASME Code, Section III and B31.7 design codes provide rules for the explicit determination of CUF. The B31.1 rules provide rules for evaluation of cyclic loads using a cyclic reduction factor method that addresses sustained stresses and cyclic thermal moment stresses, but does not produce CUF values.

Evaluating components with a CUF calculation is straightforward and may require a relatively simple evaluation process to estimate and apply environmental fatigue correction factors ($F_{en}$) to the existing CUF values. If necessary, components without CUF calculations may require a more extensive evaluation process to evaluate plant locations on a similar stress basis and apply $F_{en}$ factors.

## Definition of Terms

*U*: Design CUF (documented in Design Stress Reports (DSR) and design analyses)

*Fatigue Table:* The compilation of incremental CUF values ($U_{incr}$) determined from load pairs in a fatigue analysis (found in DSRs and design analyses)

*Load Pair*: A row in a fatigue table representing a local maximum and minimum value of stress.

*U**: Estimated CUF produced on a common basis (produced in the Common Basis Stress Evaluation Procedure)

$U_{6260}^*$: Estimated CUF produced on a common basis for NUREG/CR-6260 location (produced in the Common Basis Stress Evaluation Procedure)

$U_{incr}^*$: Estimated incremental CUF for a load pair (produced in the Common Basis Stress Evaluation Procedure)

$F_{en}$*: Estimated $F_{en}$ produced by an evaluation that does not evaluate the $F_{en}$ value for each Load Pair in a fatigue analysis (produced in the $F_{en}$ Estimation Procedure)

$U_{en}$*: Estimated $U_{en}$ produced from Estimated CUF and Estimated $F_{en}$

$U_{en\ incr}$*: Estimated incremental $U_{en}$* for a load pair produced from $U_{incr}$* and $F_{en}$* values for each transient pair (produced in the Common Basis Stress Evaluation Procedure as $U_{en\ incr}^* = F_{en}^* \times U_{incr}^*$)

$U_{en\ 6260}$*: Estimated $U_{en}$ produced from Estimated CUF for a NUREG/CR-6260 location (produced in the Common Basis Stress Evaluation Procedure as $U_{en\ 6260}^* = F_{en}^* \times U_{6260}^*$)

$U_{en\ max}$*: The maximum value of $U_{en}$* in a set of locations

$U_{en\ max-1}$*: The second highest value of $U_{en}$* in a set of locations

$U_{en\ max-2}$*: The third highest value of $U_{en}$* in a set of locations

*Leading Transient*: A thermal or pressure transient that contributes significantly to CUF of a component. Prime examples are temperature shocks from starting and stopping flow at nozzles.

*Bundled Transients*: Enveloping of multiple plant transients by one conservative plant transient.

*Common Basis Model*: A model in which components are assessed on a common basis for ranking comparison purposes by evaluating load pairs with unbundled transients on a linear elastic basis.

## Basics of Fatigue Analysis

According to the ASME Code [1], the CUF is a value computed as the summation of incremental fatigue contributions arising from thermal and mechanical stress fluctuations in a metal component. The CUF is compared to a maximum value of 1.0 to demonstrate acceptable design behavior.

As specified in the ASME Code, the fatigue analysis procedure consists of the following steps:

- Determine the stress tensor (composed of six stress components) for all normal service conditions. Select one or more controlling component locations for evaluation. For piping components, the analysis is typically conducted for all welds and fitting locations.
- Determine the stress differences (i.e. stress intensity ranges) for all pair-wise combinations of service conditions.
- Determine the alternating stress amplitude ($S_{alt}$) for each transient pair. This must include stress concentration effects, if present. Also any additional

strain due to plasticity must be accounted for. For newer Code editions, one way to do this is with a local plastic strain "penalty" factor $K_e$, which is calculated in accordance with Section III, NB-3228.5 if the primary plus secondary stress range exceeds three times the design stress intensity ($3S_m$).

- Determine the allowable cycles ($N_a$) for each transient pair, using a fatigue curve. Prior to entering the fatigue curve (graph showing allowable cycles at various $S_{alt}$ values) to determine the allowable number of cycles, the stress amplitude must be modified by multiplying it by the ratio of the modulus of elasticity on the fatigue curve, divided by that used in the stress analysis.
- Determine fatigue usage contribution for all transient pairs where the modified stress amplitude exceeds the endurance limit of the fatigue curve. The fatigue endurance limit is the value of $S_{alt}$ corresponding to an “infinite” number of allowable cycles $N_a$ (taken as either $10^6$ or $10^{11}$ cycles depending upon material and edition of the ASME Code).

The fatigue usage factor, *u*, is determined by the summation:

$$U = \sum_{i=1}^{M} \frac{n_i}{N_i} \qquad \text{Equation 2-1}$$

where:

$n_i$ = number of stress cycles for transient pair *i*

$N_i$ = *number of allowable cycles for the transient pair i*

$M$ = total number of sets of transient pairs

In the fatigue analysis process, a component meets the design criterion if the maximum CUF for all analysis locations is less than or equal to 1.0.

## Background on EAF Calculations

Over the past two decades, concerns about EAF have arisen because the rules for design of Class 1 components in nuclear power plants do not explicitly address the effects of light water reactor (LWR) coolant environments, and these deleterious effects may be significant. Recent laboratory work has identified the influence of key parameters on fatigue crack initiation and has established the effects of these key parameters on the fatigue life of selected carbon and low alloy steels, austenitic stainless steels and nickel alloy steels used in nuclear plants [7]. With respect to these parameters, an environmental fatigue correction factor ($F_{en}$) approach has been developed to evaluate the effects of LWR environments into ASME Section III fatigue evaluations.

NRC report NUREG-1801, Revision 2, [8] the “Generic Aging Lessons Learned (GALL) Report,” identifies acceptable aging management programs for fatigue and cyclic operation for the period of extended operation. It describes a process for assessing the impact of the reactor coolant environment on a set of

sample critical components for the plant, examples of which are identified in NUREG/CR-6260 [4].

NRC guidance in NUREG/CR-5704 [5], NUREG/CR-6583 [6] and NUREG/CR-6909 [7] defines the $F_{en}$ correction factor as the ratio of the number of stress cycles in air at room temperature ($N_{air,RT}$) to that in reactor water at the service temperature ($N_{water}$). This ratio is used because fatigue usage for one stress cycle is the inverse of its allowable number of cycles (1/N), thus:

$$F_{en} = N_{air,RT} / N_{water} \qquad \textit{Equation 2-2}$$

Environmental effects are incorporated into ASME NB-3200 [1] fatigue analyses by multiplying the partial ASME usage factors for each stress cycle by the $F_{en}$ correction factor computed for that stress cycle. For example, given $n$ different stress cycle pairs, the cumulative environmental fatigue usage is:

$$U_{en} = U_1 \cdot F_{en,1} + U_2 \cdot F_{en,2} + U_3 \cdot F_{en,3} + U_i \cdot F_{en,i} \ldots + U_n \cdot F_{en,n}$$

*[7, Eq. A.20]* *Equation 2-3*

where:

$U_i$ = computed fatigue usage using the air fatigue curve for the "i'th" stress cycle

$F_{en,i}$ = computed $F_{en}$ for the "i'th" stress cycle

An effective $F_{en}$ factor is computed as follows:

$$F_{en-effective} = U_{en} / U \qquad \textit{Equation 2-4}$$

Where U is:

$$U = \sum_n U_n \qquad \textit{Equation 2-5}$$

### $F_{en}$ Formulations

Per Reference [8], plants have the option of computing $U_{en}$ in accordance with guidance from EITHER NUREG/CR-5704 [5] (for austenitic stainless steels), NUREG/CR-6583 [6] (for carbon and low alloy steels) and NUREG/CR-6909 [7] (for nickel alloy steels), OR NUREG/CR-6909 (for all materials).

In the case of NUREG/CR-5704 and NUREG/CR-6583 for austenitic stainless steel and carbon/low-alloy steel, respectively, the ASME Code fatigue curve is used, and the $F_{en}$ factors are applied to the ASME Code fatigue usage values. When using NUREG/CR-6909 rules, the special fatigue curve provided in Appendix A, page A.5 of Reference [7] must be used for austenitic stainless steel

materials, and optionally for ferritic materials (the ASME curve is more conservative for ferritic materials).

A generic equation for computing $F_{en}$ factors is:

$$F_{en} = \exp(A - (B \times S^* \times T^* \times O^* \times \dot{\varepsilon}^*)) \qquad \textit{Equation 2-6}$$

Parameters used in these equations are presented in Table 2-1 and further discussed in Appendix A.

*A* and *B* – Numeric constants derived from statistical analysis; the specific values depend on which material is involved and which NUREG is being used; see *Table 2-1*.

$S^*$ – a factor based on Sulfur Content (this term is only used for some materials—in all other cases, use $S^* = 1$).

$T^*$ – a factor based on Service Temperature.

$O^*$ – a factor based on Dissolved Oxygen.

$\dot{\varepsilon}^*$– a factor based on Strain Rate.

*Table 2-1*
*$F_{en}$ Equation Parameters*

| Material | NUREG/ CR- | A | B | $S^*$ | Fatigue Curve |
|---|---|---|---|---|---|
| Carbon Steel | 6583 | 0.554(1) | 0.101 | In equation | ASME |
| Low-Alloy Steel | 6583 | 0.898(1) | 0.101 | In equation | ASME |
| Austenitic Stainless Steel | 5704 | 0.935 | 1 | Not in equation | ASME |
| Ni-Cr-Fe Alloy | 5704 | 0.935 | 1 | Not in equation | ASME (2) |
| Carbon Steel | 6909 | 0.632 | 0.101 | In equation | Fig. A.1 (6909) |
| Low-Alloy Steel | 6909 | 0.702 | 0.101 | In equation | Fig. A.2 (6909) |
| Austenitic Stainless Steel | 6909 | 0.734 | 1 | Not in equation | Fig. A.3 (6909) |
| Ni-Cr-Fe Alloy | 6909 | 0 | 1 | Not in equation | Fig. A.3 (6909) (3) |

(1) Equation evaluated at a Test Temperature of room temperature (25°C) [6].
(2) Guidance prior to GALL, Revision 2 [8] allowed use of austenitic stainless steel equation and ASME fatigue curve for Ni-Cr-Fe steels.
(3) GALL, Revision 2 [8] cites only this option for Ni-Cr-Fe steels.

*Service Temperature* – temperature of the metal in contact with the primary fluid "environment." Per equation rules, the temperature is in °C. For all materials, higher temperature = higher $F_{en}$ (up to a maximum value).

*Test Temperature* – temperature during the tests (25°C).

*Dissolved Oxygen* – level of dissolved oxygen (DO) in the primary fluid.

*Strain Rate* – rate of strain in the metal during the increasingly tensile time periods. This is the parameter that has the largest effect on the value of $F_{en}$ for all materials.

*Sulfur Content*–the weight percent of sulfur in the material.

*Strain Amplitude Threshold* – a minimum strain amplitude below which LWR environments have no effect on the fatigue life of these steels.

### Maximum $F_{en}$ Values

In cases where parameters are uncertain, it is conservative to use the following worst-case values for the following parameters:

NUREG/CR-6909 [7]

For carbon steel and low-alloy steel:

S = 0.015 weight percent or higher

DO = 0.5 ppm or higher

T = 350˚C or higher

$\dot{\varepsilon}$ = 0.001%/sec or lower

For austenitic stainless steels and Ni-Cr-Fe alloys:

T = 325˚C or higher

$\dot{\varepsilon}$ = 0.0004%/sec or lower

NUREG/CR-5704 [5]

For austenitic stainless steel:

T = 200˚C or higher

DO = 0.05 ppm or higher

$\dot{\varepsilon}$ = 0.0004%/sec or lower

NUREG/CR-6583 [6]

For carbon steel and low-alloy steel:

S = 0.015 weight percent or higher

DO = 0.5 ppm or higher

T = 350˚C or higher

$\dot{\varepsilon}$ = 0.001%/sec or lower

### Requirements for License Renewal

Part 54 to Title 10 of the U.S. Code of Federal Regulations (10CFR54) specifies the "Requirements for Renewal of Operating Licenses for Nuclear Power Plants".

NUREG/CR-6260 [4], published in 1995, established a set of six locations (by plant model and vintage) that were expected to be representative of components that had higher CUFs and/or were important from a risk perspective. It made an explicit assumption that if those sample locations could be shown to have acceptable EAF values, then it would be possible to demonstrate the same for other similar locations in the plant.

The NRC has recently questioned whether the NUREG/CR-6260 locations effectively cover all locations in the plant (see Requests for Additional Information (RAIs) on recent license renewal applications and GALL Revision 2 [8]). This concern has led to requests for plants to demonstrate the validity of the NUREG/CR-6260 locations, or else augment the list with additional locations to cover any outliers.

Determination of this list is not as easy as multiplying each design CUF value by a factor or factors. Examples of the complicating factors are:

- Not all CUF values represent the same degree of analytical rigor.
  - Analysis of design severity plant transients produces different CUF values for a component than analysis of actual severity plant transients.
  - Analysis using "bundled transients" yield significantly higher CUF values than analyses of the same component with "un-bundled" transients
- For a given plant transient, $F_{en}$ factors often will trend counter to the computed CUF values, thus potentially complicating the ranking of the $U_{en}$ values for a component.
  - Faster rise times for a thermal transient will tend to produce lower $F_{en}$ factors, but larger CUF values. Since $U_{en} = F_{en} \times U$, the product of the two is not known a priori without further analysis.
- Analysis of design numbers of plant transients can yield different rankings of CUF and $U_{en}$ values than analyses of projected numbers of plant transients.
  - The two different mixes of plant transients, each with their unique transient characteristics, can cause the weighted $F_{en}$ factors and $U_{en}$ values to vary significantly.
- Different materials of construction exhibit different EAF characteristics, even in the same component.
  - The same plant transients applied to one component will produce different $U_{en}$ values for different material of construction.
  - DO content affects materials of construction differently:
    - Higher DO gives lower $F_{en}$ values for austenitic stainless steels [5]
    - Constant effect for all DO content for austenitic stainless steels [7]
    - Higher DO gives higher $F_{en}$ values for carbon and low-alloy steels [6,7]
    - Constant effect for all DO content for Ni-Cr-Fe steels for PWRs and HWC BWR water and constant effect for all DO content for Ni-Cr-Fe steels for NWC BWR water [7]

Further factors that influence the evaluations are:

- Use of the alternate rules of NUREG/CR-5704 [5] (stainless steel) and NUREG/CR-6583 [6] (carbon and low alloy steel) will produce somewhat different values of $F_{en}$ than the newer rules of NUREG/CR-6909 [7] for those materials.

- Components in similar plants will likely have similar estimated EAF characteristics, although some may have computed CUF values and others may not. This conclusion is based on an EPRI review of piping fatigue [9] where it was determined that:
  - Although ANSI B31.1 and ASME Code, Section III, Class 1 piping rules are fundamentally different, experience in operating plants has shown that piping systems designed to B31.1 are adequate.
  - The operation of B31.1 plants is also not different from that of plants designed to ASME Code, Section III.

Providing a robust solution without resorting to a complete reanalysis requires a new approach. EPRI has developed a process for screening the primary coolant-wetted Class 1 reactor coolant pressure boundary fatigue-sensitive components in a plant by ranking them in terms of $U_{en}$ and then determining a set of *Sentinel Locations* such that every plant component is covered by one or more Sentinel Locations.

1. A *Sentinel Location* is a specific location in a piping system or component that serves as a leading indicator for EAF damage accumulation. It serves by itself or as one of a small group of locations that bound other locations in a *Thermal Zone* for a given material of construction. These Sentinel Locations are expected to remain bounding as plant transients occur in plant life. Thus, monitoring of the Sentinel Locations would maintain assurance that the system or component remains bounded throughout its operating life and can be used to trigger any necessary actions with sufficient time to provide appropriate remedies for the system or component. Sentinel Locations should be periodically reevaluated as plant transients accumulate to ensure that they continue to serve the sentinel function.
2. *Thermal Zone* is defined and discussed in Section 3.

EPRI Technical Report 1022873, Improved Basis and Requirements for Break Location Postulation, October, 2011 [10] concludes that fatigue usage calculated using ASME fatigue curves, including environmental effects, does not directly correlate with the probability of failure. In addition, for the range of components evaluated, a $U_{en}$ of less than 1.0 has an insignificant safety impact, since the estimated core damage frequency was in all cases below 1E-6. Thus, the consideration of special EAF limit for the purposes of screening or ranking is not considered in this report.

# Section 3: Process Outline and Technical Basis

This section provides an outline and technical basis of the screening process for reviewing plant components susceptible to fatigue, categorizing them into groups, and identifying one or more *Sentinel Locations* for each group that can be analyzed and monitored for EAF usage. In this context, a Sentinel Location is a location in a plant system that is expected to accumulate more EAF usage than other locations in that system.

The idea of Sentinel Locations extends the basic approach that was used in NUREG/CR-6260 [4]. It retains the core concept of analyzing a few challenging locations to represent the entire plant, but it adds a semi-quantitative ranking system to demonstrate that each plant component is represented by at least one Sentinel Location.

It is necessary to evaluate components and/or locations in a component on a uniform common basis to accomplish valid ranking and identification of Sentinel Locations in each Thermal Zone. Plants with fatigue design bases can have:

- Sets of components evaluated to a reduced, "bundled" set of plant transients and/or a mixture of bundled and unbundled transients.
- Components or locations in components evaluated to additional refined analyses while other components or locations are not.

To assure uniform determination of relative fatigue accumulation, these differences must be accounted for or eliminated. The screening processes described in this report are designed to make this common basis determination.

The reader is reminded that this report is NOT provided as a Quality Assured document. Application of the processes described will require appropriate review and quality dedication on a site-specific basis.

## Outline of Process Steps

This screening process consists of four stages: data collection, determination of Thermal Zones, evaluation of locations, and ranking and identification of Sentinel Locations. Each of these stages is explained in the sections that follow.

1. Data Collection

Data collection is necessary to equip the user to perform screening evaluations. Input data such as component geometry and material properties, plant transient characteristics and projections of plant transients for the licensed operating period are required to compute relative stress, CUF and $U_{en}$ values for evaluated components.

2. Determination of Thermal Zones

   A *Thermal Zone* is defined as a collection of piping and/or vessel components which undergo essentially the same group of thermal and pressure transients during plant operations. Thermal Zones are determined on the basis of common plant transients during plant operation. Components are assigned to appropriate Thermal Zones and evaluated as a group. This allows definitive rankings to be determined.

3. Evaluation of Locations

   Locations in plant components are evaluated to establish *relative* stress, CUF and EAF values. In keeping with the principle that no new detailed stress and fatigue analyses are required to perform this screening, locations in each Thermal Zone will be evaluated with a common basis approach. This common basis approach mitigates the skewing effects of refined analyses (such as elastic-plastic analysis) for selected components. The purpose of ranking on a common basis is to assure that the most highly stressed and cycled locations in each Thermal Zone are identified as leading indicators of fatigue damage for the Thermal Zone.

   Two analytical evaluation procedures are developed to aid in the evaluation process; one to perform *Common Basis Stress Evaluations* and the other to perform *$F_{en}$ Estimation Evaluations*. A brief description of the analytical flow follows. Detailed descriptions of the procedures are provided in Section 4.

   The evaluation incorporated in the *Common Basis Stress Evaluation Procedure* is based on the rules of ASME NB-3600 modified to address a screening evaluation for relative ranking of locations. Rationales for this approach are that:

   - The majority of the components in the screening population are piping components for which the rules of NB-3600 are appropriate.
   - The NB-3600 equations are explicitly defined and require minimal analyst interpretation so that they can be easily included in a spreadsheet.
   - The NB-3600 rules are representative of the more general rules of ASME NB-3200 design by analysis, which are appropriate for all plant components.

   For cases where a component or location has no explicit design fatigue analysis available or where is it desired to put components with an analysis on a common basis, the user may need to estimate a common basis CUF (i.e., an estimated CUF value that is determined on the same transient basis with all other locations in the system). The *Common Basis Stress Evaluation* is used to

perform the following stress computations to determine the common basis CUF:

- Through-wall transient thermal stresses are computed for leading transients. Transients with thermal shocks are found to be the leading fatigue usage contributor in component stress analyses.
- Piping moment range stresses and pressure stresses are extracted from the plant piping Class 1 stress report. Use of actual piping results avoids the use of piping umbrella loads and helps differentiate moment loadings for locations within a piping system.
- Peak Stresses at discontinuities are accounted for using SCF/FSRFs taken from the ASME Code.

The *$F_{en}$ Estimation Evaluation Procedure* is developed to estimate $F_{en}$ for locations in plant components on the basis of the relevant parameters – Dissolved Oxygen (DO), maximum temperature and estimated tensile strain rate – of the leading transient. This procedure can be used as a source of estimated $F_{en}$ values for plants both with and without fatigue design analyses.

The Procedures are developed to:

- Use for plants with and without explicit fatigue design analyses available.
- Use with design transients or actual transients (as long as they are consistently applied).
- Use with design numbers of transients or licensed operating period (e.g., 60-year) projected numbers of transients.

4. Ranking and Identification of Sentinel Locations

   An estimated $U_{en}$ ($U_{en}$*)is determined by multiplying the common basis CUF by the estimated $F_{en}$. Those locations within each group with the highest estimated $U_{en}$ are reviewed to determine one or more Sentinel Locations. These leading locations for environmental fatigue accumulation from ongoing plant transients should be managed by the plant FMP to assure adequate margin for fatigue considering EAF.

The end result of this screening process is a listing of fatigue-sensitive plant components, organized into groups, ranked by $U_{en}$* severity, with at least one Sentinel Location identified for each group of components. Thus, a Sentinel Location may represent a number of other locations. This information can be used to augment the existing plant Fatigue Management Program to assure that bounding EAF locations in each Thermal Zone will be monitored and serve as early warning beacons and action triggers for components which might approach $U_{en}$ = 1.0.

## Process Development Assumptions and Characteristics

Several assumptions are inherent in the process developed in this report, as provided in the following list:

- Estimated $F_{en}$ method is sufficient for a screening process; it is not intended to qualify components using this process.

Several characteristics of the process are important.

- Common stress evaluation basis, consistent S-N curves should be used.
- Linear elastic stress analysis and superposition of stress contributions are used.
- The $F_{en}$ factor is applied only for increasingly tensile portions of transients, based on the guidance of MRP-47, Revision 1 [11].
- The $K_e$ factor is included in both the determination of strain range and in the estimated strain rate determination. This approach is recommended in proposed ASME-Code Case N-792-1.
- Thermal Zones are employed to provide consistency in development of estimated $F_{en}$ values and common basis stress approximations.
- Common analytical basis (un-bundled transients) is used to put analyses in a Thermal Zone on the same transient basis.
- Calculated plant piping loads and stresses are used instead of piping attachment point umbrella loads.
- Design severity transients (can use actual severity, if available and consistently applied) are used.
- Geometric factors are applied to stress terms.
- Materials of construction are evaluated together as a group in each Thermal Zone.

### Determining Thermal Zones

For the purpose of this process, a *Thermal Zone* is defined as a collection of piping and/or vessel components which undergoes essentially the same group of thermal and pressure transients during plant operations. The idea of Thermal Zones is similar to the way that design specifications separate components into groups with common transient definitions. Within a Thermal Zone, thermal shocks and thermal bending stresses vary depending only on the materials, geometry, and location of the component in the system. Therefore it is possible to rank the locations in a Thermal Zone for fatigue independent of the transients that actually occur.

An important step in this process is dividing plant systems into Thermal Zones. For each system, one or more Thermal Zones must be determined on the basis of similar thermal and pressure transients. Operating procedures, design specifications and piping isometric drawings are used to determine which components undergo essentially the same set of thermal and pressure transients in terms of the transient variation in temperature and pressure. Components in the same flow path or in the same sector of a vessel would be included in the same Thermal Zone. When performing this step, it is important to make the

Thermal Zones as inclusive as possible, to capture the largest number of components in the ranking. Some components may be considered to be part of two adjacent Thermal Zones.

For instance, the Class 1 portions of the Charging and Volume Control System (CVCS) in a PWR is comprised of piping that connects the Regenerative Heat Exchanger (RHX) to the cold leg charging nozzle(s) and the pressurizer spray system and from the letdown nozzle on a cold leg to the RHX. Considering one charging flow path from the RHX to one of the cold legs, components in that charging flow path experience essentially the same transients during operation, with only minor variations depending upon location in the flow path. This characteristic establishes these components as a Thermal Zone (see Section 4.2 for more detail of the process).

## Technical Basis for $F_{en}$ Estimation Evaluation

This section describes the technical basis and procedure developed to compute estimated $F_{en}$ ($F_{en}$*) and estimated $U_{en}$ ($U_{en}$*) values for locations in individual plant components where there is a DSR from which CUF values can be extracted.

For the purpose of this screening, the rules for calculating $F_{en}$ values may either be taken from (1) NUREG/CR-5704 [5] for stainless steel material, NUREG/CR-6583 [6] for carbon/low alloy steel material and NUREG/CR-6909 [7] for Ni-Cr-Fe material, or (2) from NUREG/CR-6909 [7] for all materials. It is noted that the substitution of the rules of NUREG/CR-6909 for stainless steel and carbon/low alloy steels causes the values of $U_{en}$ to differ from the values using NUREG/CR-5704 and NUREG/CR-6583.

These rules allow calculation of $F_{en}$ factors based on the material at the postulated failure location (SS, CS, LAS and Ni-Cr-Fe) and the following environmental parameters:

- Estimated strain rate ($\dot{\varepsilon}$) during the transients, in [%/sec].
- Concentration of dissolved oxygen (DO) in the water, in [ppm].
- Maximum fluid/metal temperature (T) during the transients, in [°C].

(Note: sulfur content of the metal (S) is also a factor for CS and LAS. However, this procedure will conservatively assume all CS/LAS components have the worst possible sulfur content.)

Since the procedure is developed as an aid to a screening evaluation for the purpose of relative ranking, the exact values of these parameters will not be calculated from qualified design input. Instead, estimated values will be determined based on knowledge of the operation of the various plant systems and components during both normal operation and the transient conditions as defined in the plant design specifications. Specifically:

- Any components which have no exposure to the "environment" (i.e., heated primary/secondary coolant water) will be assigned an $F_{en}$ value of 1.0. This includes components such as bolts and studs, and components where the critical location with respect to fatigue is on the outside surface (i.e., exposed to ambient air). These components will not rise to the $U_{en}$ necessary to be considered as a limiting location, and are not further evaluated in this report.
- Material identification for the component may be obtained from available drawings, such as flow diagrams, piping isometrics and material specifications.
- A qualitative estimate of the strain rate for the controlling fatigue transient(s) whose resultant stresses become increasingly tensile during the course of the transient will be determined, based on knowledge of the corresponding plant system. Each component will be identified with one of eight possible $\dot{\varepsilon}$ categories shown in Table 3-1. Transient pairs composed of seismic loadings will be assigned $F_{en}$ = 1.0.
- The effect of $K_e$ should be accounted for in the estimation of strain rate. This method is justified by additional study documented in the Background for Revision to CC N-792 that determined that it is acceptable to include the stress due to $K_e$ in the strain rate calculation.

*Table 3-1*
*Strain Rate Categories*

| Strain Rate Category | Estimated $\dot{\varepsilon}$ [%/sec] |
|---|---|
| Extreme | ≥ 5.0 |
| V.High | ~ 1.3 |
| High | ~ 0.33 |
| Mid-High | ~ 0.087 |
| Medium | ~ 0.023 |
| Low-Mid | ~ 0.0059 |
| Slow | ~ 0.0015 |
| V.Slow | ≤ 0.0004 |

Note: any components which have no exposure to the "environment" (i.e., heated primary coolant water) will be assigned an $F_{en}$ value of 1.0. This includes exterior locations and vessel head and manway studs, for example.

- An estimated DO value of "Low" (≤ 0.04 ppm) will be applied for all components exposed to reactor water for PWRs. This determination is based on the observation that for the entire history of most PWRs, the concentration of dissolved oxygen is maintained below 0.04 ppm at all times when water temperature is ≥ 150°C (302°F) (with rare exceptions). (Note: when water temperature is below 150°C, DO is no longer a factor in the value of $F_{en}$ for any of the materials considered in this procedure.) For BWRs, the DO values must be determined based on the procedural policies of the plant for water chemistry control. If plant reactor water or metal

surface DO data is known, the actual DO values may be used. For NUREG/CR-6909 applications, a consistent O' is used for austenitic stainless steel materials, which is invariant to DO level.

- An estimated upper-bound T value will be determined based on the collected design transients for the respective plant systems. (It will be converted to °C in the $F_{en}$ procedure as necessary.) If plant reactor water temperature data is known, the actual temperature values may be used.

For each component, this evaluation computes two hypothetical $F_{en}$ values, one using the estimated parameter values described above, and the second using the same estimated values for DO and T, but using the worst possible (i.e. most conservative) value for strain rate, $\dot{\varepsilon}$. These two computed values are averaged to produce an expected $F_{en}$ for each component. This two-part expected $F_{en}$ is based on experience with performing detailed $F_{en}$ analyses; in general, the estimated $F_{en}$ from a detailed analysis is close to the $F_{en}$ value computed for just the controlling transient pairs, but slightly higher due to contributions from the less-significant fatigue pairs. A simple average is judged to magnify the contributions of the less-significant transient pairs to yield a reasonably conservative value suitable for ranking without performing a detailed analysis. This method is considered to be a reasonable approach, but other methods for assigning $F_{en}$ values may be used, as determined by the user.

## Technical Basis for Common Basis Stress Evaluation

The *Common Basis Stress Evaluation Procedure* is used to compute an estimated CUF (U*), estimated $F_{en}$ ($F_{en}$*) and estimated $U_{en}$ ($U_{en}$*) values in components where:

- There is no Design Stress Report (DSR) (or the DSR does not include an explicit CUF calculation).
- There is a DSR with CUF values available, but there is reason to put the components on a common stress basis for comparison purposes.

### *Rationale for the Procedure*

Taking guidance from the EPRI Fatigue Management Handbook [12], formulas have been developed to compute stresses arising from maximum transient through-wall temperature distributions, axial temperature differences, thermal and mechanical bending stresses and geometric characteristics for piping and vessel components. These formulas ensure a common level of analysis so that the computed stresses are directly comparable between locations.

These formulas assume that stresses are linear elastic, and so may be combined using linear superposition. Non-linear plasticity effects are accounted for using elastic-plastic penalty factors ($K_e$) in accordance with ASME Code Subarticles NB-3200 and NB-3600 [1]. Use of linear elastic rules for computing CUF retains technical parity among the components in a Thermal Zone. By contrast, using elastic-plastic non-linear techniques in a fatigue analysis may significantly

reduce the computed CUF for that component, which would give it a much lower CUF than other locations with comparable fatigue duty.

The linear elastic stress state for a location may be computed as the linear summation of the individual stresses caused by various types of loads. Most pressure vessels and piping system components include stresses due to internal pressure, thermal (due to temperature distribution in the component), and boundary interface loads, such as forces and moments caused by thermal expansion, thermal stratification, anchor displacement, seismic movement, etc. Deadweight and residual stresses may be ignored, because they do not vary with time and therefore do not impact the computed stress range.

For a linear elastic stress analysis, stress contributions may be classified as one of two types:

1. Stresses due to loads, such as pressure, piping thermal expansion, etc. that are directly scalable to pertinent parameters (pressure, temperature, etc.), and
2. Time-dependent thermal stresses, which depend on the axial and radial temperature distributions in the component rather than any single instantaneous parameter.

Stress contributions of the second type depend on the temperature history and are typically calculated by a time integration of the product of a predetermined Green's function, or influence function, and the transient temperature data. Performing this integration is more complex than is desired for this screening process. Instead, an estimate of the maximum stress range during each significant thermal transient is computed, as described below. This estimate applies a uniform level of conservatism, and is sufficiently precise to determine a relative ranking among the components in a Thermal Zone.

The stress computation combines stresses from the following terms:

- Through-wall transient thermal stresses are computed using the graph shown in Figure 3-1. For each transient, two non-dimensional factors (k/hL) and ($kt_0/\rho c_p L^2$) are computed as entry into the curve for the determination of the normalized thermal peak stress.
- Piping moment range and pressure stresses are extracted from the plant piping Class 1 stress report. Umbrella loads (conservative loads assigned to the system to facilitate design of adjoining systems) are not recommended, as they don't inform the relative severity at different locations.
- Thermal stratification moment stresses are assumed to be negligible or included in the computed piping moment stress range.
- Seismic stresses.
- Peak Stresses at discontinuities are accounted for using appropriate SCFs.

*Actual values of these stresses may be used, if applied consistently within a Thermal Zone evaluation.*

The *Common Basis Stress Evaluation Procedure* is used to determine approximate stress ranges arising from pairs of selected significant transients, compute alternating stress values including simplified elastic-plastic ($K_e$) effects, and produce estimated incremental CUF ($U_{incr}$*) for input numbers of plant transients (either design numbers or projected numbers). These estimated incremental CUF ($U_{incr}$*) values if summed would produce the common basis CUF (U*). Estimated $F_{en}$ values ($F_{en}$*) are computed (using either the older or newer EAF rules), and multiplied by $U_{incr}$* to produce an estimated incremental $U_{en}$ ($U_{en\ incr}$*) for each transient pair. These $U_{en\ incr}$* values are summed over the significant transients to yield an estimated $U_{en}$* for that location.

## *Formulas and Equations*

The evaluation addresses the following equation.

$$S_{peak} = S_{press} + S_{mom} + S_{trans} + S_{ts}, \qquad \textit{Equation 3-1}$$

*(valid for cylindrical or flat plate components)*

Where:

$S_{peak}$ = total peak stress range (including $K_e$ factor as appropriate)

$S_{press}$ = peak stress range from pressure

$S_{mom}$ = peak stress range from moments (includes seismic loads and stratification loads)

$S_{trans}$ = peak stress range (axial) from gross structural or material discontinuity

$S_{ts}$ = peak stress range from through-wall thermal gradient

In this process, stresses, moments and thermal transients are input and several thermal-hydraulic characteristics are computed. Stresses are addressed as follows:

For $S_{press}$ and $S_{mom}$: extract values for transients from the Class 1 stress reports

For $S_{trans}$: ignore for screening purposes or extract from the Class 1 stress reports

For $S_{ts}$: evaluate by determining the maximum through-wall thermal stress using the graphical relationship in Figure 3-1.

Inputs for this determination for a piping component are:

- Geometric: Piping Diameter (D), Wall thickness (t), Moment SCF ($K_2$) and Peak SCF ($K_3$).
- Materials: $S_m$ ,Thermal Conductivity (k), Elastic Modulus (E), Coefficient of Thermal Expansion (α), Density (ρ), Heat Capacity ($C_p$).

- Thermal-hydraulic: Flow Rate (Q), Maximum Temperature $T_{max}$, Transient Temperature Change ($\Delta T$), Transient Time ($t_0$), Transient Pressure Change ($\Delta P$).
- Computed values are Heat Transfer Coefficient (h), k/ht, $kt_0/\rho c_p t^2$, normalized thermal stress ($\rho_{max}$ / [(E$\alpha\Delta$T)/(1-ν)]) and Transformed Transient Strain Rate ( $\dot{\varepsilon}^*$ ). The equations in Figure 3-1 use the terminology of "L" for wall thickness, instead of "t".

The heat transfer coefficient, h, must be computed for each of the plant transients (up and down portions), as follows:

$h = 0.023 * (\rho VD/\nu)^{0.3} Pr^{0.4} (k/D)$ (valid for turbulent flow conditions – Re > 2000)

Rearranging to define in terms of Q and D and accounting for units:

$h = 12.92874 * (Pr^{0.4} k/\nu^{0.8}) * (Q^{0.8} / D^{1.8})$

$h = \Phi * (Q^{0.8} / D^{1.8})$

Φ has been curve fit for use in spreadsheets as (valid for temperatures from 0°C (32°F) to 315.6°C (600°F)):

$\Phi = 56.45 + 1.270 * T - 2.927 \times 10^{-3} * T^2 + 3.952 \times 10^{-6} * T^3 - 2.654 \times 10^{-9} * T^4$

where:

h = heat transfer coefficient (Btu/hr-ft$^2$-°F)

ρ = density (lbf/ft$^3$)

V = velocity (ft/sec)

D = inside diameter (inches)

ν = kinematic viscosity (ft$^2$/hr)

Pr = Prandtl Number (dimensionless)

$c_p$ = specific heat (lbf/ft$^3$-°F)

k = thermal conductivity (Btu/hr-ft-°F)

$S_{ts}$ is computed by computing two non-dimensional factors (k/ht and $kt_0/\rho c_p t^2$) and using those factors to determine the value of maximum thermal stress as a percentage of the maximum thermal stress from a steep temperature step with infinite heat transfer from the graph on Figure 3-1.

1. $S_{press}$ is computed or extracted from the plant's Class 1 stress reports.
2. $S_{mom}$ is developed by extracting thermal and mechanical moments for transients from the plant's Class 1 stress reports.

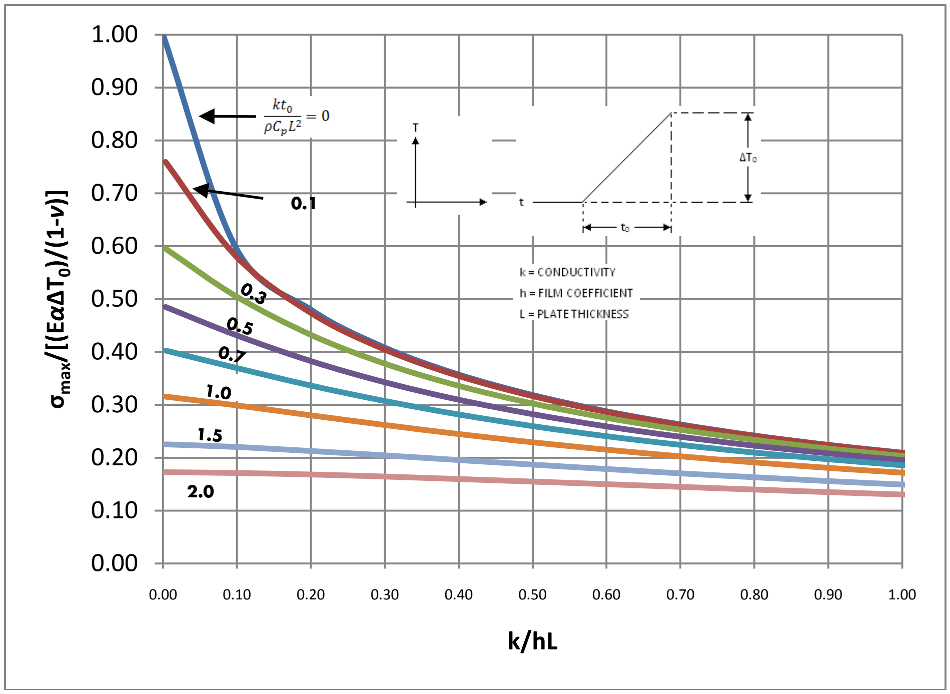


*Figure 3-1*
*Determination of Transient Stresses for Ramp Transients*

## Computations and Resulting Outputs from Common Basis Stress Evaluation Procedure

For each selected component location, the procedure is used to:

- Define the leading transients for evaluation.
- Compute stresses for each of the leading transients, accounting for FSRFs.
- Pair tensile and compressive stress pairs of leading transients.
- Compute transient pair $S_a$ from $S_{p-p}$, $S_m$ and $K_e$.
- Compute transient pair $U_{incr}$* from $N_{allow}$ and $n_{events}$. Note: $n_{events}$ is based on the design basis number of events; it may also be of interest to evaluate based on the projected number of events (and such a comparison is shown in Table 5-3), however, the recommended approach for ranking is to use the design basis number of events.
- Compute transient pair $U_{en\ incr}$* from $U_{incr}$*and $F_{en}$*
- Sum all transient pair $U_{en\ incr}$* into component $U_{en}$*.

### ***Comparison of Common Basis Stress Evaluation Screening Rules to NB-3600 Evaluation Process***

An evaluation of the analysis procedure presented in NB-3600 for the evaluation of piping components is made to compare the elements of the screening process to the ASME rules.

A comparison of the Common Basis Stress Evaluation process is made to the basis equations of NB-3600 (provided in Appendix B as quoted in Section 2 of [4]). For the express purpose of screening, certain judgments are made about the equations in NB-3653.

For the purpose of this fatigue ranking, it is assumed that the basic primary stress limit is assumed to have been addressed in the component design and will not be evaluated for screening.

NB-3653.1 Equation 10 [1] is accommodated in the screening by manual input of a "PPSMAX" value for each component, based on previous analysis and judgment of the maximum ratio of Primary + Secondary / Primary + Secondary + Peak Stresses (defined as PPSMAX). The peak stress range will be multiplied by the PPSMAX value to supply the secondary stress range used to compute the elastic-plastic penalty factor, $K_e$ in NB-3653.1 Equation 14 [1].

Judgment is used to determine the comparison to the terms in the NB-3653.1 fatigue Equation 11. The prevailing fatigue-driving loading in nuclear plant Class 1 piping is thermal shock, where cold water enters a hot pipe in a step change or short time duration. Pressure and moment loadings play an important secondary role and must also be considered. Common Basis Stress Evaluation Procedure addresses the following terms of Equation 11 [Eq. 2-7 of Ref. 4]:

$$S_p = K_1 C_1 \frac{P_o D_o}{2t} + K_2 C_2 \frac{D_o}{2I} M_i + K_3 C_3 E_{ab} \times \left| a_a T_a - a_b T_b \right| + \frac{1}{2(1-\nu)} K_3 E a \left| \Delta T_1 \right| + \frac{1}{1-\nu} E a \left| \Delta T_2 \right|$$

*Equation 3-2*

The first three terms of Equation 11 are represented by the first three terms of the equation below. These terms generally do not provide the primary loadings for components undergoing thermal shock and experiencing relatively high CUF accumulations. The fourth term of Equation 11 is represented by the last term of the equation below. The fifth term of Equation 11 is generally a small contributor to CUF compared to the fourth term and requires more detailed analysis to define. For the purposes of a screening evaluation where thermal shock plays a large role, it can be ignored.

$$S_{peak} = S_{press} + S_{mom} + S_{trans} + S_{ts}$$

*Equation 3-3*

Stresses are addressed in Common Basis Stress Evaluation Procedure as follows:

For $S_{press}$ and $S_{mom}$: extract values for transients from the Class 1 stress reports

For $S_{trans}$: ignore for screening purposes

For $S_{ts}$: evaluate

NB-3653.1 Equations 12 and 13 [1] are not evaluated in the screening process because it is assumed that they met their limits in the piping component design and will not change in the course of screening.

NB-3653.1 Equation 14 [1] is used to compute CUF in the screening process,

A thermal ratcheting evaluation is not performed in the screening process because it is assumed that limit was satisfied in the piping component design and will not change in the course of screening.

### *Limitations and Assumptions of the Process*

Stresses caused by complex loading, such as thermal stratification, are not used in the Common Basis Stress Evaluation process. It is typically not practical to compute stratification stresses using a basic methodology. However, for components subjected to this type of loading, fatigue calculations are expected to have been performed already. Such is the case, for example, with PWR surge lines.

Likewise, axial thermal gradient stresses produced by geometry or material transitions are also not considered in this process. Branch nozzles without thermal sleeves are commonly subject to stresses caused by axial thermal gradients. Such loading may be attributed to the injection of colder fluid into a hot header, giving rise to significant thermal stresses of a steady state nature near the nozzle corner. Sophisticated fatigue analyses are typically employed to disposition these types of components, and many of them, such as the charging and safety injection nozzles, are the NUREG/CR-6260 locations (the GALL report requires evaluation of the 6260 locations at a minimum).

The Common Basis Stress Evaluation Procedure is strictly valid only for cylindrical or flat plate components where the simplified methodology is most appropriately applied.

# Section 4: The Screening Process

This section describes a screening process with step-by-step instructions for evaluating the Class 1 systems in a plant to produce a list of Sentinel Locations to include in the Fatigue Management Program. This process is depicted in Figure 4-1 and Figure 4-2. Thermal Zones are established in each Class 1 system. Candidate Sentinel Locations are determined in each Thermal Zone. Each of the materials of construction of the candidate Sentinel Locations are evaluated as a group using one of two procedures to produce an estimated $U_{en}$ ranking for comparison with estimated $U_{en}$ values determined for any NUREG/CR-6260 locations in the Thermal Zone. The NUREG/CR-6260 locations will be retained as Sentinel Locations in this process. Those candidate Sentinel Locations of all materials in all Thermal Zones in each system which meet certain grouping criteria are selected as Sentinel Locations.

The use of readily available data is encouraged for a screening evaluation, however a set of consistent design input is necessary. For example, within a given Thermal Zone screening evaluation, one of the methods should be used exclusively (design results with design numbers of cycles, design results with projected numbers of cycles, monitoring system results with design numbers of cycles, or monitoring system results with projected numbers of cycles).

## Detailed Screening Procedure

Refer to Figure 4-1 for this procedure.

### *Gather Required Inputs for all Systems Containing Class 1 Reactor Coolant Pressure Boundary Components*

Determine the following data for the components in each Class 1 system:

- Materials (austenitic stainless steel (SS), carbon steel (CS), low-alloy steel (LAS) or Inconel (Ni-Cr-Fe))
- Layout (connectivity and flow paths)
- Geometry (ID, OD, material/geometric discontinuities)
- Fatigue Strength Reduction Factors: $K_2$ and $K_3$
- CUFs (for those components with fatigue analyses)
- DO history (for the contained fluid)

- List of thermal and pressure transients
- For the leading transients (available from stress report or transient monitoring program)
  - $T_{max}$ and $T_{avg}$
  - Material properties ($S_m$, k, E, α, ρ, $c_p$) at average temperature
  - Thermal-hydraulic characteristics (Q (or V), $T_{max}$, $T_{min}$, $t_0$, ΔP)
  - Estimated strain rate $\dot{\varepsilon}$
  - Moment ranges for thermal and seismic loadings ($M_{thermal}$ and $M_{seismic}$)
  - Pressure stress range
  - (optional) projection of number of cycles ($n_{projected}$)
- This input data can be obtained from:
  - Vessel design drawings or P&ID and piping isometric drawings:
    - Vessel and pipe ID, OD and geometric factors
    - Connectivity and flow paths
  - Flow diagrams:
    - Connectivity and flow paths
  - Plant design specifications or design stress report (DSR):
    - Material properties at average temperature
    - Properties of each leading transient (see above)
  - Either DSR or FSAR or transient monitoring system:
    - Moments, stresses and loadings (see above)
    - Design CUF (U)
  - Plant operating logs and procedures:
    - DO history

### *Determine Thermal Zones for Each System*

For each system, one or more Thermal Zones must be determined on the basis of similar thermal and pressure transients. Operating procedures, design specifications and piping isometric drawings are used to determine which components undergo essentially the same set of thermal and pressure transients in terms of the transient variation in temperature and pressure. Components in the same flow path or in the same sector of a vessel would be included in the same Thermal Zone.

Starting with a location at the boundary of the candidate component, identify the thermal and pressure transients at that location, then expand the selection to any neighboring components which are subject to the same set of transients. At the point where the transients change, such as at a branch, pump, or temperature source, another boundary is established. When there are no additional locations left to evaluate, the Thermal Zone consists of all of the components within the boundaries.

For example, consider the Charging and Volume Control System (CVCS) in a PWR. It is comprised of piping that connects the Regenerative Heat Exchanger (RHX) to the cold leg charging nozzle(s) and the auxiliary spray tee on the pressurizer spray system and from the cold leg letdown nozzle to the RHX. There may be one or two charging nozzles in these systems (this example assumes two charging nozzles where flow is alternated between them every operating cycle).

The major transients in the CVCS are Loss of Letdown and Loss of Charging with prompt and delayed returns and charging flow adjustments. The transients are significantly different between the charging and auxiliary spray flow paths through the RHX tube side and the letdown flow path through the RHX shell side. Thus, the first division of potential Thermal Zones is between the charging flow path and the letdown flow path. Downstream components in the RHX tube side flow path will experience very similar thermal and pressure transients, only differing by which flow paths are active (which of the charging nozzles is in operation and whether or not auxiliary spray operations are active). The flow path for the active charging nozzle will experience essentially the same severity and duration of the transients from that charging nozzle to the RHX while the other charging nozzle flow path is stagnant and experiencing minimal transient behavior. The components in the active charging path comprise a single Thermal Zone. Since the currently inactive charging path will alternately be the active flow path, this flow path will also experience essentially the same transients from the charging nozzle to the RHX and would also qualify as a second Thermal Zone.

The flow path through the auxiliary spray piping will also experience unique transient depending upon the activity of that line and qualifies as a third Thermal Zone. The letdown piping is a single run of piping conducting the same letdown flow from the cold leg nozzle to the RHX and qualifies as the fourth Thermal Zone.

A determination must be made about the components on the boundaries of Thermal Zones. Locations on a boundary will typically experience transients from both Thermal Zones. A boundary component should generally be assigned to one of the two Thermal Zones. Focusing on the purpose of the screening process – to identify the leading fatigue accumulation locations – the important factor is whether the boundary location experiences all of the transients of the Thermal Zone. If it does, then it should be included in the Thermal Zone. In some cases, it may belong to both Thermal Zones, or it may become a Thermal Zone of its own.

For instance, the cold leg charging nozzles are at the boundary of the CVCS and the RCS cold leg piping. These nozzles could belong either to a unique Thermal Zone, the charging piping Thermal Zone or the cold leg Thermal Zone, or in several of them. The decision must be made about where they best fit. The charging nozzles experience all of the transients common to the rest of the charging line piping, but also exhibit additional transients caused by cold leg fluid reflood when charging flow is terminated. Due to the presence of additional

transients at the charging nozzle compared to the remainder of the charging piping, the charging nozzles should be included in the Thermal Zone with the associated charging line piping, instead of in a unique Thermal Zone. The charging nozzles do experience all of the global transients in the RCS cold leg piping, but review of the design fatigue analysis shows the primary fatigue duty for them are CVCS transients rather than RCS transients. This fact leads to the decision not to include the charging nozzles in the RCS cold leg Thermal Zone.

This is expected to be the usual assignment of boundary components on the RCS. The RCS piping nozzles connecting to adjoining systems, such as the CVCS, Safety Injection, Residual Heat Removal or Shutdown Cooling, etc., will be expected to be included in the Thermal Zones of those adjoining systems.

Continuing the evaluation would lead to the identification of the following four Thermal Zones for the CVCS:

1. One charging nozzle connected to a cold leg and piping back to the RHX
2. The other charging nozzle connected to a cold leg and piping back to the RHX
3. Auxiliary spray piping from the main spray line back to the charging line
4. Letdown nozzle on cross-over leg and piping between RHX and cross-over leg

### *Identify Materials and Candidate Locations,*

For Each Thermal Zone in the system,

- Select candidate locations by material
- Identify thermal, pressure and seismic transients

Determine candidate locations by examining vessel nozzles and wall thickness transitions and piping nozzles, elbows, tees and wall transition changes. Look for the largest changes in wall transitions, nozzles connected to piping or components with largest temperature differences, etc. The NUREG/CR-6260 locations will be included as candidate Sentinel Locations, evaluated on a common stress basis, and retained as Sentinel Locations in this process.

Identify the major thermal and pressure transients. Among these will be transients with a large change of value over a short time duration (e.g., step temperature shocks or rapid pressure drops).

### *Calculate $U_{en}$* for Each Candidate Location*

For each candidate location in the Thermal Zone

- Determine if a fatigue table exists from design stress reports
  - If YES
    - Do CUFs fit Common Basis model?

  - If YES, Go to $U_{en}$ Estimation Evaluation Procedure (A)
  - If NO, Go to Common Basis Stress Evaluation Procedure (B)
- If NO
  - Go to Common Basis Stress Evaluation Procedure (B)

The common basis concept is needed so that candidate Sentinel Locations are not disproportionately promoted or demoted due to assumptions of the analysis. This happens because many calculations use transient lumping and other strategies to reduce computational complexity. It is important to apply the same level of rigor in transient definition across all locations to assure proper relative ranking of fatigue results.

For CUF locations, a common basis determination is made by examining the fatigue table for each component. Examine the load set pairs with the largest $U_{incr}$ that account for at least 75% of the total CUF. If those load set pairs do not include any lumped transients, then the CUF satisfies the common basis and Part A below can be used. Otherwise, a common basis CUF needs to be calculated as part of the evaluation (Part B should be used).

For locations without a CUF, follow the procedure in Part B.

## $U_{en}$ Estimation Evaluation Procedure

The *$F_{en}$ Estimation Evaluation Procedure* (included below) is used to calculate an estimated $F_{en}$ ($F_{en}$*) and a final step is added to calculate an estimated $U_{en}$ ($U_{en}$*) for each evaluated location.

The *$U_{en}$ Estimation Evaluation Procedure* is comprised of the following activities:

1. For each location, the user:
   - Inputs the material type, DO, estimated strain rate $\dot{\varepsilon}$ and $T_{max}$ for the leading transient (from stress report or transient monitoring system)
   - Enters design (or computed) CUF value for the location
2. For each location, the procedure is used to compute:
   - Estimated $F_{en}$ value ($F_{en}$*)
   - Estimated $U_{en}$ value ($U_{en}$*)

## *$F_{en}$ Formulations*

The *$F_{en}$ Estimation Evaluation Procedure* allows the user to select from the $F_{en}$ formulations in NUREG/CR-5704 [5] and NUREG/CR-6583 [6], or NUREG/CR-6909 [7]. The details of these formulations are given in Appendix A. The user must select whether to use the older guidance [5 and 6] or the newer guidance [7]. When the old guidances are used, $F_{en}$ formulations for austenitic stainless steel materials are conservatively used for nickel alloy materials.

### *Determine Input Values*

For each plant location, determine the following five variables:

- ***CUF***: The common basis CUF
- ***Material Type***: Either **SS**, **LAS**, **CS**, or **Ni-Cr-Fe**
- ***DO***: Either **Low**, **Med**, or **High**
- $\boldsymbol{T_{max}}$: The maximum fluid temp to which the component is exposed, in [°F]—it will be converted to °C in the $F_{en}$ calculation. The value of the average temperature of each load pair ($\boldsymbol{T_{ave}}$) is used instead of $\boldsymbol{T_{max}}$ if the rules in [7] are used.
- ***Strain Rate***: A qualitative estimate of the strain rate ( $\dot{\varepsilon}$ ) for the typical or controlling fatigue transient(s), based on knowledge of the corresponding plant system. The strain rate must be specified as one of eight possible categories listed in Table 3-1.

### $F_{en}$ Estimation Evaluation Procedure

**Note**: When this procedure is applied to Part A, the Strain Rate defined above is used and only one $F_{en}$* is calculated for the location. When this procedure is applied to Part B, it is performed separately for each transient pair and each transient pair has a different $F_{en}$*. This is illustrated in Table 5-3.

1. Compute an Expected $F_{en}$ for each location as

$$F_{en} = \exp\left(A - (B \times S^{*}) * T^{*} O^{*} \dot{\varepsilon}^{*}\right) \qquad \textit{Equation 4-1}$$

where the constants ($A$, $B$) and the transformed parameters (T*, O*, etc.) are defined according to the specific NUREG guidance document; see Appendix A for details.

2. Compute Maximum $F_{en}$ for each location. Calculate as above, but use the most conservative (lower bound) strain rate for the ***Material Type*** instead of the ***Strain Rate*** value determined above.
3. Compute Estimated $F_{en}$ as: $F_{en}$* = (Expected $F_{en}$ + Maximum $F_{en}$)/2.

### *Final Step for $U_{en}$ Estimation Evaluation Procedure*

Compute Estimated $U_{en}$* for each location as: $U_{en}$* = ($F_{en}$*) × (CUF).

### Common Basis Stress Evaluation Procedure

Refer to Figure 4-2 for this procedure.

The *Common Basis Stress Evaluation Procedure* is used to calculate an estimated CUF (U*), an estimated $F_{en}$ ($F_{en}$*) and an estimated $U_{en}$ ($U_{en}$*) for each evaluated location.

The *Common Basis Stress Evaluation Procedure* consists of the following activities:

1. For each location, the user:
   - Inputs the material type and geometric properties, thermal-hydraulic characteristics, DO, estimated strain rate $\dot{\varepsilon}$ and $T_{max}$ for the leading transient (judgment and/or evaluation of stress report)
2. For each location, the procedure is used to compute:
   - Estimated U value (U*)
   - Estimated $F_{en}$ value ($F_{en}$*)
   - Estimated $U_{en}$ value ($U_{en}$*)

### *Determine Input Values*

1. For each plant location, determine the following variables:
   - Material Type: Either **SS**, **LAS**, **CS**, or **Ni-Cr-Fe**
   - $DO_{max}$,
   - $M_i$, Z, $(T_a\text{-}T_b)$, $M_{strat}$: the moment ranges, moment of inertia, axial temperature difference, and moment stress from stratification (for piping locations only), taken from the DSR.
   - Geometric values [material type, ID, OD, $C_1$, $C_2$, $K_2$, $K_3$]
   - Material properties at average transient temperature [k, E, α, $c_p$, ρ, ν, $S_m$]
   - Major transients in terms of fluid velocity (V), $T_{init}$, $T_{final}$, $t_o$, $P_{min}$, $P_{max}$, $n_{projected}$
2. Calculate several dependent parameters:

   h = convective heat transfer coefficient (use the formulas in Section 3)
   t = wall thickness = (OD – ID)/2
   $\Delta T_o$ = temperature change = $(T_{init} - T_{final})$
   $\dot{\varepsilon}$ = estimated strain rate = $\Delta T_o / t_o$ (determined as shown in Figure 3-1)
   R = (OD + ID) / 2

### *Perform Stress Evaluation*

1. Create load set pairs from the leading transients (or up-down pairs from single transients) using:
   - Extract $S_{press}$ as the maximum pressure stress range for the transient pair (i.e., $C_1(P_o D_o/2t)$) from the component's DSR ($P_o$ defined as the range of pressure for the transient pair and $D_o$ is OD)
   - Determine $S_{mom}$ as the maximum thermal and mechanical moment stress for the transient pair – ( i.e., $C_2(D_o/2I)M_i$) from the component's DSR ($M_i$ is the range of moments for the transient pair and $D_o$ is OD).
   - Determine $S_{ts}$ for each paired transient. Compute the two non-dimensional factors (k/ht and $kt_0/\rho c_p t^2$), then using Figure 3-1 to determine a value for $\sigma_{max}/(E\alpha\, \Delta T_0/(1\text{-}\nu))$. Note: the equations in Figure 3-1 use the letter "L" for wall thickness, instead of "t". Multiply that value by $E\alpha\, \Delta T_0/(1\text{-}\nu)$ to get $\sigma_{max} = S_{ts}$ for the transient. Note that $S_{ts}$ for step-up transients will be negative (less than zero), while $S_{ts}$ for step-down transients will be positive.

2. Compute $S_{p-p}$ for each transient pair as: $S_{p-p} = K_1S_{press} + K_2S_{mom} + K_3(S_{ts,max} - S_{ts,min})$
3. Determine a PPSMAX (based on experience with fatigue analyses of the component)
4. Compute $K_e$ for each transient pair using the rules in NB-3653.6, with $S_n$ = PPSMAX × $S_{p-p}$ as: $K_e = 1.0 + [(1-n)/n(m-1)] (S_n / 3S_m-1)$ within the bounds of $3S_m < S_n < 3mS_m$
5. Compute the alternating stress amplitude for each pair as: $S_a = S_{p-p}/2 \times K_e$.
6. Compute $N_{allow}$ for each pair using the appropriate fatigue curve from either NUREG/CR-6909 or the ASME Code.

### *Perform Fatigue Evaluation*

1. Compute $U_{incr}^*$ ($n_{events}$ / $N_{allow}$) for each transient pair. The total common-basis CUF (U*) is equal to the sum of these values for all transient pairs. Note: $n_{events}$ is based on the design basis number of events, it may also be of interest to evaluate based on the projected number of events (and such a comparison is shown in Table 5-3), however, the recommended approach for ranking is to use the design basis number of events
2. Determine EAF inputs for appropriate material and estimate $F_{en}$ for each transient pair. Use $F_{en}$ Estimation Evaluation Procedure described in Part A using the input gathered above to compute $F_{en}^*$ for each transient pair.
3. Compute $U_{en\ incr}^*$ for each transient pair as: $U_{en\ incr}^* = F_{en}^* \times U_{incr}^*$
4. Sum all $U_{en\ incr}^*$ to determine $U_{en}^*$ for each component

Repeat the process for the next candidate location of that material in the Thermal Zone in the system. When $U_{en}^*$ has been determined for all candidate locations of that material in the Thermal Zone in the system, go to the next step of ranking the candidate locations.

### RANKING and SENTINEL LOCATION IDENTIFICATION for Each Material in Each Thermal Zone

1. Identify all NUREG/CR-6260 locations as official Sentinel Locations for that material in that Thermal Zone and denote their $U_{en}^*$ values as $U_{en\ 6260}^*$. Sort all remaining locations by $U_{en}^*$ from highest to lowest value.
2. Identify candidate Sentinel Locations for final consideration.
   - Is $U_{en\ max}^* \geq 1.0$
     - If NO, evaluate candidate Sentinel Location $U_{en\ max}^*$ in step 3, remove all other locations from further consideration.
     - If YES, Is $U_{en\ max}^* > 2 \times U_{en\ max-1}^*$?
       - If YES, evaluate candidate Sentinel Location $U_{en\ max}^*$ in step 3, remove all other locations from further consideration.

- If NO, are the top 3 ($U_{en\ max}$*, $U_{en\ max-1}$* and $U_{en\ max-2}$*) within 25% (i.e., ($U_{en\ max}$* – $U_{en\ max-2}$*) / $U_{en\ max}$* < 0.25)?
  - ❖ If NO, evaluate 2 candidate Sentinel Locations ($U_{en\ max}$* and $U_{en\ max-1}$*) in step 3, remove all other locations from further consideration.
  - ❖ If YES, evaluate 3 candidate Sentinel Locations ($U_{en\ max}$*, $U_{en\ max-1}$* and $U_{en\ max-2}$*) in step 3, remove all other locations from further consideration.

3. Consolidate candidate Sentinel Locations with any resident NUREG/CR-6260 $U_{en}$* ($U_{en\ 6260}$*) location(s)
   - Does the Thermal Zone include one or more NUREG/CR-6260 locations?
     - o If YES,
       - Is candidate Sentinel Location $U_{en}$* ≥ 0.5 x lowest $U_{en\ 6260}$* location?
         - ❖ If YES, candidate Sentinel Location is promoted to an official Sentinel Location
         - ❖ If NO, remove from candidate Sentinel Location list
     - o If NO, candidate Sentinel Location is promoted to an official Sentinel Location

### *Evaluate Next Candidate Location*

Repeat the process for the next candidate Sentinel Location of that material in the Thermal Zone in the system. When the ranking of all candidate Sentinel Locations of that material in the Thermal Zone have been determined, move to the next material in Thermal Zone.

### Evaluate Next Thermal Zone

When all candidate Sentinel Locations for all materials in the Thermal Zone in the system are evaluated and ranked, move to the next Thermal Zone and repeat the process.

### Evaluate Next System

When all candidate Sentinel Locations for all materials in all Thermal Zones in a system are evaluated and ranked, move to the next system and repeat the process.

### Compile Final List of Sentinel Locations

When all candidate Sentinel Locations for all materials in all Thermal Zones in all systems are evaluated and ranked, compile the list of Sentinel Locations for inclusion in the Fatigue Management Program.

### *Guidelines for Reducing Number of Sentinel Locations*

This screening and ranking process can produce a fairly large number of Sentinel Locations, with at least one Sentinel Location assigned for each material in each Thermal Zone in each system. With specific engineering justification, a specific Sentinel Location may be justified to bound one or more other Sentinel Locations and allow the bounded Sentinel Location(s) to be removed from the final list. Possible criteria that could be used to make these judgments and guidelines for their evaluation are included below:

Possible Criteria for Determination of Sentinel Location Boundedness:

- *One Thermal Zone can bound another Thermal Zone in a System*

  This circumstance could be achieved if within the same system, both the CUF and $F_{en}$ values for one Sentinel Location in one Thermal Zone are *each* higher than the CUF and $F_{en}$ values for the Sentinel Locations in other Thermal Zones. It is expected that the $U_{en}$ for the former Sentinel Location would be more than double the $U_{en}$ values of the other Sentinel Locations. The determination that this highest Sentinel Location of one material could bound the other locations could be justified on this basis.

- *One material in a Thermal Zone can bound other materials in the same Thermal Zone*

  This circumstance could be achieved if within the same Thermal Zone, both the CUF and $F_{en}$ values for one Sentinel Location composed of one material are *each* higher than the CUF and $F_{en}$ values for the Sentinel Locations composed for all other materials. It is expected that the $U_{en}$ for the former Sentinel Location would be more than double the $U_{en}$ values of the other Sentinel Locations. The determination that this highest Sentinel Location of one material could bound the other locations could be justified on this basis.

- *One material in a Thermal Zone can bound other materials in another Thermal Zone*

  This circumstance combines the guidelines of the two listed above and must satisfy both criteria listed.

- *A non-NUREG/CR-6260 location can bound and replace a NUREG/CR-6260 location*

  This is not allowed within the guidelines for NRC review of GALL, Revision 2 [8], which states in paragraph 4.3.3.1.3: "If an applicant has chosen to assess the impact of the reactor coolant environment on a sample of critical components, the reviewer verifies the following: 1. The critical components include a sample of high-fatigue usage locations. This sample is to include the locations identified in NUREG/CR-6260, *as a minimum* (emphasis added), and proposed additional locations based on plant specific considerations…"

- *A location with $U_{en}^*$ < 0.8 may be removed from the Sentinel Location list*

This judgment relies upon the relative value of the $U_{en}^*$ value of the Sentinel Location. Generally, if the Sentinel Location $U_{en}^*$ for the projected number of design cycles is low (e.g., $U_{en}^* < 0.8$), that Sentinel Location may be removed from the final list due to the small likelihood that it will be the leading Sentinel Location in a system. If, however, the Sentinel Location $U_{en}^*$ for the projected number of design cycles is fairly high (e.g., $U_{en}^* \geq 0.8$), the possibility exists that it could remain the Sentinel Location for its group and should be included in the monitoring program that ensures that it does not exceed a value of 1.0.

Evaluation of the guidelines listed above leads to the possibility that a Sentinel Location is not required for every Thermal Zone and that a Sentinel Location is not required for every system.

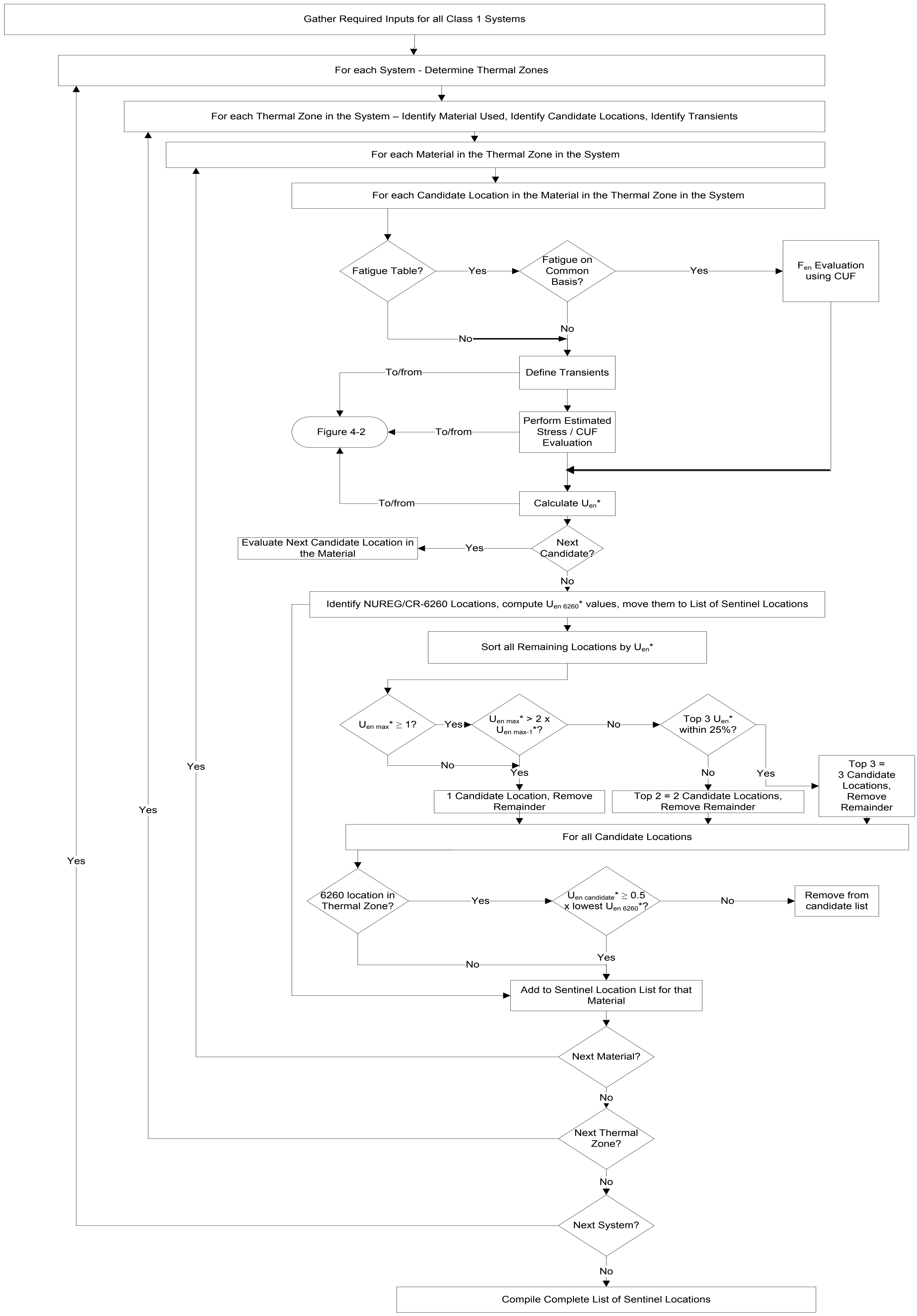


*Figure 4-1*
*Screening Flow Chart*

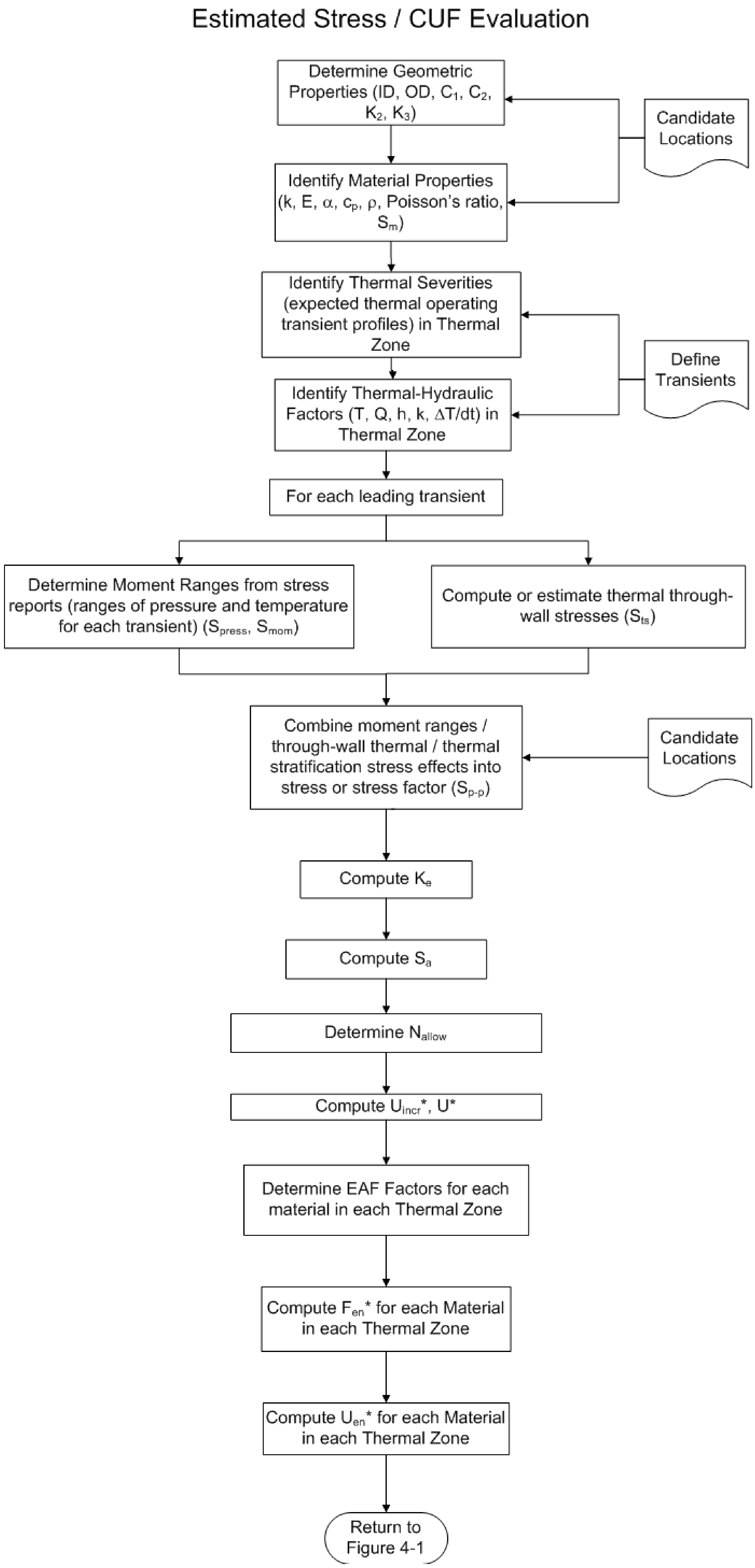


*Figure 4-2*
*Estimated Stress / CUF Evaluation*

# Section 5: Pilot Plant Application

Table 5-1 provides the list of candidate Sentinel Locations and their evaluation as initial and final Sentinel Locations using the $U_{en}$ Estimation Evaluation Procedure process of screening and the guidelines for reducing the number of Sentinel Locations. The $U_{en}$ Estimation Evaluation procedure employed calculated CUF values; the pilot plant was designed to ASME Section III requirements and plant components had these data available. Components to be evaluated are selected to include locations constructed of each material residing in each Thermal Zone.

For each row in the table, the “Component” is evaluated in this process to determine whether it will be a “Sentinel Location” for its Thermal Zone and material. For instance, the top line shows that RPV Outlet Nozzle is the location evaluated to be the Sentinel Location for the LAS RPV Nozzle Thermal Zone. Some of the entries represent multiple Sentinel Locations in a Thermal Zone, arising from the rules permitting establishment of two and three Sentinel Locations depending upon the relative $U_{en}$* values of the top three locations.

Details of this evaluation presented in Table 5-1 follow:

*In the Input section*: The Design Basis CUF, material of construction, expected DO, maximum temperature for the leading transients and strain-rate are tabulated in individual columns. Based on knowledge of fatigue analyses of many plant components, a strain-rate category was selected to represent each component. This selection was based on identifying the transients that would govern the CUF at the given component, and ranking them with respect to how quickly the maximum and minimum stress states are established. For instance, if the majority of CUF for this component would generally derived from very large temperature step changes, the “Extreme” strain rate category was used. Conversely, components governed by hours-long ramps of temperature and pressure would be assigned to the “V.Slow” category.

*In the $F_{en}$ Computation section*: The parameters T*, O*, S*, $\dot{\varepsilon}^*$, and the three values of $F_{en}$, are calculated as follows:

The Expected $F_{en}$ ($F_{en}$ Exp.) is calculated from the values of the T*, O*, S* and $\dot{\varepsilon}^*$ parameters using the the rules of NUREG/CR-5704 [5] for stainless steel material, NUREG/CR-6583 [6] carbon and low-alloy steel materials and NUREG/CR-6909 [7] for Ni-Cr-Fe steels material.

The Maximum $F_{en}$ ($F_{en}$ Max.) is calculated using the same formulas as above, except for using a bounding value for $\dot{\varepsilon}^*$, i.e., $\dot{\varepsilon}^* = \ln(0.001)$ for CS, LAS and SS, or $\dot{\varepsilon}^* = \ln(0.0004/5.0)$ for Ni-Cr-Fe.

The $F_{en}$ Average ($F_{en}^*$) is computed as [(Expected $F_{en}$) + (Maximum $F_{en}$)] / 2.

*In the Comparison section*: The Estimated $U_{en\ design}$ ($U_{en}^*$) is computed as the product of the $F_{en}$ Average ($F_{en}^*$) and the Design Basis CUF.

*In the Identification of Sentinel Locations section:*

Evaluations were made to determine whether each Evaluated Component qualified as an Initial Sentinel Location. This determination was made using the rules of the detailed screening process in Section 4. Descriptive reasons for the determination are provided in the column entitled *Criterion for Initial Sentinel Locations*. These descriptive reasons are related to the Ranking and Sentinel Location Identification logic presented in paragraph 5 of the Detailed Screening Procedure (page 4-8).

The *Initial Sentinel Location Count* column indexes those Evaluated Components that are established as Initial Sentinel Locations.

The last three columns present the results of using the Guidelines for Reducing the Number of Sentinel Locations (pages 4-9 and 4-10). The *Criterion for Final Sentinel Location* column provides a descriptive reason about why each Initial Sentinel Location is or is not a Final Sentinel Location. The *Final Sentinel Location Count* column indexes those Evaluated Components that are established as Final Sentinel Locations.

Table 5-2 provides a summary list of the results of Table 5-1. The last column shows the components that are identified as the final list of Sentinel Locations. These components represent the Sentinel Locations assigned for each material in each Thermal Zone of each system or vessel evaluated.

The sample results of an evaluation of the pilot PWR plant using the Common Basis Stress Evaluation Procedure are provided in Tables 5-3 through 5-5. Geometric and material properties for evaluated components are listed in Table 5-3. The input and computed thermal-hydraulic and stress characteristics of two transients are listed in Table 5-4. Table 5-5 shows the resulting combined stresses, $F_{en}^*$computation and $U_{en}^*$ values. Results of this procedure demonstrate concurrence with the conclusion of the $U_{en}$ Estimation Evaluation Procedure using design CUF input. Both methods indicate that the charging nozzle will bound the charging piping, although the $U_{en}^*$ values are not exactly the same. This table also includes a comparison of $U_{en}^*$ calculated with the design number of transients and $U_{en}^*$ calculated with the projected number of transients based on actual operating experience.

*Table 5-1*
*Results from Pilot PWR Plant Evaluation Using $F_{en}$ Estimation Evaluation Procedure*

| | | | Input | | | | | $F_{en}$ Computation | | | | | | | | | Comparison | Identification of Sentinel Locations | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **System or Vessel** | **Thermal Zone** | **Component** | **Design Basis CUF** | **Material** | **D0** | **Temp** | **Strain Rate** | **T*** | **O*** | **S*** | $\dot{\varepsilon}^*$ | **$F_{en}$ Exp.** | **Worst O*** | **Worst $\dot{\varepsilon}^*$** | **$F_{en}$ Max.** | **$F_{en}$ Average ($F_{en}$*)** | **$U_{en\ design}$ ($U_{en}$*)** | **Criterion for Initial Sentinel Locations** | **Initial Sentinel Location** | **Count** | **Criterion for Final Sentinel Location** | **Final Sentinel Location** | **Count** |
| Reactor Pressure Vessel | RPV Nozzle | RPV Outlet Nozzle | 0.1078 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.265 | NUREG/CR-6260 | Y | **1** | NUREG/CR-6260 | Y | **1** |
| | | RPV Inlet Nozzle | 0.0795 | LAS | Low | 558 | Slow | 14.36 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.195 | NUREG/CR-6260 | Y | **1** | NUREG/CR-6260 | Y | **1** |
| | RPV Upper Head | RPV Core Exit Thermocouple Nozzle Assembly Upper Nozzle Housing | 0.37 | SS | Low | 619 | Slow | 1 | 0.26 | 1 | -5.586 | 10.885 | 0.26 | -6.908 | 15.348 | 13.117 | 4.853 | $U_{en}$ > 1.0 | Y | **1** | Highest SS Uen in T.Z. | Y | **1** |
| | | RPV Core Exit Thermocouple Nozzle Assembly Head Port Adapter | 0.123 | SS | Low | 619 | Slow | 1 | 0.26 | 1 | -5.586 | 10.885 | 0.26 | -6.908 | 15.348 | 13.117 | 1.613 | < 50% of top location | N | **0** | | N | **0** |
| | | RPV Vessel Flange | 0.196 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.481 | $U_{en}$ < 1.0; sole location | Y | **1** | Uen < 0.8 | N | **0** |
| | | CRDM Housing | 0.1093 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.268 | | N | **0** | | N | **0** |
| | | RPV Head Flange | 0.0155 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.038 | | N | **0** | | N | **0** |
| | RPV Bottom Head | RPV Vessel Wall Transition | 0.0105 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.026 | $U_{en}$ < 1.0; sole location | Y | **1** | Uen < 0.8 | N | **0** |
| | | RPV Bottom Head-to-Shell Juncture | 0.007 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.017 | NUREG/CR-6260 | Y | **1** | NUREG/CR-6260 | Y | **1** |
| | | RPV Bottom Head Instrument Tubes (pos. 2) | 0.3184 | Ni-Cr-Fe | Low | 619 | Slow | 1 | 0.16 | 1 | -8.112 | 3.662 | 0.16 | -9.433 | 4.524 | 4.093 | 1.303 | $U_{en}$ > 1.0 | Y | **1** | Highest Ni-Cr-Fe Uen in T.Z. | Y | **1** |
| | | RPV Bottom Head Instrument Tubes (pos. 1) | 0.0014 | Ni-Cr-Fe | Low | 619 | Slow | 1 | 0.16 | 1 | -8.112 | 3.662 | 0.16 | -9.433 | 4.524 | 4.093 | 0.006 | < 50% of top location | N | **0** | | N | **0** |
| Pressurizer | Pressurizer Lower Head | Pressurizer Heater Penetration | 0.562 | SS | Low | 653 | Slow | 1 | 0.26 | 1 | -5.586 | 10.885 | 0.26 | -6.908 | 15.348 | 13.117 | 7.372 | $U_{en}$ > 1.0 | Y | **1** | Highest SS Uen in T.Z. | Y | **1** |
| | | Pressurizer Immersion Heater | 0.123 | SS | Low | 653 | Slow | 1 | 0.26 | 1 | -5.586 | 10.885 | 0.26 | -6.908 | 15.348 | 13.117 | 1.613 | < 50% of top location | N | **0** | | N | **0** |
| | | Pressurizer Heater Well | 0.128 | SS | Low | 653 | Low-Mid | 1 | 0.26 | 1 | -4.217 | 7.624 | 0.26 | -6.908 | 15.348 | 11.486 | 1.470 | < 50% of top location | N | **0** | | N | **0** |
| | | Pressurizer Thermowells | 0 | SS | Low | 653 | Slow | 1 | 0.26 | 1 | -5.586 | 10.885 | 0.26 | -6.908 | 15.348 | 13.117 | 0.000 | | N | **0** | | N | **0** |
| | | Pressurizer Shell at Support Lug | 0.992 | LAS | Low | 653 | Low-Mid | 19.7 | 0 | 0.015 | -5.133 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 2.435 | $U_{en}$ > 1.0 | Y | **1** | Highest LAS Uen in T.Z. | Y | **1** |
| | | Pressurizer Surge Nozzle | 0.963 | LAS | Low | 653 | V.Slow | 19.7 | 0 | 0.015 | -6.908 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 2.364 | > 50% of top location | Y | **1** | > 50% of top location | Y | **1** |
| | | Pressurizer Lower Head/Support Skirt | 0.734 | LAS | Low | 653 | Low-Mid | 19.7 | 0 | 0.015 | -5.133 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 1.802 | Within 25% of top location | Y | **1** | Within 25% of top location | Y | **1** |
| | | Pressurizer Instrument Nozzle | 0.236 | LAS | Low | 653 | Low-Mid | 19.7 | 0 | 0.015 | -5.133 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.579 | Not in top 3 locations | N | **0** | | N | **0** |
| | | Pressurizer Manway Pad | 0.141 | LAS | Low | 653 | Slow | 19.7 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.346 | | N | **0** | | N | **0** |
| | | Pressurizer Lower Head | 0.112 | LAS | Low | 653 | Low-Mid | 19.7 | 0 | 0.015 | -5.133 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.275 | | N | **0** | | N | **0** |
| | Pressurizer Spray Nozzle | Pressurizer Spray Nozzle | 0.411 | SS | Low | 653 | High | 1 | 0.26 | 1 | -0.192 | 2.678 | 0.26 | -6.908 | 15.348 | 9.013 | 3.704 | $U_{en}$ > 1.0 | Y | **1** | Highest SS Uen in T.Z. | Y | **1** |
| | Pressurizer SRV/PORV | 6-inch and 3-inch Pressurizer Safety and Relief Valve Piping | 0.975 | SS | Low | 653 | Low-Mid | 1 | 0.26 | 1 | -4.217 | 7.624 | 0.26 | -6.908 | 15.348 | 11.486 | 11.199 | $U_{en}$ > 1.0 | Y | **1** | Highest SS Uen in T.Z. | Y | **1** |
| | | Pressurizer 3-inch x 6-inch Power Operated Relief Valve Solenoid | 0.68 | SS | Low | 653 | Low-Mid | 1 | 0.26 | 1 | -4.217 | 7.624 | 0.26 | -6.908 | 15.348 | 11.486 | 7.811 | > 50% of top location | Y | **1** | > 50% of top location | Y | **1** |
| | | Pressurizer Safety/Relief Valve Nozzle | 0.169 | SS | Low | 653 | Low-Mid | 1 | 0.26 | 1 | -4.217 | 7.624 | 0.26 | -6.908 | 15.348 | 11.486 | 1.941 | Not within 25% of top value | N | **0** | | N | **0** |
| | | Pressurizer 3-inch x 6-inch Power Operated Relief Valve | 0.139 | SS | Low | 653 | Low-Mid | 1 | 0.26 | 1 | -4.217 | 7.624 | 0.26 | -6.908 | 15.348 | 11.486 | 1.597 | Not in top 3 locations | N | **0** | | N | **0** |
| | | Pressurizer 6-inch Pressurizer Safety Valve | 0.018 | SS | Low | 653 | Low-Mid | 1 | 0.26 | 1 | -4.217 | 7.624 | 0.26 | -6.908 | 15.348 | 11.486 | 0.207 | | N | **0** | | N | **0** |
| | Pressurizer Upper Head | Pressurizer Instrument Nozzle | 0.236 | SS | Low | 653 | Low-Mid | 1 | 0.26 | 1 | -4.217 | 7.624 | 0.26 | -6.908 | 15.348 | 11.486 | 2.711 | $U_{en}$ > 1.0 | Y | **1** | The SS 6-inch and 3-inch Pressurizer Safety and Relief Valve Piping bound the SS Pressurizer Instrument Nozzle | N | **0** |
| | | Pressurizer Thermowells | 0 | SS | Low | 653 | Slow | 1 | 0.26 | 1 | -5.586 | 10.885 | 0.26 | -6.908 | 15.348 | 13.117 | 0.000 | < 50% of top location | N | **0** | | N | **0** |

*Table 5-1 (continued)*
*Results from Pilot PWR Plant Evaluation Using $F_{en}$ Estimation Evaluation Procedure*

| | | | Input | | | | | $F_{en}$ Computation | | | | | | | | | Comparison | Identification of Sentinel Locations | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| System or Vessel | Thermal Zone | Component | Design Basis CUF | Material | D0 | Temp | Strain Rate | T* | O* | S* | $\dot{\varepsilon}^*$ | $F_{en}$ Exp. | Worst O* | Worst $\dot{\varepsilon}^*$ | $F_{en}$ Max. | $F_{en}$ Average ($F_{en}$*) | $U_{en\ design}$ ($U_{en}$*) | Criterion for Initial Sentinel Locations | Initial Sentinel Location | Count | Criterion for Final Sentinel Location | Final Sentinel Location | Count |
| | | Pressurizer Upper Head/Upper Shell | 0.928 | LAS | Low | 653 | Low-Mid | 19.7 | 0 | 0.015 | -5.133 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 2.278 | $U_{en}$ > 1.0 | Y | **1** | The SS 6-inch and 3-inch Pressurizer Safety and Relief Valve Piping bound the LAS Pressurizer Upper Head/Upper Shell. | N | **0** |
| Surge Piping | Surge Line Piping | 14-inch Hot Leg Surge Nozzle | 0.3 | SS | Low | 653 | Low-Mid | 1 | 0.26 | 1 | -4.217 | 7.624 | 0.26 | -6.908 | 15.348 | 11.486 | 3.446 | NUREG/CR-6260 | Y | **1** | NUREG/CR-6260 | Y | **1** |
| | | 14-inch Pressurizer Surge Line (includes Thermowell) | 0.099 | SS | Low | 653 | Slow | 1 | 0.26 | 1 | -5.586 | 10.885 | 0.26 | -6.908 | 15.348 | 13.117 | 1.299 | < 50% of top location | N | **0** | | N | **0** |
| Spray Piping | Spray Line Piping | 4-inch Spray Piping at Pressurizer Spray Nozzle | 0.84 | SS | Low | 558 | Mid-High | 0.813 | 0.26 | 1 | -1.526 | 3.516 | 0.26 | -6.908 | 10.964 | 7.240 | 6.082 | $U_{en}$ > 1.0 | Y | **1** | SS Pressurizer Spray Nozzle bounds SS Pressurizer Spray Piping | N | **0** |
| | | Auxiliary Spray Piping | 0.72 | SS | Low | 500 | Medium | 0.629 | 0.26 | 1 | -2.856 | 4.062 | 0.26 | -6.908 | 7.877 | 5.970 | 4.298 | Accounted for in CVCS T.Z. | N | **0** | | N | **0** |
| | | 4-inch Pressurizer Spray Piping | 0.25 | SS | Low | 558 | Mid-High | 0.813 | 0.26 | 1 | -1.526 | 3.516 | 0.26 | -6.908 | 10.964 | 7.240 | 1.810 | Not within 25% of top value | N | **0** | | N | **0** |
| | | Pressurizer Spray Line Thermowell | 0.021 | SS | Low | 558 | Mid-High | 0.813 | 0.26 | 1 | -1.526 | 3.516 | 0.26 | -6.908 | 10.964 | 7.240 | 0.152 | | N | **0** | | N | **0** |
| CVCS | Charging Loop 1 | CVCS 3-inch Cold Leg Loop 1 Normal Charging Nozzle | 0.9 | SS | Low | 558 | Mid-High | 0.813 | 0.26 | 1 | -1.526 | 3.516 | 0.26 | -6.908 | 10.964 | 7.240 | 6.516 | NUREG/CR-6260 | Y | **1** | NUREG/CR-6260 | Y | **1** |
| | | CVCS 3-inch Normal Charging Loop 1 Piping | 0.93 | SS | Low | 500 | Medium | 0.629 | 0.26 | 1 | -2.856 | 4.062 | 0.26 | -6.908 | 7.877 | 5.970 | 5.552 | > 50% of top location | Y | **1** | The CVCS 3-inch Cold Leg Loop 1 Normal Charging Nozzle bounds the CVCS 3-inch Normal Charging Loop 1 Piping. | N | **0** |
| | Charging Loop 4 | CVCS 3-inch Cold Leg Loop 4 Alternate Charging Nozzle | 0.9 | SS | Low | 558 | Mid-High | 0.813 | 0.26 | 1 | -1.526 | 3.516 | 0.26 | -6.908 | 10.964 | 7.240 | 6.516 | NUREG/CR-6260 | Y | **1** | NUREG/CR-6260 | Y | **1** |
| | | CVCS 3-inch Alternate Charging Loop 4 Piping | 0.93 | SS | Low | 500 | Medium | 0.629 | 0.26 | 1 | -2.856 | 4.062 | 0.26 | -6.908 | 7.877 | 5.970 | 5.552 | > 50% of top location | Y | **1** | The CVCS 3-inch Cold Leg Loop 4 Normal Charging Nozzle bounds the CVCS 3-inch Normal Charging Loop 4 Piping. | N | **0** |
| | Letdown Loop 3 | CVCS 3-inch Normal Letdown Piping, Crossover Loop 3 | 0.95 | SS | Low | 558 | Medium | 0.813 | 0.26 | 1 | -2.856 | 4.657 | 0.26 | -6.908 | 10.964 | 7.811 | 7.420 | $U_{en}$ > 1.0 | Y | **1** | The SS Charging Nozzle bounds the SS Letdown Nozzle on a Common Basis Stress Evaluation basis. | N | **0** |
| | | CVCS 3-inch Crossover Leg Loop 3 Normal Letdown Nozzle | 0.1 | SS | Low | 558 | Medium | 0.813 | 0.26 | 1 | -2.856 | 4.657 | 0.26 | -6.908 | 10.964 | 7.811 | 0.781 | < 50% of top location | N | **0** | | N | **0** |
| | Letdown Loop 4 | CVCS 2-inch Crossover Leg Loop 4 Excess Letdown Nozzle | 0.804 | SS | Low | 558 | Medium | 0.813 | 0.26 | 1 | -2.856 | 4.657 | 0.26 | -6.908 | 10.964 | 7.811 | 6.280 | $U_{en}$ > 1.0 | Y | **1** | The SS Charging Nozzle bounds the SS Letdown Nozzle on a Common Basis Stress Evaluation basis. | N | **0** |
| | | CVCS 2-inch Excess Letdown Piping, Crossover Loop 4 | 0.099 | SS | Low | 558 | Medium | 0.813 | 0.26 | 1 | -2.856 | 4.657 | 0.26 | -6.908 | 10.964 | 7.811 | 0.773 | < 50% of top location | N | **0** | | N | **0** |
| | Auxiliary Spray | Auxiliary Spray Piping | 0.72 | SS | Low | 500 | Medium | 0.629 | 0.26 | 1 | -2.856 | 4.062 | 0.26 | -6.908 | 7.877 | 5.970 | 4.298 | $U_{en}$ > 1.0 | Y | **1** | Highest SS $U_{en}$ in T.Z. | Y | **1** |
| | Drain | CVCS Drain Line, Loop 2 | 0.95 | SS | Low | 558 | Slow | 0.813 | 0.26 | 1 | -5.586 | 8.292 | 0.26 | -6.908 | 10.964 | 9.628 | 9.147 | $U_{en}$ > 1.0 | Y | **1** | The SS Charging Nozzle bounds the SS Drain Nozzle on the basis of Common Basis Stress Evaluation. | N | **0** |
| | | CVCS Drain Line, Loop 3 | 0.95 | SS | Low | 558 | Slow | 0.813 | 0.26 | 1 | -5.586 | 8.292 | 0.26 | -6.908 | 10.964 | 9.628 | 9.147 | > 50% of top location | Y | **1** | The SS Charging Nozzle bounds the SS Drain Nozzle on a Common Basis Stress Evaluation basis. | N | **0** |
| | | CVCS Drain Line, Loop 1 | 0.09 | SS | Low | 558 | Slow | 0.813 | 0.26 | 1 | -5.586 | 8.292 | 0.26 | -6.908 | 10.964 | 9.628 | 0.867 | Not within 25% of top value | N | **0** | | N | **0** |
| | | CVCS Drain Line, Loop 4 | 0.09 | SS | Low | 558 | Slow | 0.813 | 0.26 | 1 | -5.586 | 8.292 | 0.26 | -6.908 | 10.964 | 9.628 | 0.867 | Not within 25% of top value | N | **0** | | N | **0** |

*Table 5-1 (continued)*
*Results from Pilot PWR Plant Evaluation Using $F_{en}$ Estimation Evaluation Procedure*

| | | | Input | | | | | $F_{en}$ Computation | | | | | | | | | Comparison | Identification of Sentinel Locations | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| System or Vessel | Thermal Zone | Component | Design Basis CUF | Material | D0 | Temp | Strain Rate | T* | O* | S* | $\dot{\varepsilon}^*$ | $F_{en}$ Exp. | Worst O* | Worst $\dot{\varepsilon}^*$ | $F_{en}$ Max. | $F_{en}$ Average ($F_{en}$*) | $U_{en\ design}$ ($U_{en}$*) | Criterion for Initial Sentinel Locations | Initial Sentinel Location | Count | Criterion for Final Sentinel Location | Final Sentinel Location | Count |
| | Seal Water | CVCS 1-1/2-inch, 2-inch Seal Water Injection Loops 3 Piping | 0.114 | SS | Low | 558 | Slow | 0.813 | 0.26 | 1 | -5.586 | 8.292 | 0.26 | -6.908 | 10.964 | 9.628 | 1.098 | $U_{en}$ > 1.0 | Y | **1** | The SS Charging Nozzle, Letdown Nozzle and Drain Line bound the SS Seal Water Piping. | N | **0** |
| | | CVCS 1-1/2-inch, 2-inch Seal Water Injection Loops 4 Piping | 0.067 | SS | Low | 558 | Slow | 0.813 | 0.26 | 1 | -5.586 | 8.292 | 0.26 | -6.908 | 10.964 | 9.628 | 0.645 | > 50% of top location | Y | **1** | $U_{en}$ < 0.8 | N | **0** |
| | | CVCS 1-1/2-inch, 2-inch Seal Water Injection Loops 1, 2 Piping | 0.066 | SS | Low | 558 | Slow | 0.813 | 0.26 | 1 | -5.586 | 8.292 | 0.26 | -6.908 | 10.964 | 9.628 | 0.635 | Not within 25% of top value | N | **0** | | N | **0** |
| RCS | RCS Cold Leg | RCP Casing/Discharge Nozzle Junction | 0.915 | SS | Low | 558 | Slow | 0.813 | 0.26 | 1 | -5.586 | 8.292 | 0.26 | -6.908 | 10.964 | 9.628 | 8.810 | $U_{en}$ > 1.0 | Y | **1** | Highest SS $U_{en}$ in T.Z. | Y | **1** |
| | | RCS 2-inch Crossover Leg Loops 1, 2 Drain Nozzles | 0.7 | SS | Low | 558 | Slow | 0.813 | 0.26 | 1 | -5.586 | 8.292 | 0.26 | -6.908 | 10.964 | 9.628 | 6.740 | > 50% of top location | Y | **1** | The SS Charging Nozzle bounds the crossover leg components on a Common Basis Stress Evaluation basis. | N | **0** |
| | | RCS Crossover Leg Loops 1, 2, 3, 4 | 0.5 | SS | Low | 558 | Slow | 0.813 | 0.26 | 1 | -5.586 | 8.292 | 0.26 | -6.908 | 10.964 | 9.628 | 4.814 | Not within 25% of top value | N | **0** | | N | **0** |
| | | RCS Cold Leg Loops 1, 2, 3, 4 | 0.37 | SS | Low | 558 | Slow | 0.813 | 0.26 | 1 | -5.586 | 8.292 | 0.26 | -6.908 | 10.964 | 9.628 | 3.562 | Not in top 3 locations | N | **0** | | N | **0** |
| | | RCS Cold Leg Thermowells | 0.025 | Ni-Cr-Fe | Low | 558 | Slow | 0.899 | 0.16 | 1 | -8.112 | 3.212 | 0.16 | -9.433 | 3.885 | 3.549 | 0.089 | $U_{en}$ < 1.0; sole location | Y | **1** | $U_{en}$ < 0.8 | N | **0** |
| | RCS Hot Leg | RCS Hot Leg Loops 1, 2, 3, 4 | 0.95 | SS | Low | 619 | Slow | 1 | 0.26 | 1 | -5.586 | 10.885 | 0.26 | -6.908 | 15.348 | 13.117 | 12.461 | $U_{en}$ > 1.0 | Y | **1** | The Hot Leg Surge Nozzle bounds the hot leg components on a Common Basis Stress Evaluation basis | N | **0** |
| | | RCS Hot Leg Thermowells | 0.017 | Ni-Cr-Fe | Low | 619 | Slow | 1 | 0.16 | 1 | -8.112 | 3.662 | 0.16 | -9.433 | 4.524 | 4.093 | 0.070 | $U_{en}$ < 1.0; sole location | Y | **1** | $U_{en}$ < 0.8 | N | **0** |
| RHR | RHR Inlet | RHR 12-inch Hot Leg Loops 1, 4 RHR Nozzles | 0.81 | SS | Low | 619 | Medium | 1 | 0.26 | 1 | -2.856 | 5.352 | 0.26 | -6.908 | 15.348 | 10.350 | 8.384 | $U_{en}$ > 1.0 | Y | **1** | NUREG/CR-6260 | Y | **1** |
| | | RHR 12-inch RHR Loops 1, 4 (Hot Leg SI portion) | 0.661 | SS | Low | 390 | Low-Mid | 0.279 | 0.26 | 1 | -4.217 | 3.460 | 0.26 | -6.908 | 4.207 | 3.833 | 2.534 | < 50% of top location | N | **0** | | N | **0** |
| | | RHR 12-inch RHR Pump Suction Isolation Valves | 0.64 | SS | Low | 390 | Low-Mid | 0.279 | 0.26 | 1 | -4.217 | 3.460 | 0.26 | -6.908 | 4.207 | 3.833 | 2.453 | < 50% of top location | N | **0** | | N | **0** |
| | | RHR 12-inch RCS Hot Leg to RHR Pump Isolation Valves | 0.64 | SS | Low | 390 | Low-Mid | 0.279 | 0.26 | 1 | -4.217 | 3.460 | 0.26 | -6.908 | 4.207 | 3.833 | 2.453 | < 50% of top location | N | **0** | | N | **0** |
| | | RHR 12-inch RHR Suction , Loops 1, 4 | 0.296 | SS | Low | 390 | Low-Mid | 0.279 | 0.26 | 1 | -4.217 | 3.460 | 0.26 | -6.908 | 4.207 | 3.833 | 1.135 | < 50% of top location | N | **0** | | N | **0** |
| | RHR Outlet | RHR 6-inch SI & RHR System Loop 2 & 3 Recirculation Supply Header Check Valves | 0.17 | SS | Low | 390 | Low-Mid | 0.279 | 0.26 | 1 | -4.217 | 3.460 | 0.26 | -6.908 | 4.207 | 3.833 | 0.652 | $U_{en}$ < 1.0; sole location | Y | **1** | $U_{en}$ < 0.8 | N | **0** |
| | | RHR 6-inch SI/RHR Isolation Valves | 0.17 | SS | Low | 390 | Low-Mid | 0.279 | 0.26 | 1 | -4.217 | 3.460 | 0.26 | -6.908 | 4.207 | 3.833 | 0.652 | $U_{en}$ < 1.0; sole location | Y | **1** | $U_{en}$ < 0.8 | N | **0** |
| | | RHR 6-inch RHR Pumps to RCS Cold Leg Check Valves | 0.165 | SS | Low | 390 | Low-Mid | 0.279 | 0.26 | 1 | -4.217 | 3.460 | 0.26 | -6.908 | 4.207 | 3.833 | 0.633 | | N | **0** | | N | **0** |
| SI | BIT | SI 3-inch Cold Leg (All Loops) Boron Injection Nozzle | 0.999 | SS | Low | 558 | Medium | 0.813 | 0.26 | 1 | -2.856 | 4.657 | 0.26 | -6.908 | 10.964 | 7.811 | 7.803 | NUREG/CR-6260 | Y | **1** | NUREG/CR-6260 | Y | **1** |
| | | SI 1.5-inch BIT Line Loops 1, 2, 3, 4 | 0.93 | SS | Low | 160 | Medium | 0 | 0.26 | 1 | -2.856 | 2.547 | 0.26 | -6.908 | 2.547 | 2.547 | 2.369 | < 50% of top location | N | **0** | | N | **0** |
| | | SI 1.5-inch BIT Line Common Header | 0.773 | SS | Low | 160 | Medium | 0 | 0.26 | 1 | -2.856 | 2.547 | 0.26 | -6.908 | 2.547 | 2.547 | 1.969 | < 50% of top location | N | **0** | | N | **0** |
| | Accumulator | SI 10-inch Cold Leg (All Loops) Accumulator Nozzle | 0.95 | SS | Low | 558 | Medium | 0.813 | 0.26 | 1 | -2.856 | 4.657 | 0.26 | -6.908 | 10.964 | 7.811 | 7.420 | $U_{en}$ > 1.0 | Y | **1** | Highest SS $U_{en}$ in T.Z. | Y | **1** |
| | | SI 10-inch Accumulator Line Loops 1, 2, 3, 4 Piping | 0.98 | SS | Low | 160 | Medium | 0 | 0.26 | 1 | -2.856 | 2.547 | 0.26 | -6.908 | 2.547 | 2.547 | 2.496 | < 50% of top location | N | **0** | | N | **0** |
| | | SI 10-inch Cold Leg (All Loops) Accumulator Nozzle Check Valves | 0.26 | SS | Low | 160 | Medium | 0 | 0.26 | 1 | -2.856 | 2.547 | 0.26 | -6.908 | 2.547 | 2.547 | 0.662 | | N | **0** | | N | **0** |
| | | SI 10-inch Accumulator Line Loops 1, 2, 3, 4 Outlet Upstream Check Valves | 0.26 | SS | Low | 160 | Medium | 0 | 0.26 | 1 | -2.856 | 2.547 | 0.26 | -6.908 | 2.547 | 2.547 | 0.662 | | N | **0** | | N | **0** |

*Table 5-1 (continued)*
*Results from Pilot PWR Plant Evaluation Using $F_{en}$ Estimation Evaluation Procedure*

| | | | Input | | | | | $F_{en}$ Computation | | | | | | | | | Comparison | Identification of Sentinel Locations | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **System or Vessel** | **Thermal Zone** | **Component** | **Design Basis CUF** | **Material** | **D0** | **Temp** | **Strain Rate** | **T*** | **O*** | **S*** | $\dot{\varepsilon}^*$ | **$F_{en}$ Exp.** | **Worst O*** | **Worst** $\dot{\varepsilon}^*$ | **$F_{en}$ Max.** | **$F_{en}$ Average ($F_{en}$*)** | **$U_{en\ design}$ ($U_{en}$*)** | **Criterion for Initial Sentinel Locations** | **Initial Sentinel Location** | **Count** | **Criterion for Final Sentinel Location** | **Final Sentinel Location** | **Count** |
| | SIS | SI 6-inch Hot Leg Loops 2, 3 SIS Nozzle | 0.1 | SS | Low | 619 | Medium | 1 | 0.26 | 1 | -2.856 | 5.352 | 0.26 | -6.908 | 15.348 | 10.350 | 1.035 | $U_{en}$ > 1.0 | Y | **1** | Highest SS $U_{en}$ in T.Z. | Y | **1** |
| | SIS | SI 6-inch Hot Leg Loops 2,3 SIS Piping | 0.09 | SS | Low | 160 | Medium | 0 | 0.26 | 1 | -2.856 | 2.547 | 0.26 | -6.908 | 2.547 | 2.547 | 0.229 | < 50% of top location | N | **0** | | N | **0** |
| Steam Generator | Primary Head | RSG Primary Manway Drain Tube | 0.391 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.960 | $U_{en}$ < 1.0; sole location | Y | **1** | The LAS RSG Tubesheet (Continuous Region) bounds the LAS Primary Head Locations. | N | **0** |
| Steam Generator | Primary Head | RSG Primary Manway Cover | 0.349 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.857 | | N | **0** | | N | **0** |
| Steam Generator | Primary Head | RSG Primary Nozzle Drain Tube | 0.316 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.776 | | N | **0** | | N | **0** |
| Steam Generator | Primary Head | RSG Primary Inlet Nozzle | 0.009 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.022 | | N | **0** | | N | **0** |
| Steam Generator | Primary Head | RSG Primary Outlet Nozzle | 0.007 | LAS | Low | 558 | Slow | 14.36 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.017 | | N | **0** | | N | **0** |
| Steam Generator | Tubesheet | RSG Tubesheet (Continuous Region) | 0.428 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 1.051 | $U_{en}$ > 1.0 | Y | **1** | Highest LAS Uen in T.Z. | Y | **1** |
| Steam Generator | Tubesheet | RSG Tubesheet (Perforated Region) | 0.122 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.299 | < 50% of top location | N | **0** | | N | **0** |
| Steam Generator | Tubesheet | RSG Tube-to-Tubesheet Connection | 0.068 | Ni-Cr-Fe | Low | 619 | Slow | 1 | 0.16 | 1 | -8.112 | 3.662 | 0.16 | -9.433 | 4.524 | 4.093 | 0.278 | $U_{en}$ < 1.0; sole location | Y | **1** | $U_{en}$ < 0.8 | N | **0** |
| Steam Generator | Channel Head | RSG Channel Head at Primary Manway | 0.059 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.145 | $U_{en}$ < 1.0; sole location | Y | **1** | $U_{en}$ < 0.8 | N | **0** |
| Steam Generator | Channel Head | RSG Channel Head near Tubesheet | 0.004 | LAS | Low | 619 | Slow | 17.79 | 0 | 0.015 | -6.502 | 2.455 | 0 | -6.908 | 2.455 | 2.455 | 0.010 | | N | **0** | | N | **0** |
| | | | | | | | | | | | | | | | | | | | | **44** | | | **22** |

*Table 5-2*
*Final Sentinel Locations for PWR Pilot Plant*

| System or Vessel | Thermal Zone | Material | Component | Sentinel Locations | |
|---|---|---|---|---|---|
| | | | | **Initial** | **Final** |
| Reactor Pressure Vessel | RPV Nozzle | RPV Outlet Nozzle<br>RPV Inlet Nozzle | LAS<br>LAS | 1<br>1 | 1<br>1 |
| | RPV Upper Head | RPV Core Exit Thermocouple Nozzle Assembly Upper Nozzle Housing<br>RPV Vessel Flange | SS<br>LAS | 1<br>1 | 1<br>0 |
| | RPV Bottom Head | RPV Vessel Wall Transition<br>RPV Bottom Head-to-Shell Juncture<br>RPV Bottom Head Instrument Tubes (pos. 2) | LAS<br>LAS<br>Ni-Cr-Fe | 1<br>1<br>1 | 0<br>1<br>1 |
| Pressurizer | Pressurizer Lower Head | Pressurizer Heater Penetration<br>Pressurizer Shell at Support Lug<br>Pressurizer Surge Nozzle<br>Pressurizer Lower Head/Support Skirt | SS<br>LAS<br>LAS<br>LAS | 1<br>1<br>1<br>1 | 1<br>1<br>1<br>1 |
| | Pressurizer Spray Nozzle | Pressurizer Spray Nozzle | SS | 1 | 1 |
| | Pressurizer SRV/PORV | 6-inch and 3-inch Pressurizer Safety and Relief Valve Piping<br>Pressurizer 3-inch x 6-inch Power Operated Relief Valve Solenoid | SS<br>SS | 1<br>1 | 1<br>1 |
| | Pressurizer Upper Head | Pressurizer Instrument Nozzle<br>Pressurizer Upper Head/Upper Shell | SS<br>LAS | 1<br>1 | 0<br>0 |
| Surge Piping | Surge Line Piping | 14-inch Hot Leg Surge Nozzle | SS | 1 | 1 |
| Spray Piping | Spray Line Piping | 4-inch Spray Piping at Pressurizer Spray Nozzle | SS | 1 | 0 |
| CVCS | Charging Loop 1 | CVCS 3-inch Cold Leg Loop 1 Normal Charging Nozzle<br>CVCS 3-inch Normal Charging Loop 1 Piping | SS<br>SS | 1<br>1 | 1<br>0 |
| | Charging Loop 4 | CVCS 3-inch Cold Leg Loop 4 Alternate Charging Nozzle<br>CVCS 3-inch Alternate Charging Loop 4 Piping | SS<br>SS | 1<br>1 | 1<br>0 |
| | Letdown Loop 3 | CVCS 3-inch Normal Letdown Piping, Crossover Loop 3 | SS | 1 | 0 |
| | Letdown Loop 4 | CVCS 2-inch Crossover Leg Loop 4 Excess Letdown Nozzle | SS | 1 | 0 |
| | Auxiliary Spray | Auxiliary Spray Piping | SS | 1 | 1 |
| | Drain | CVCS Drain Line, Loop 2<br>CVCS Drain Line, Loop 3 | SS<br>SS | 1<br>1 | 0<br>0 |
| | Seal Water | CVCS 1-1/2-inch, 2-inch Seal Water Injection Loops 3 Piping<br>CVCS 1-1/2-inch, 2-inch Seal Water Injection Loops 4 Piping | SS<br>SS | 1<br>1 | 0<br>0 |
| RCS | RCS Cold Leg | RCP Casing/Discharge Nozzle Junction<br>RCS 2-inch Crossover Leg Loops 1, 2 Drain Nozzles<br>RCS Cold Leg Thermowells | SS<br>SS<br>Ni-Cr-Fe | 1<br>1<br>1 | 1<br>0<br>0 |
| | RCS Hot Leg | RCS Hot Leg Loops 1, 2, 3, 4<br>RCS Hot Leg Thermowells | SS<br>Ni-Cr-Fe | 1<br>1 | 0<br>0 |
| RHR | RHR Inlet | RHR 12-inch Hot Leg Loops 1, 4 RHR Nozzles | SS | 1 | 1 |
| | RHR Outlet | RHR 6-inch SI & RHR System Loop 2 & 3 Recirculation Supply Header Check Valves<br>RHR 6-inch SI/RHR Isolation Valves | SS<br>SS | 1<br>1 | 0<br>0 |
| SI | BIT | SI 3-inch Cold Leg (All Loops) Boron Injection Nozzle | SS | 1 | 1 |
| | Accumulator | SI 10-inch Cold Leg (All Loops) Accumulator Nozzle | SS | 1 | 1 |
| | SIS | SI 6-inch Hot Leg Loops 2, 3 SIS Nozzle | SS | 1 | 1 |
| Steam Generator | Primary Head | RSG Primary Manway Drain Tube | LAS | 1 | 0 |
| | Tubesheet | RSG Tubesheet (Continuous Region)<br>RSG Tube-to-Tubesheet Connection | LAS<br>Ni-Cr-Fe | 1<br>1 | 1<br>0 |
| | Channel Head | RSG Channel Head at Primary Manway | LAS | 1 | 0 |
| | | | | **44** | **22** |

*Table 5-3*
*Sample Results from Pilot PWR Plant Evaluation Using Common Basis Stress Evaluation Procedure (Properties)*

| **System** | **Component** | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Type | Material | OD | t | ID | k | E (ksi) | $\alpha$ | $K_2$ | $K_3$ | $S_m$ | $S_{1e6}$ |
| CVCS | Charging Nozzle | SS | 3.5 | 0.437 | 2.626 | 10 | 2.67E+04 | 1.00E-05 | 1.1 | 1.1 | 16.7 | 28.3 |
| | Charging Piping | SS | 3.5 | 0.437 | 2.626 | 10 | 2.67E+04 | 1.00E-05 | 1.1 | 1.1 | 16.7 | 28.3 |

*Table 5-4*
*Sample Results from Pilot PWR Plant Evaluation Using Common Basis Stress Evaluation Procedure (Transient Details)*

| System | Component | | Transient 1 | | | | | | | | | | | Detailed | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Type | Name | V | Q (gpm) | $T_{mean}$ | $T_{max}$ | $\phi$ | h | k/ht | $t_0$ | $\rho c_p$ | $kt_0/\rho c_p t$^2 | $\Delta T$ | $E\alpha\Delta T/(1-\nu)$ | $(\sigma_{max})/(E\alpha\Delta T/(1-\nu))$ | $S_{ts\ (detailed)}$ | $S_{mom}$ | $S_{p\ (detailed)}$ |
| CVCS | Charging Nozzle | LOC/LOL Trip and Return Down | 5 | 84.24901871 | 330 | 560 | 267.35 | 1632.30 | 0.168 | 10 | 100 | 0.21 | 460 | 175 | 0.48 | 84 | 55 | 139 |
| | | LOC Delayed Down | 5 | 84.24901871 | 330 | 560 | 267.35 | 1632.30 | 0.168 | 0.01 | 100 | 0.00 | 460 | 175 | 0.50 | 88 | 55 | 143 |
| | | LOL Delayed Down 1 | 3.8 | 64.02925422 | 315 | 530 | 263.46 | 1291.48 | 0.213 | 100 | 100 | 2.09 | 430 | 164 | 0.15 | 25 | 48 | 72 |
| | | LOL Delayed Down 2 | 3.8 | 64.02925422 | 335 | 570 | 268.57 | 1316.52 | 0.209 | 10 | 100 | 0.21 | 470 | 179 | 0.45 | 81 | 55 | 136 |
| | | LOL Prompt Down | 5 | 84.24901871 | 300 | 500 | 259.23 | 1582.71 | 0.173 | 100 | 100 | 2.09 | 400 | 153 | 0.15 | 23 | 44 | 67 |
| | Charging Piping | LOC/LOL Trip and Return Down | | 55 | 300 | 500 | 259.23 | 1125.22 | 0.244 | 100 | 100 | 2.09 | 400 | 153 | 0.14 | 21 | 41 | 63 |
| | | LOC Delayed Down | | 55 | 330 | 560 | 267.35 | 1160.48 | 0.237 | 100 | 100 | 2.09 | 460 | 175 | 0.14 | 25 | 48 | 72 |
| | | LOL Delayed Down 1 | | 55 | 300 | 500 | 259.23 | 1125.22 | 0.244 | 100 | 100 | 2.09 | 400 | 153 | 0.14 | 21 | 41 | 63 |
| | | LOL Prompt Down | | 55 | 300 | 500 | 259.23 | 1125.22 | 0.244 | 100 | 100 | 2.09 | 400 | 153 | 0.14 | 21 | 41 | 63 |
| | | | | | | | | | | | | | | | | | PPSMAX | 0.66 |

| System | Component | | Transient 2 | | | | | | | | | | | Detailed | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Type | Name | V | gpm | $T_{mean}$ | $T_{max}$ | $\phi$ | h | k/ht | $t_0$ | $\rho c_p$ | $kt_0/\rho c_p t$^2 | $\Delta T$ | $E\alpha\Delta T/(1-\nu)$ | $(\sigma_{max})/(E\alpha\Delta T/(1-\nu))$ | $S_{ts\ (detailed)}$ | $S_{mom}$ | $S_{p\ (detailed)}$ |
| CVCS | Charging Nozzle | LOC/LOL Trip and Return Up | 5 | 84.24901871 | 300 | 500 | 259.23 | 1582.71 | 0.173 | 90 | 100 | 1.885 | 400 | 153 | 0.15 | 23 | 44 | 67 |
| | | LOC Delayed Up | 5 | 84.24901871 | 300 | 500 | 259.23 | 1582.71 | 0.173 | 90 | 100 | 1.885 | 400 | 153 | 0.15 | 23 | 44 | 67 |
| | | LOL Delayed Up 1 | 3.8 | 64.02925422 | 335 | 570 | 268.57 | 1316.52 | 0.209 | 0.01 | 100 | 0.000 | 470 | 179 | 0.48 | 86 | 55 | 141 |
| | | LOL Delayed Up 2 | 3.8 | 64.02925422 | 315 | 530 | 263.46 | 1291.48 | 0.213 | 100 | 100 | 2.095 | 430 | 164 | 0.15 | 25 | 48 | 72 |
| | | LOL Prompt Up | 5 | 84.24901871 | 300 | 500 | 259.23 | 1582.71 | 0.173 | 100 | 100 | 2.09 | 400 | 153 | 0.15 | 23 | 44 | 67 |
| | Charging Piping | LOC/LOL Trip and Return Up | | 55 | 300 | 500 | 259.23 | 1125.22 | 0.244 | 100 | 100 | 2.095 | 400 | 153 | 0.15 | 23 | 44 | 67 |
| | | LOC Delayed Up | | 55 | 330 | 560 | 267.35 | 1160.48 | 0.237 | 100 | 100 | 2.095 | 460 | 175 | 0.15 | 26 | 51 | 77 |
| | | LOL Delayed Up 1 | | 55 | 300 | 500 | 259.23 | 1125.22 | 0.244 | 100 | 100 | 2.095 | 400 | 153 | 0.15 | 23 | 44 | 67 |
| | | LOC/LOL Trip and Return Up | | 55 | 300 | 500 | 259.23 | 1125.22 | 0.244 | 100 | 100 | 2.095 | 400 | 153 | 0.15 | 23 | 44 | 67 |
| | | | | | | | | | | | | | | | | | PPSMAX | 0.66 |

*Table 5-5*
*Sample Results from Pilot PWR Plant Evaluation Using Common Basis Stress Evaluation Procedure (Computed Values)*

| System | Component | Transient Pair | Combined | | | | Design | | | | Projected | | | | $F_{en}$ factor | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | Detailed | | | | | | Tensile | Compressive | | | Tensile | Compressive | | | | | | | | | |
| | Type | | $S_{peak}$ | $K_e$ | $S_a$ | $N_{allow}$ | $n_{design}$ | $U_{incr}$ | $U_{incr}$ | $U_{incr}$ | $n_{proj}$ | $U_{incr\ proj}$ | $U_{incr}$ | $U_{incr}$ | T* | O* | $\dot{\varepsilon}$ | $\dot{\varepsilon}$ used | $\dot{\varepsilon}^*$ | Worst $\dot{\varepsilon}^*$ | $F_{en\ expect}$ | $F_{en\ worst}$ | $F_{en\ average}$ |
| CVCS | Charging Nozzle | LOC/LOL Trip and Return Up/Down | 207 | 2.370 | 245 | 305 | 60 | 0.1967 | 0.1326 | 0.0641 | 1 | 0.0033 | 0.0022 | 0.0011 | 0.82 | 0.26 | 0.0521 | 0.0521 | -2.037491708 | -6.907755279 | 3.93 | 9.07 | 6.50 |
| | | LOC Delayed Up/Down | 210 | 3.190 | 335 | 86 | 20 | 0.2326 | 0.1580 | 0.0745 | 1 | 0.0116 | 0.0079 | 0.0037 | 0.82 | 0.26 | 53.4564 | 0.4000 | 0 | -6.907755279 | 2.55 | 9.07 | 5.81 |
| | | LOL Delayed Up 1/Down 1 | 213 | 3.190 | 340 | 86 | 20 | 0.2326 | 0.0789 | 0.1537 | 1 | 0.0116 | 0.0039 | 0.0077 | 0.85 | 0.26 | 0.0027 | 0.0027 | -4.994484904 | -6.907755279 | 7.69 | 9.60 | 8.65 |
| | | LOL Delayed Up 2/Down 2 | 208 | 2.430 | 253 | 270 | 20 | 0.0741 | 0.0483 | 0.0258 | 1 | 0.0037 | 0.0024 | 0.0013 | 0.85 | 0.26 | 0.0508 | 0.0508 | -2.063301759 | -6.907755279 | 4.02 | 9.60 | 6.81 |
| | | LOL Prompt Up/Down | 135 | 2.054 | 138 | 567 | 200 | 0.3527 | 0.1764 | 0.1764 | 1 | 0.0018 | 0.0009 | 0.0009 | 0.63 | 0.26 | 0.0025 | 0.0025 | -5.066805566 | -6.907755279 | 5.83 | 6.44 | 6.14 |
| | Charging Piping | LOC/LOL Trip and Return Up/Down | 130 | 2.054 | 134 | 567 | 60 | 0.1058 | 0.0511 | 0.0547 | 1 | 0.0018 | 0.0009 | 0.0009 | 0.63 | 0.26 | 0.0024 | 0.0024 | -5.135798437 | -6.907755279 | 5.90 | 6.44 | 6.17 |
| | | LOC Delayed Up/Down | 150 | 2.180 | 163 | 436 | 20 | 0.0459 | 0.0221 | 0.0237 | 1 | 0.0023 | 0.0011 | 0.0012 | 0.82 | 0.26 | 0.0027 | 0.0027 | -4.996036495 | -6.907755279 | 7.38 | 9.07 | 8.23 |
| | | LOL Delayed Up 1/Down 1 | 130 | 2.054 | 134 | 567 | 20 | 0.0353 | 0.0170 | 0.0182 | 1 | 0.0018 | 0.0009 | 0.0009 | 0.63 | 0.26 | 0.0024 | 0.0024 | -5.135798437 | -6.907755279 | 5.90 | 6.44 | 6.17 |
| | | LOC/LOL Trip and Return Up/Down | 130 | 2.054 | 134 | 567 | 200 | 0.3527 | 0.1703 | 0.1824 | 1 | 0.0018 | 0.0009 | 0.0009 | 0.63 | 0.26 | 0.0024 | 0.0024 | -5.135798437 | -6.907755279 | 5.90 | 6.44 | 6.17 |

| System | Component | Transient Pair | Design $U_{en}$ ($U_{en}$*) | | | | | Projected $U_{en}$ ($U_{en}$*) | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | Tensile | Compressive | Total | Sum | Rank | Tensile | Compressive | Total | Sum | Rank |
| | Type | | $U_{en\ incr}$ | $U_{en\ incr}$ | $U_{en\ total}$ | $U_{en}$ | | $U_{en\ incr}$ | $U_{en\ incr}$ | $U_{en\ total}$ | $U_{en}$ | |
| CVCS | Charging Nozzle | LOC/LOL Trip and Return Up/Down | 0.8620 | 0.0641 | 0.9262 | 4.3676 | 1 | 0.0069 | 0.0011 | 0.0080 | 0.1239 | 1 |
| | | LOC Delayed Up/Down | 0.9180 | 0.0745 | 0.9925 | | | 0.0216 | 0.0037 | 0.0254 | | |
| | | LOL Delayed Up 1/Down 1 | 0.6817 | 0.1537 | 0.8355 | | | 0.0664 | 0.0077 | 0.0741 | | |
| | | LOL Delayed Up 2/Down 2 | 0.3291 | 0.0258 | 0.3548 | | | 0.0088 | 0.0013 | 0.0101 | | |
| | | LOL Prompt Up/Down | 1.0823 | 0.1764 | 1.2586 | | | 0.0054 | 0.0009 | 0.0063 | | |
| | Charging Piping | LOC/LOL Trip and Return Up/Down | 0.3152 | 0.0547 | 0.3699 | 1.9321 | 2 | 0.0056 | 0.0009 | 0.0065 | 0.0306 | 2 |
| | | LOC Delayed Up/Down | 0.1822 | 0.0237 | 0.2059 | | | 0.0098 | 0.0012 | 0.0109 | | |
| | | LOL Delayed Up 1/Down 1 | 0.1051 | 0.0182 | 0.1233 | | | 0.0056 | 0.0009 | 0.0065 | | |
| | | LOC/LOL Trip and Return Up/Down | 1.0506 | 0.1824 | 1.2330 | | | 0.0056 | 0.0009 | 0.0065 | | |

# Section 6: Concluding Remarks

This report provides the technical basis of a screening process that can be used to evaluate a plant to determine EAF limiting locations for fatigue monitoring. Procedures for this example screening evaluation are described and applied to a pilot PWR plant.

A primary reason for developing this process is to equip license renewal applicants with a consistent method to identify EAF limiting locations additional to the sample locations evaluated in NUREG/CR-6260 for their reactor type and vintage.

Guiding principles for the process included:

1. Consistent technical basis.
2. Analytical method using readily available design input from P&IDs, piping isometric drawings and piping stress reports.
3. Only basic stress or fatigue analysis required.

The following are the basic areas of new technology developed by this project:

1. Procedure for Estimating $F_{en}$ Factors.
2. Procedure for Estimating $U_{en}$.

Each of these areas is discussed in detail. An example of the process is shown in Section 5.

The process developed in this report provides guidance for the evaluation and relative ranking of estimated $U_{en}$ values for locations in components and systems where EAF is a concern to minimize the possibility of the need for a formal fatigue evaluation. The estimated values for a number of Sentinel Locations are compared to $U_{en}$ values for locations in each given system/component that have been specifically identified in regulatory guidance as of concern for EAF. Locations from previous guidance are to be managed and if estimated values for other locations are higher or as high, these other locations should also be managed. For components/systems where there are no locations included in previous regulatory guidelines, the recommendation is to manage up to the three locations with highest estimated $U_{en}$ values.

Further analysis beyond the basic screening steps may be applied to reduce the number of locations.

# Section 7: References


1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers.
2. ANSI/ASME B31.7-1969, "Nuclear Power Piping," American National Standards Institute.
3. ANSI/ASME B31.1, "Power Piping," American National Standards Institute.
4. NUREG/CR-6260 (INEL-95/0045), Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components, March 1995.
5. NUREG/CR-5704 (ANL-98/31), Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels, 1999.
6. NUREG/CR-6583 (ANL-97/18), Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels, 1998.
7. NUREG/CR-6909 (ANL-06/08), Effects of LWR Coolant Environments on the Fatigue Life of Reactor Materials Final Report, February 2007.
8. NUREG-1801, Revision 2, *Generic Aging Lessons Learned (GALL) Report*, U. S. Nuclear Regulatory Commission, December 2010.
9. EPRI Report "Fatigue Comparison of Piping Designed to ANSI B31.1 and ASME Section III, Class 1 Rules," TR-102901, EPRI, Palo Alto, CA, December 1993.
10. EPRI Technical Report, "Improved Basis and Requirements for Break Location Postulation," EPRI, Palo Alto, CA: 2011. 1022873.
11. Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, MRP-47, Revision 1, September 2005.
12. Materials Reliability Program: Fatigue Management Handbook, Revision 1 (MRP-235 w/ corrections), EPRI, Palo Alto, CA: 2009. 1015010.
13. EPRI/BWRVIP Memo. No. 2005-271, Potential Error in Existing Fatigue Reactor Water Environmental Effects Analyses, July 1, 2005.
14. Regulatory Guide 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components due to the Effects of Light-Water Reactor Environment for New Reactors," March 2007.

# Appendix A: Summary of $F_{en}$ Formulations as Accepted by the NRC

## $F_{en}$ Formulations for Ferritic Materials

### *NUREG/CR-6583 (Old Rules)*

The following are the appropriate $F_{en}$ relationships from NUREG/CR-6583 [6] for carbon and low alloy steels. These expressions are:

*For Carbon Steel [6, p. 69]:*

$$F_{en} = \exp(0.585 - 0.00124T' - 0.101S^*T^*O^*\dot{\varepsilon}^*) = \exp(0.554 - 0.101S^*T^*O^*\dot{\varepsilon}^*)$$

*Equation A-1*

*For Low Alloy Steel [6, p. 69]:*

$$F_{en} = \exp(0.929 - 0.00124T' - 0.101S^*T^*O^*\dot{\varepsilon}^*) = \exp(0.898 - 0.101S^*T^*O^*\dot{\varepsilon}^*)$$

*Equation A-2*

*Note that the above expressions have been corrected as summarized in Reference [13].*

where:

| | | |
|---|---|---|
| $F_{en}$ | = | fatigue life correction factor |
| $T'$ | = | 25°C (NUREG/CR-6583, Section 6, $F_{en}$ relative to room temperature air) |
| $S^*$ | = | $S$ for $0 < S \leq 0.015$ wt. % |
| | = | 0.015 for $S > 0.015$ wt. % |
| $S$ | = | weight percent sulfur of steel |
| $T^*$ | = | 0 for $T < 150$°C |
| | = | $(T - 150)$ for $150 \leq T \leq 350$°C |
| $T$ | = | service temperature (°C) |
| $O^*$ | = | 0 for $DO < 0.05$ parts per million (ppm) |
| | = | $\ln(DO/0.04)$ for $0.05 \text{ ppm} \leq DO \leq 0.5$ ppm |
| | = | $\ln(12.5)$ for $DO > 0.5$ ppm |

| | | |
|---|---|---|
| $DO$ | = | dissolved oxygen |
| $\dot{\varepsilon}^*$ | = | 0 for $\dot{\varepsilon}$ > 1%/sec |
| | = | ln($\dot{\varepsilon}$) for 0.001 ≤ $\dot{\varepsilon}$ ≤ 1%/sec |
| | = | ln(0.001) for $\dot{\varepsilon}$ < 0.001%/sec |
| $\dot{\varepsilon}$ | = | strain rate, %/sec |

## *NUREG/CR-6909 (New Rules)*

*For Carbon Steel (CS) [7, p. A.1]:*

$$F_{en} = \exp(0.632 - 0.101S^*T^*O^*\dot{\varepsilon}^*) \qquad \text{Equation A-3}$$

*For Low Alloy Steel (LAS) [7, p. A.1]:*

$$F_{en} = \exp(0.702 - 0.101S^*T^*O^*\dot{\varepsilon}^*) \qquad \text{Equation A-4}$$

Where $S^*$, $T^*$, $O^*$, and $\dot{\varepsilon}^*$ are the transformed sulfur content, service temperature, dissolved oxygen (DO), and strain rate, respectively, which are defined as follows [7, p. A.1 and A.2]:

| | | |
|---|---|---|
| $F_{en}$ | = | fatigue life correction factor |
| $S^*$ | = | 0.001 for $S \le 0.001$ wt.% |
| | = | $S$ for $S \le 0.015$ wt. % |
| | = | 0.015 for $S > 0.015$ wt. % |
| $T^*$ | = | 0 for $T < 150$°C |
| | = | ($T$ - 150) for 150 ≤ $T$ ≤ 350°C |
| $T$ | = | service temperature (°C) |
| $O^*$ | = | 0 for dissolved oxygen, $DO \le 0.04$ parts per million (ppm) |
| | = | ln($DO$/0.04) for 0.04 ppm < $DO \le 0.5$ ppm |
| | = | ln(12.5) for $DO > 0.5$ ppm |
| $\dot{\varepsilon}^*$ | = | 0 for strain rate, $\dot{\varepsilon}$ > 1%/sec |
| | = | ln($\dot{\varepsilon}$) for 0.001 ≤ $\dot{\varepsilon}$ ≤ 1%/sec |
| | = | ln(0.001) for $\dot{\varepsilon}$ < 0.001%/sec |

For both carbon and low-alloy steels, a threshold value of 0.07% for strain amplitude (one-half the strain range for the cycle) is defined, below which environmental effects on the fatigue life of these steels do not occur. This strain threshold corresponds to 21 ksi (145 MPa) alternating stress intensity from the fatigue analysis. That is, if $\sigma_{alt} \le 21$ ksi then $F_{en} = 1.0$.

$F_{en} = 1$ for strain amplitude, $\varepsilon_a \le 0.07\%$ or $S_{alt} \le (E_c)(0.07\%)/(100\%) = 21$ ksi (145 MPa)

## $F_{en}$ Formulations for Austenitic Stainless Steel Materials

### *NUREG/CR-5704 (Old Rules)*

*For Types 304 and 316 Stainless Steel [5]:*

$$F_{en} = \exp(0.935 - T^{*}\dot{\varepsilon}^{*}O^{*}) \qquad \text{Equation A-5}$$

where:

| | | |
|---|---|---|
| $F_{en}$ | = | fatigue life correction factor |
| $T$ | = | service temperature of transient, °C |
| $T^{*}$ | = | 0 for $T < 200$°C |
| | = | 1 for $T \geq 200$°C |
| $\dot{\varepsilon}^{*}$ | = | 0 for strain rate, $\dot{\varepsilon} > 0.4$%/sec |
| | = | $\ln(\dot{\varepsilon}/0.4)$ for $0.0004 \leq \dot{\varepsilon} \leq 0.4$%/sec |
| | = | $\ln(0.0004/0.4)$ for $\dot{\varepsilon} < 0.0004$%/sec |
| $O^{*}$ | = | 0.260 for dissolved oxygen, $DO < 0.05$ parts per million (ppm) |
| | = | 0.172 for $DO \geq 0.05$ ppm |

### *NUREG/CR-6909 (New Rules)*

*For wrought and cast austenitic stainless steels (SS) [7, p. A.2]:*

$$F_{en} = \exp(0.734 - T'O'\dot{\varepsilon}') \qquad \text{Equation A-6}$$

where:

| | | |
|---|---|---|
| $F_{en}$ | = | fatigue life correction factor |
| $T'$ | = | 0 for $T < 150$°C |
| | = | $(T - 150)/175$ for $150 \leq T < 325$°C |
| | = | 1 for $T \geq 325$°C |
| $T$ | = | service temperature (°C) |
| $\dot{\varepsilon}'$ | = | 0 for strain rate, $\dot{\varepsilon} > 0.4$%/sec |
| | = | $\ln(\dot{\varepsilon}/0.4)$ for $0.0004 \leq \dot{\varepsilon} \leq 0.4$%/sec |
| | = | $\ln(0.0004/0.4)$ for $\dot{\varepsilon} < 0.0004$%/sec |
| $O'$ | = | 0.281 for all dissolved oxygen levels |

For wrought and cast austenitic stainless steels, a threshold value of 0.10% for strain amplitude (one-half the strain range for the cycle) is defined, below which environmental effects on the fatigue life of these steels do not occur. This strain threshold corresponds to 28.3 ksi (195 MPa) alternating stress intensity from the fatigue analysis. That is, if $S_{alt} \leq 28.3$ ksi then $F_{en} = 1.0$. Thus,

$F_{en} = 1$ for strain amplitude, $\varepsilon_a \leq 0.10\%$ or $S_{alt} \leq (E_c)(0.10\%)/(100\%) = 28.3$ ksi (195 MPa)

## $F_{en}$ Formulations for Nickel Alloy Materials

### *NUREG/CR-6909 (New Rules)*

New rules are required for Nickel Alloy materials per both the GALL report (license renewal) [8] and Reg. Guide 1.207 [14] (new plants).

*For Ni-Cr-Fe alloys [7, p. A.2]:*

$$F_{en} = \exp(-T'\dot{\varepsilon}'O') \qquad \text{Equation A-7}$$

where:

| | | |
|---|---|---|
| $F_{en}$ | = | fatigue life correction factor |
| $T'$ | = | $T/325$ for $T < 325$°C |
| | = | 1 for $T \geq 325$°C |
| $T$ | = | service temperature (°C) |
| $\dot{\varepsilon}'$ | = | 0 for strain rate, $\dot{\varepsilon} > 5.0$%/sec |
| | = | $\ln(\dot{\varepsilon}/5.0)$ for $0.0004 \leq \dot{\varepsilon} \leq 5.0$%/sec |
| | = | $\ln(0.0004/5.0)$ for $\dot{\varepsilon} < 0.0004$%/sec |
| $O'$ | = | 0.09 for NWC BWR water |
| | = | 0.16 for PWR or HWC BWR water |

For Ni-Cr-Fe alloys, a threshold value of 0.10% for strain amplitude (one-half the strain range for the cycle) is defined, below which environmental effects on the fatigue life of these steels do not occur. This strain threshold corresponds to 28.3 ksi (195 MPa) alternating stress intensity from the fatigue analysis. That is, if $S_{alt} \leq 28.3$ ksi then $F_{en} = 1.0$. Thus,

$F_{en} = 1$ for strain amplitude, $\varepsilon_a \leq 0.10$% or $S_{alt} \leq (E_c)(0.10\%)/(100\%) = 28.3$ ksi (195 MPa)

### *Dissolved Oxygen (DO)*

This is a value determined by the evaluator. For PWRs, the value has typically been established as below 50 ppm for plant conditions in which EAF is active (temperature above 150°C). For BWRs, this value is determined based on several factors, such as use of Hydrogen Water Chemistry program or not, and others. This value is typically obtained from individuals responsible for maintaining and monitoring plant water chemistry.

### *Strain Rate ($\dot{\varepsilon}$)*

This is determined as the strain difference over the increasingly tensile strain change divided by the time duration of the strain change.

# Appendix B: Rules for Evaluation of Class 1 Piping

## Rules for Evaluation of Class 1 Piping

This section is extracted in part from References [4] The ASME Section III rules for evaluation of Class 1 piping components have many similarities to the "design by analysis" rules of NB-3200. NB-3650 is based on the maximum shear stress theory, and primary, secondary and peak stress categories are evaluated. The allowable stress limits for the different stress categories are the same as for NB-3200, and if the limitations on primary plus secondary stresses are exceeded, simplified elastic-plastic analysis with consideration of thermal stress ratcheting is allowed. The major difference between NB-3200 and NB-3650 is that the latter takes a "design by formula" approach, with the design being considered acceptable if it passes a series of equations for the various loadings to which the component is exposed.

After primary stress limits are satisfied, Equation 10 must be satisfied for all pairs of load sets:

$$S_n = C_1 \frac{P_o D_o}{2t} + C_2 \frac{D_o}{2I} M_i + C_3 E_{ab} \left| \alpha_a T_a - \alpha_b T_b \right| \leq 3 S_m$$

*(Section III, Cl. 1, Eq. 10) [Eq. 2-6 of Ref. [4]]* *Equation B-1*

Where from [4]:

$C_1, C_2, C_3$ = secondary stress indices for the specific component under investigation

$D_o$ = outside diameter of pipe, in. (mm)

$t$ = nominal wall thickness of product, in. (mm)

$I$ = moment of inertia, $in^4$ ($mm^4$)

$S_m$ = allowable design stress intensity, psi (MPa)$P_o$
= range of service pressure, psi (MPa)

$M_i$ = resultant range of moment which occurs when the system goes from one service load set to another, in.-lb. (N-mm)

$E_{ab}$ = average modulus of elasticity of the two sides of a material or structural discontinuity at room temperature, psi (MPa)

$\alpha_a, \alpha_b$ = coefficient of thermal expansion on side a and side b of a structural or material discontinuity, in/in°F (mm/mm/°C)

$T_a$, $T_b$ = range of average temperature on side a and side b of a structural discontinuity, when the system goes from one service load to another, °F (°C)

The fatigue resistance of each piping component is assessed by evaluating the range of peak stress. For every pair of load sets, S, values are calculated using Equation 11:

$$S_p = K_1 C_1 \frac{P_o D_o}{2t} + K_2 C_2 \frac{D_o}{2I} M_i + K_3 C_3 E_{ab} \times \left| a_a T_a - a_b T_b \right| + \frac{1}{2(1-v)} K_3 Ea \left| \Delta T_1 \right| + \frac{1}{1-v} Ea \left| \Delta T_2 \right|$$

*(Section III, Cl. 1, Eq. 11) [Eq. 2-7 of Ref. [4]]* *Equation B-2*

where:

$K_1$, $K_2$, $K_3$ = local stress indices for the specific component under investigation

$E\alpha$ = modulus of elasticity *(E)* times the mean coefficient of thermal expansion ($\alpha$), both at room temperature, psi /°F (MPa/°C)

$|\Delta T_1|$ = absolute value of range of the temperature difference for each load set pair between the temperature of the outside surface $T_0$ and the temperature of the inside surface $T_i$ of the piping product assuming a moment generating equivalent linear temperature distribution, °F (°C)

$|\Delta T_2|$ = absolute value of range for that portion of the nonlinear thermal gradient through the wall thickness not included in $\Delta T_1$, °F (°C)

A load set pair is defined as two loading sets or cases, which are used to compute a stress range.

If Equation 10 cannot be satisfied for all load set pairs, the alternative analysis described below may still permit qualifying the component. Only those load set pairs which do not satisfy Equation 10 need to be considered.

1. Equation 12 shall be met:

$$S_e = C_2 \frac{D_o}{2I} M_i * \leq 3S_m$$

*Section III, C1. 1, Eq. 12) [Eq. 2-8 of Ref. [4]]* *Equation B-3*

where:

$S_e$ = nominal value of expansion stress, psi (MPa)

$M_i^*$ = same as $M_i$ in Equation 10, except that it includes only moments due to thermal expansion and thermal anchor movements, in-lb (N-mm)

2. The primary plus secondary membrane plus bending stress intensity, excluding thermal bending and thermal expansion stresses, shall be $<3S_m$. This requirement is satisfied by meeting Equation 13:

$$C_1 \frac{P_o D_o}{2t} + C_2 \frac{D_o M_i}{2I} + C'_3 E_{ab} \left| \alpha_a T_a - \alpha_b T_b \right| \le 3S_m$$

*(Section III, Cl. 1, Eq. 13) [Eq. 2-9 of Ref. [4]]* *Equation B-4*

where:

$C'_3$ = stress index

3. If these conditions are met, the value of $S_{alt}$ (also called $S_a$ later in the Code and in this document) shall be calculated by Equation 14:

$$S_{alt} = K_e \frac{S_p}{2}$$

*(Section III, C1. 1, Eq. 14) [Eq. 2-10 of Ref. [4]]* *Equation B-5*

Where from [4]:

$S_{alt}$ = alternating stress intensity, psi (MPa)

$S_p$ = peak stress intensity value calculated by Equation 11, psi (MPa)

$K_e$ = $1.0$ *for* $S_n \le 3S_m$ [Eq. 2-11 of Ref. [4]]

= $1.0 + [(1-n)/n(m-1)](S_n/3S_m - 1)$,

*for* $3S_m < S_n < 3mS_m$ [Eq. 2-12 of Ref. [4]]

= $1/n$, *for* $S_n \ge 3mS_m$ [Eq. 2-13 of Ref. [4]]

$S_n$ = primary plus secondary stress intensity value calculated in Equation 10, psi (MPa)

$m, n$ = material parameters

The alternating stress for all load set pairs is then computed as one-half of the peak stress intensity values adjusted by $K_e$. The fatigue analysis is then performed using the applicable Code fatigue curve and the number of design cycles for each load case from the Design Specification.

**The Electric Power Research Institute Inc.,** (EPRI, www.epri.com) conducts research and development relating to the generation, delivery and use of electricity for the benefit of the public. An independent, nonprofit organization, EPRI brings together its scientists and engineers as well as experts from academia and industry to help address challenges in electricity, including reliability, efficiency, health, safety and the environment. EPRI also provides technology, policy and economic analyses to drive long-range research and development planning, and supports research in emerging technologies. EPRI's members represent more than 90 percent of the electricity generated and delivered in the United States, and international participation extends to 40 countries. EPRI's principal offices and laboratories are located in Palo Alto, Calif.; Charlotte, N.C.; Knoxville, Tenn.; and Lenox, Mass.

Together...Shaping the Future of Electricity

***Programs:***

Nuclear Power

Boiling Water Reactors

Pressurized Water Reactors

Advanced Nuclear Technology



1024995

**Electric Power Research Institute**
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

**Attachment 3**

**NRC Staff's License Renewal Team Inspection Report 05000247/2013010**

**September 19, 2013**

**(ML13263A020)**

**Publicly available at:**

**http://pbadupws.nrc.gov/docs/ML1326/ML13263A020.pdf**

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BOULEVARD, SUITE 100
KING OF PRUSSIA, PENNSYLVANIA 19406-2713

September 19, 2013

Mr. John Ventosa
Site Vice President
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT 2 – NRC LICENSE RENEWAL TEAM INSPECTION REPORT 05000247/2013010

Dear Mr. Ventosa:

On September 12, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Indian Point Nuclear Generating Unit 2. The enclosed inspection report documents the results of our review of your completed actions for the remaining 10 commitments, which were discussed on September 12, 2013, with you and members of your staff.

The inspectors examined activities conducted by your staff to complete commitments Entergy made as part of your application for a renewed facility operating license. The inspectors also reviewed selected procedures and records, observed activities, and interviewed personnel. This inspection was conducted to follow-up on several commitments that were determined to merit additional inspection during a previous NRC License Renewal Team Inspection.

No findings were identified during this inspection. The NRC determined that the commitments reviewed associated with the license renewal application had been appropriately implemented.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

***/RA/***

James M. Trapp, Chief
Engineering Branch 1
Division of Reactor Safety

Docket No. 50-247
License No. DPR-26

Mr. John Ventosa
Site Vice President
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT 2 – NRC LICENSE RENEWAL TEAM INSPECTION REPORT 05000247/2013010

Dear Mr. Ventosa:

On September 12, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Indian Point Nuclear Generating Unit 2. The enclosed inspection report documents the results of our review of your completed actions for the remaining 10 commitments, which were discussed on September 12, 2013, with you and members of your staff.

The inspectors examined activities conducted by your staff to complete commitments Entergy made as part of your application for a renewed facility operating license. The inspectors also reviewed selected procedures and records, observed activities, and interviewed personnel. This inspection was conducted to follow-up on several commitments that were determined to merit additional inspection during a previous NRC License Renewal Team Inspection.

No findings were identified during this inspection. The NRC determined that the commitments reviewed associated with the license renewal application had been appropriately implemented.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

***/RA/***

James M. Trapp, Chief
Engineering Branch 1
Division of Reactor Safety

Docket No. 50-247
License No. DPR-26

Distribution: See Next Page

DOCUMENT NAME: G:\DRS\Engineering Branch 1\-- Meyer\Indian Pt Sept\20130912 05000247 010 IPEC Commitment Inpsection Final Report.docx
ADAMS ACCESSION NUMBER: ML13263A020

| ☑ SUNSI Review | | ☑ Non-Sensitive<br>☐ Sensitive | | ☑ Publicly Available<br>☐ Non-Publicly Available | |
|---|---|---|---|---|---|
| OFFICE | RI/DRS | RI/DRP | RI/DRS | | |
| NAME | GMeyer/MM | ABurrit | JTrapp | | |
| DATE | 9/18/13 | 9/19/13 | 9/19/13 | | |

Enclosure:
Inspection Report 05000247/2013010
w/Attachment: Supplementary Information

cc w/encl: Distribution via ListServ

<u>Distribution w/encl:</u> (via E-mail)
W. Dean, RA **(R1ORAMAIL RESOURCE)**
D. Lew, DRA **(R1ORAMAIL RESOURCE)**
D. Roberts, DRP **(R1DRPMAIL RESOURCE)**
M. Scott, DRP **(R1DRPMAIL RESOURCE)**
R. Lorson, DRS (**R1DRSMAIL RESOURCE)**
J. Rogge, DRS **(R1DRSMAIL RESOURCE)**
A. Burritt, DRP
T. Setzer, DRP
S. McCarver, DRP
L. McKown, DRP
S. Stewart, DRP, SRI
K. Dunham, DRP, RI
Ami Patel, DRP, RI
D. Hochmuth, DRP, AA
D. Rich, RI OEDO
RidsNrrPMIndianPoint Resource
RidsNrrDorlLpl1-1 Resource
ROPReports Resources

**U.S. NUCLEAR REGULATORY COMMISSION**

**REGION I**

| | |
|---|---|
| Docket No.: | 50-247 |
| License No.: | DPR-26 |
| Report No.: | 05000247/2013010 |
| Applicant: | Entergy Nuclear Northeast (Entergy) |
| Facility: | Indian Point Energy Center Unit 2 |
| Location: | 450 Broadway<br>Buchanan, NY 10511-0249 |
| Dates: | September 9-12, 2013 |
| Inspectors: | G. Meyer, Senior Reactor Inspector<br>M. Modes, Senior Reactor Inspector |
| Approved By: | James M. Trapp, Chief<br>Engineering Branch 1<br>Division of Reactor Safety |

## SUMMARY OF FINDINGS

IR 05000247/2013010; 09/09/2013 – 09/12/2013; Indian Point Nuclear Generating Unit 2; License Renewal Inspection.

This report covers an announced one week inspection, using the guidance provided in NRC inspection procedure Temporary Instruction 2516/001, “Review of License Renewal Activities,” of activities conducted by Entergy to complete commitments made to the NRC as a part of the Indian Point Energy Center, Unit 2, application for a renewed operating license. The commitments reviewed during this inspection are recorded in Supplement 1 to NUREG-1930, “Safety Evaluation Report Related to the License Renewal of Indian Point Generating Units Numbers 2 and 3,” Attachment 1, dated August 2011, and in other related correspondence.

**Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity**

No findings were identified. The inspectors concluded Entergy had made sufficient progress to complete our review of 44 commitments.

## REPORT DETAILS

## 4. OTHER ACTIVITIES

4OA5 Review of License Renewal Activities (TI 2516/001)

.1 Background

The expiration date of the operating license for Indian Point Unit 2 is midnight on September 28, 2013. Indian Point Unit 2 meets the criteria in Title 10 of the *Code of Federal Regulations* (10 CFR) 2.109(b), "Effect of timely renewal application," and will likely operate beyond the current operating license expiration date. Due to the Commission's decision to revise the Waste Confidence Decision and Rule and because of the ongoing Atomic Safety and Licensing Board hearings, the Commission is not expected to issue a renewed license for Indian Point Unit 2 before the expiration date of the original license. Therefore, Indian Point would continue operations under the timely renewal provisions of 10 CFR 2.109(b).

The team used NRC Inspection Manual Temporary Instruction 2516/001 to conduct this inspection. The Temporary Instruction was written specifically for plants like Indian Point Unit 2, where the holders of an operating license meet the criteria of 10 CFR 2.109, for timely renewal, but a final decision by the NRC on the license renewal application is not expected prior to the period of extended operation. The inspection objectives and requirements of the Temporary Instruction are to report the status of license renewal commitment implementation, the status of aging management program implementation, and to verify the description of programs and activities for managing the effects of aging are consistent with the Updated Final Safety Analysis Report.

The NRC has conducted three separate license renewal inspections that have reviewed a total of 44 license renewal commitments. Our first license renewal inspection conducted during a refueling outage, reviewed four commitments as documented in NRC Inspection Report 05000247/2012008 (ML12110A315). On May 23, 2013, the NRC completed a License Renewal Commitment Team Inspection, as documented in NRC Inspection Report 05000247/2013009 (ML13186A179). The team inspection concluded that Entergy had made sufficient progress to complete review of 30 commitments and identified 11 commitments that merited further assessment during a planned follow-up inspection. During the current planned follow-up inspection, the inspectors completed our review of ten of the 11 commitments identified by the team as requiring additional review. Commitment 47, one of the 11 commitments previously identified for additional review, was not assessed during this inspection because Entergy revised the completion date for this commitment to March 1, 2015, in a letter to the NRC (ML13142A202).

.2 Commitment Reviews

.2.1 Commitment 6: Enhance the Fatigue Monitoring Program to monitor steady state cycles and feedwater cycles or perform an evaluation to determine monitoring is not required. Review the number of allowed events and resolve discrepancies between reference documents and monitoring procedures.

a. Inspection Scope

In the prior inspection, the inspectors noted that Entergy had awarded contracts to perform calculations to determine whether monitoring of steady state cycles and feedwater cycles was required. The inspectors also noted that Entergy planned to revise procedure 2-PT-2Y015, Thermal Cycle Monitoring Program, if the calculations demonstrated that a change in the number of allowable steady state cycles and feedwater cycles was identified.

During this inspection, the inspectors reviewed Westinghouse calculation IPP-13-20, Revision 1, dated August 14, 2013, reporting the "Steady State Fluctuations and Feedwater Cycling Transient Disposition," presenting LTR-PAFM-13-87, Revision 2. This result determined that both transients do not significantly affect the fatigue of the primary system and can be removed from the transient cycle counting program. It was determined the feedwater cycle transient must still be tracked for the secondary side of the steam generator and feedwater piping, with a 25,000 cycle limit. The inspectors verified that procedure 2-PT-2Y015, "Thermal Cycle Monitoring Program," Revision 4, tracked the feedwater cycle transients.

Findings and Observations

No findings were identified.

.2.2 Commitment 13: Enhance the Metal-Enclosed Bus (MEB) Inspection Program to add a 480 volt bus, visually inspect the external surface of MEB enclosure assemblies, include acceptance criteria, inspect bolted connections, and remove reference to "re-torquing" connections from the applicable site procedure.

a. Inspection Scope

In the prior inspection, the inspectors identified that Entergy had not included all accessible portions of the MEB within the scope of the maintenance inspection program. The inspectors noted that sections of the emergency diesel generator 480 volt MEB in the electrical tunnel had not been visually inspected and were not included in the scope of the maintenance procedure which performed the MEB inspections and tests. As a result of the NRC's observations, Entergy initiated a Condition Report (CR-IP2-2013-01786) to revise site procedures and conduct visual inspections of those additional sections of the bus ducts prior to the period of extended operations.

During this inspection the inspectors reviewed the revised inspection and test procedures for the portions of the MEB previously considered to be inaccessible. The

inspectors reviewed the work orders under which the inspections and tests were completed for the three MEB sections and corrective action documents which resolved any identified conditions.

Findings and Observations

No findings were identified.

.2.3 Commitment 19: Implement the One-Time Inspection Program as described in License Renewal Application, Section B.1.27. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XIM32, One-Time Inspection.

a. Inspection Scope

In the prior inspection, the inspectors noted the One-Time Inspection Program involved over 400 inspections. Because Entergy had only completed approximately half of the planned inspections, the inspectors determined that additional NRC review was merited to assess the remaining inspection results; any further actions needed, and program conclusions.

During this inspection the inspectors reviewed the One-Time Inspection Summary Report, the completed inspection tracking matrix, and 20 additional inspection reports. The inspectors determined that all inspection results were acceptable and there was no need for additional action. Further, Entergy concluded that the inspection results demonstrated that the existing aging management programs for water chemistry and the diesel fuel monitoring and oil analysis had been effective in managing aging. The test program results also indicated that additional inspections on components specified in the License Renewal Application had not identified any unacceptable degradation.

Findings and Observations

No findings were identified.

.2.4 Commitment 23: Implement the Selective Leaching Program as described in License Renewal Application, Section B.1.33. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XIM33 Selective Leaching of Materials.

a. Inspection Scope

In the prior inspection, the inspectors noted Entergy was in the progress of completing selective leaching inspections. The inspectors determined that additional NRC inspection was merited to review the results of the remaining inspections, any further actions needed, and program conclusions.

During this inspection, the inspectors reviewed Selective Leaching Summary Report, including the final sampling plan, the destructive evaluation results of seven

components, and associated corrective action documents. Entergy determined that the visual inspections of 22 gray cast iron components and 17 copper alloy components showed an absence of selective leaching. However, destructive evaluations demonstrated that significant selective leaching (i.e., graphitization, had occurred in gray cast iron components) was occurring. While the commitment actions were complete, Entergy concluded that they would refine the process and documented a corrective action to perform engineering evaluations to develop a continuing monitoring program to manage and evaluate selective leaching.

Findings and Observations

No findings were identified

.2.5 Commitment 26: Implement the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program as described in License Renewal Application, Section B.1.37. This new program will be implemented consistent with the corresponding program described in NUREG 1801, Section XI.M12, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Program.

a. Inspection Scope

In the prior inspection, the inspectors noted that Entergy was planning to screen all cast austenitic stainless steel components to determine which were potentially susceptible to a loss of fracture toughness. These components were to be further evaluated, using a refined analytical technique, to determine the components susceptibility to reduction in fracture toughness. Entergy chose to perform a unique fracture mechanics analysis of these components, which was in progress at the time of the prior inspection. The inspector noted that there may be a question regarding the submittal of the analysis under ASME requirements, as stipulated in 10 CFR 50.55a.

Subsequently the inspector, in consultation with the Division of License Renewal, determined submittal of the analysis was not required. During this inspection the inspectors reviewed the completed evaluation of the screened components' susceptibility to reduction in fracture toughness: Report 1300066.403, Revision 0, "Aging Management of CASS Piping at Indian Point 2, Flaw Tolerance Evaluation of CASS Piping at Indian Point," dated August 2013. The methodology of probabilistic fracture mechanics determined a postulated starting reference flaw of one-quarter thickness would remain below the maximum allowable flaw size during a 60 year life for all components.

Findings and Observations

No findings were identified.

.2.6 Commitment 27: Implement the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program as described in License Renewal Application, Section B.1.38. This new program will be implemented consistent with the

corresponding program described in NUREG-1801, Section XI.M13, Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel program.

a. Inspection Scope

In the prior inspection, the inspectors noted a site document had not been developed that defined and implemented the screening criteria of Electric Power Research Institute (EPRI) Technical Report 1013234, "Materials Reliability Program: Screening Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design," (MRP-191), listed in Table 3-5, as applied to the Indian Point components listed in Table 5-1 of EPRI Report 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," (MRP-227-A).

During this inspection the inspectors reviewed Entergy Nuclear Engineering Report: "Indian Point Energy Center: Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel Aging Management Program," IP-RPT-13-00049, Revision 0, dated August 14, 2013. This report implemented the above screening criteria and ranking methodology for Indian Point.

Findings and Observations

No findings were identified.

.2.7 Commitment 33: For the locations identified in License Renewal Application, Table 4.3-13 (IP2), update the fatigue usage calculations using refined fatigue analyses to determine if Cumulative Usage Factors (CUF) remain less than 1.0 when accounting for the effects of reactor water environment, using valid Fen factors.

a. Inspection Scope

In the prior inspection, the inspectors noted that calculations documented in calculation WCAP 17149-P, "Evaluation of Pressurizer Insurge/Outsurge Transients for Indian Point Unit 2," and WCAP 17199-P, "Environmental Fatigue Evaluation for Indian Point Unit 2," concluded that cumulative fatigue usage factors, including reactor water environment effects, were below the American Society of Mechanical Engineers (ASME) Code allowable value of 1.0 for transients postulated for 60 years of operation. The inspectors noted Entergy's action plan included revising the Thermal Cycle Monitoring Program procedure to reflect changes in the number of projected cycles used in WCAP 17199-P.

During this inspection the inspectors reviewed Indian Point Programs and Components Engineering Procedure 2-PT-2Y015, Revision 4, "Thermal Cycle Monitoring Program." The inspectors noted the procedure was updated to reflect the number of allowable cycles derived from the above analysis. The procedure referenced WCAP 17199-P and included a Table (Attachment 1) that included actual plant cycles rather than the number of cycles used in the original design calculations. The procedure also was revised to better reflect operational assumptions used and the bases for the revised calculations of the WCAP.

Findings and Observations

No findings were identified.

.2.8 Commitment 40: Evaluate plant specific and appropriate industry operating experience and incorporate lessons learned in establishing appropriate monitoring and inspection frequencies to assess aging effects for the new aging management programs. Documentation of the operating experience evaluated for each new program will be available on site for NRC review prior to the period of extended operation.

a. Inspection Scope

During the prior inspection, there was insufficient material to review this commitment. Subsequently, Entergy completed an operational review for the following new aging management programs embodied in separate commitments:

#3, Buried Piping and Tanks Inspection Program
#14, Non-EQ Bolted Cable Connections Program
#15, Non-EQ Inaccessible Medium-Voltage
#16, Non-EQ Instrumentation circuits Test Review Program
#17, Non-EQ Insulated Cables and Connections Program
#19, One-Time Inspection Program
#20, One-Time Small Bore Piping Program
#23, Selective Leaching Program
#26, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program
# 27, Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program.

During this inspection the inspectors reviewed a selected sample of the reviews and noted each of the programs was subject to an operational review which included Indian Point specific experience. For example, the Buried Piping Program for Unit 2 was assessed for a period of six years, 2007 through 2013. This period of time enveloped the operational experience included in the guidance documents, such as Generic Aging Lessons Learn (NUREG-1801), Revision 2. External operational experience considered included license event reports, NRC generic letters, NRC information notices, and Institute of Nuclear Power Operations (INPO) documents.

Findings and Observations

No findings were identified.

.2.9 Commitment 43: Indian Point Energy Center will review design basis ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 locations that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting locations for the Unit 2 and Unit 3 configurations. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the

reactor coolant environment on fatigue usage. Indian Point Energy Center will use the NUREG/CR-6909 methodology in the evaluation of the limiting locations consisting of nickel alloy, if any.

a. Inspection Scope

In the prior inspection, the inspectors noted that Entergy had awarded contracts to perform calculations to support closure of this commitment. Because the results of the calculations were not available at the time, the inspectors deferred inspection of this commitment.

During this inspection the inspectors reviewed calculation CN-PAFM-13-32, Revision 0, "Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations." This calculation was the evaluation of locations previously screened by calculation CN-PAFM-12-35, Revision 1, "Indian Point Unit 2 and Unit 3 EAF Screening Evaluations," that could be more limiting than the locations identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plants Components." The inspectors noted the $CUF_{en}$ result for the pressurizer nozzle was 0.999 at 60 years. Entergy was aware that accumulation of cycles at a rate greater than assumed in the calculation would require a more refined analysis, application of a non-destructive monitoring technique, or replacement of the pressurizer nozzle.

Findings and Observations

No findings were identified.

.2.10 Commitment 48: Entergy will visually inspect in-scope underground piping prior to the period of extended operation and then on a frequency of at least once every 2 years during the period of extended operation. Visual inspections will be supplemented with surface or volumetric non-destructive testing if indications of significant loss of material are observed. Adverse indications will be entered into the plant corrective action program for evaluation of extent of condition and for determination of appropriate corrective actions (e.g., increased inspection frequency, repair, or replacement).

a. Inspection Scope

In the prior inspection the inspectors noted that work orders had been issued to perform the inspections prior to the period of extended operation. The inspectors determined that additional inspection was merited regarding review of the results of the underground piping inspections.

The inspectors reviewed the results of selected inspections performed subsequent to the May 23, 2013, NRC inspection. EN-EP-S-002-MULTI, Attachment 7.2 "Pipe/Tank Coating Visual Inspection Checklist" was reviewed for the following lines:

21-EDGE-2/EDG FOST 3-inch equalizing line
21-EDGE-2/EDG FOST 4-inch fill line
21-EDGE-2/EDG FOST 4-inch vent line
22-EDGE-2/EDG FOST 3-inch equalizing line
22-EDGE-2/EDG FOST 4-inch fill line
22-EDGE-2/EDG FOST 4-inch vent line
23-EDGE-2/EDG FOST 3-inch equalizing line
23-EDGE-2/EDG FOST 4-inch fill line
23-EDGE-2/EDG FOST 4-inch vent line

The check list included the inspection of the piping for aging affects such as mechanical damage, coating breaks (referred to as a holiday), and blistering.

Findings and Observations

No findings were identified.

4OA6 Meetings, Including Exit

On September 12, 2013, the inspectors presented the inspection results to Mr. John Ventosa, Site Vice President, and other members of the Entergy staff. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

ATTACHMENT: SUPPLEMENTARY INFORMATION

## SUPPLEMENTARY INFORMATION

## KEY POINTS OF CONTACT

Entergy Personnel

J. Ventosa, Site Vice President
N. Azevedo, Code Programs Supervisor
C. Caputo, License Renewal Team
J. Curry, Senior Project Manager
G. Dahl, Licensing Engineer
P. Guglielmino, Implementation Team Manager
L. Lubrano, Component Electrical Engineer
R. Sporbert, One-Time Inspection Coordinator

## LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Closed

The inspectors determined that the following 10 commitments had been appropriately implemented:

| | |
|---|---|
| 6 | Fatigue cycles analysis |
| 13 | Metal enclosed bus inspection |
| 19 | One-time inspection |
| 23 | Selective leaching inspection |
| 26 | Embrittlement of CASS analysis |
| 27 | Embrittlement of CASS analysis |
| 33 | Fatigue monitoring |
| 40 | Operating experience for new programs |
| 43 | Fatigue monitoring analysis |
| 48 | Underground piping inspection |

## LIST OF ACRONYMS

| | |
|---|---|
| ADAMS | Agencywide Documents Access Management System |
| ASME | American Society of Mechanical Engineers |
| CASS | Cast Austenitic Stainless Steel |
| CFR | Code of Federal Regulations |
| CUF | Cumulative Usage Factor |
| ENTERGY | Entergy Nuclear Northeast |
| EPRI | Electric Power Research Institute |
| INPO | Institute of Nuclear Power Operations |
| IPEC | Indian Point Energy Center |
| LRA | License Renewal Application |
| MEB | Metal-Enclosed Bus |

| | |
|---|---|
| MRP | Materials Reliability Project |
| NRC | Nuclear Regulatory Commission |
| UFSAR | Updated Final Safety Evaluation Report |

## LIST OF DOCUMENTS REVIEWED

Commitment 6, 33, and 43

2-PT-2Y015, Revision 3, Thermal Cycle Monitoring Program
IP-RPT-11-LRD13, Revision 0, Review of the Fatigue Monitoring Aging Management Program for License Renewal Implementation
EN-LI-100, Revision 13, Process Applicability Determination
EN-AD-101, Revision 16, Procedure Process
WCAP-17199-P, July 2010, Environmental Fatigue Evaluation for Indian Point Unit 2
WCAP-17149-P, July 2010, Evaluation of Pressurizer Insurge/Outsurge Transients for Indian Point Unit 2
Entergy Letter dated August 9, 2010 (NL-10-82), License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program Indian Point Nuclear Generating Unit Nos. 2 and 3 [ML102300504]

Commitment 13

2-ELC-016-BUS, Inspection, Cleaning and Testing of 480V Buses, Revision 2
CR-IP2-2009-03029, Water dripping at the bend in the electric tunnel
CR-IP2-2012-01903, Bus 5A surface rust noted on the interior divider panel
Work Order 52293872, Inspection of Bus 5A Switchgear and Station Service Transformer
Work Order 52294517, Inspection of Bus 5A (480 V Switchgear to EDG)
2-ELC-016-BUS, Revision 4, Inspection, Cleaning and Testing of 480V Buses
2-ELC-403-BUS, Revision 7, Inspection and Cleaning of 480 Volt Bus Duct
Work Order 351381, 21 EDG Bus Visual, Cleaning, Bolted Checks in Electrical Tunnel, completed on July 18, 2013
Work Order 351382, 22 EDG Bus Visual, Cleaning, Bolted Checks in Electrical Tunnel, completed on June 12, 2013
Work Order 351448, 23 EDG Bus Visual, Cleaning, Bolted Checks in Electrical Tunnel, completed on August 12, 2013
CR-IP2-2013-01738, MEB acceptance criteria
CR-IP2-2013-01748, Leaks in Unit 2 electrical tunnel
CR-IP2-2013-02375, 22 EDG bus inspection
CR-IP2-2013-02923, 21 EDG bus inspection
CR-IP2-2013-03330, 23 EDG bus inspection
CR-IP2-2013-02912, Thermography procedures
CR-IP2-2013-02913, Water intrusion into Unit 2 electrical tunnel

Commitment 19

IP-RPT-11-LRD28, Revision 0, Review of the One-Time Inspection Program
EN-FAP-LR-024, Revision 1, One-Time Inspection
NL-13-046, Amendment 13 to LRA for One-Time Inspection and Selective Leaching Programs, March 18, 2013
IPEC Unit 2 One-Time Inspection Tracking Matrix, May 3, 2013 and May 21, 2013
20 Inspection reports for one-time inspections
IP-RPT-13-LRD03, Revision 0, Unit 2 License Renewal One-Time Inspection Summary Report
IPEC Unit 2 One-Time Inspection Tracking Matrix, August 28, 2013
20 Additional inspection reports for one-time inspections

Commitment 23

IP-RPT-11-LRD34, Revision 0, Review of the Selective Leaching Program
EN-FAP-LR-02, Revision 3, Selective Leaching Inspection
NL-13-046, Amendment 13 to LRA for One-Time Inspection and Selective Leaching Programs, March 18, 2013
IPEC Unit 2 Selective Leaching Inspection Tracking Matrix, May 20, 2013
10 Inspection reports for copper-alloy selective leaching inspections
12 Inspection reports for gray cast iron selective leaching inspections
WO 00326036-01
WO 00326216-01
IP-RPT-13-LRD07, Revision 0, License Renewal Selective Leaching Inspection Summary Report
Altran 13-0313-TR-001, Laboratory Analysis of Several Valves for Selective Leaching, Revision 0
Altran 13-0313-TR-001, Laboratory Analysis of Several Valves for Selective Leaching, Revision 1
CR-IP2-2013-03037, Selective leaching of gray cast iron components
CR-IP2-2013-03360, Selective leaching of copper alloy components

Commitment 26

Entergy Letter, NL-09-018, “Reply to Request for Additional Information – Miscellaneous Items,” January 27, 2009
Entergy Letter NL-11-101, “Clarification for Additional Information (RAI) Aging Management Programs,” August 22, 2011
LR# 173, LR Request, Confirm each AMP will be implemented with ten elements
IP-RPT-11-LRD38, “Review the Thermal Aging Embrittlement of CASS Aging Management Program for License Renewal Implementation,” 1/2/2013
NRC Letter, “License Renewal Issue No. 98-0030, “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components,” May 19, 2000
WCAP-10977, Supplement 1, “Additional Information in Support of the Technical Justification for Eliminating Large Primary LOOP Pipe Rupture as the Structural Design Basis for Indian Point Unit 2,” January 1989

Commitment 27

IP-RPT-11-LRD39, Revision), "Review of the Thermal Aging & Neutron Embrittlement of CASS Aging Management Program for License Renewal Implementation," ED41109, 1/23/2013
EPRI 1013234 "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)," November 2006
EPRI 1022863 "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," December 2011
NRC Letter "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Nos. 2 and 3, License Renewal Application," May 15, 2012
Entergy Letter, NL-11-101, "Clarification for Request for Additional Information (RAI) Aging Management Programs," 8/22/2011
Entergy Letter NL-13-052, "Reply to Request for Additional Information Regarding the License Renewal Application," 5/7/2011
LR Request #173 Confirm new programs will be implemented consistent with 10 elements of NUREG-1801.
IP-RPT-11-LRD39 "Review of the Thermal Aging and Neutron Irradiation Embrittlement of CASS Aging Management Program for License Renewal Implementation," EC41109, 1/2/13

Commitment 48

Work Order PMRQ 00349816-01, 23 Fuel Oil Storage Tank Underground Piping Inspections
Work Order PMRQ 00349802-01, 21 Fuel Oil Storage Tank Underground Piping Inspections
Work Order PMRQ 00349814-01, 22 Fuel Oil Storage Tank Underground Piping Inspections
Work Order 00342492-01, Inspect Underground Piping by the IP2 EDG Building DF-2 Area
Work Order 00342493-01, Inspect Underground Piping by the IP2 EDG Building DF-2-1 Area
Work Order 00342494-01, Inspect Underground Piping by the IP2 EDG Building DF-2-2 Area

**Attachment 4**

**Summary of the February 19, 2015 public meeting in Rockville, Maryland to discuss reactor pressure vessel issues including a list of attendees**

**April 6, 2015**

**(ML15096A128)**

April 6, 2015

MEMORANDUM TO: John W. Lubinski Division of Engineering
Office of Nuclear Reactor Regulation

FROM: Robert O. Hardies, Sr. Level Advisor ***/RA by E-mail//***
Division of Engineering
Office of Nuclear Reactor Regulation

SUBJECT: SUMMARY OF FEBRUARY 19, 2015, PUBLIC MEETING TO DISCUSS REACTOR PRESSURE VESSEL ISSUES

On February 19, 2015, a Category 2 public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and representatives of industry to allow an exchange of information about reactor pressure vessel integrity issues. A portion of the meeting related to the staff's recent evaluation of potential non-conservatisms in NRC Branch Technical Position (BTP) 5-3, "Fracture Toughness Requirements." Other topics addressed included reactor vessel surveillance programs, Title 10, *Code of Federal Regulations*, Part 50, (10 CFR 50) Appendices G and H evaluations, status of the NRC Regulatory Guide for Alternate Pressurized Thermal Shock (PTS) Rule Implementation, a discussion of ASME Codes requirements for pressure testing while the reactor vessel is critical, and a status of NRC work activities on the Reactor Embrittlement Archive Project (REAP) Database.

The NRC staff made a two-part presentation on the recent NRC activities performed regarding the potential non-conservatism of BTP 5-3. The first part of the presentation provided a review of definitions and estimates of unirradiated reference temperature ($RT_{NDT}$) and unirradiated upper shelf energy (USE) addressed in BTP 5-3, summarized the background of the recent questions concerning the potential non-conservatism of BTP 5-3, and summarized the objectives of the staff's analysis of BTP 5-3. The second part of the staff's presentation provided an assessment of the impact of the potential non-conservatisms in BTP 5-3 on U.S. plants with regard to PTS and pressure temperature (P-T) limit evaluations based on recent docketed information. The staff concluded that one PWR PTS evaluation is potentially affected, some PWRs and BWRs P-T limits are potentially affected, and that there are no immediate safety concerns. The staff provided a tentative schedule for completion of a report that documents the staff's findings.

The industry followed with three presentations on their recent activities performed regarding the potential non-conservatism of BTP 5-3. The first presentation provided background on the industry focus groups that have been formed to address the BTP 5-3 issues, the membership of the focus groups, and the focus groups activities that are currently underway. The second presentation provided a summary of the Materials Reliability Project (MRP)/Boiling Water Reactor Vessel and Internals Project (BWRVIP) focus group activities and results to-date. The objectives of the focus group activities include conducting a survey regarding use of BTP 5-3 in the PWR and BWR fleets, evaluating the BTP 5-3 procedures which had previously been identified as potentially non-conservative, determining if application of BTP 5-3 for defining reactor vessel P-T limits provides adequate margins against failure through the 60-year license period (EOLE), and, if needed, recommending alternative procedures to ensure that adequate margins against failure are maintained through EOLE. The focus group

has completed a draft report that is under review with a target completion in June 2015. The focus group is considering similar work for a procedure developed by GE-Hitachi Nuclear Energy that is similar to BTP 5-3. The third presentation provided a summary of the Pressurized Water Reactor Owners Group (PWROG) activities regarding their Material Orientation Toughness Assessment (MOTA) for the purpose of mitigating BTP 5-3 uncertainties. The objective of the MOTA is to explore existing deterministic margin that may be potentially available in ASME Code Section XI, Appendix G and other NRC-approved sources to address potential non-conservatisms in BTP 5-3. The results of the industry's investigation to-date demonstrate that current methods for developing P-T limits are acceptable in light of the identified BTP 5-3 estimation uncertainties. The PWROG intends to document the MOTA in a final report later this year.

The NRC staff provided two presentations on the status of their 10 CFR 50 Appendix G research activities to-date. The first presentation provided a summary of overall activities to-date, and summarized the new version of the FAVOR computer code to address software errors previously identified by the industry. A release of FAVOR v15.1 that remedies the software errors is anticipated in Spring 2015. The second presentation summarized the staff's efforts regarding an advanced residual stress model under investigation for implementation into FAVOR. Further efforts are underway to explore other fracture mechanics models and to assess the ability to adopt an appropriate model into the FAVOR code at a later date.

The PWROG discussed an assessment of the margins associated with American Society of Mechanical Engineers (ASME) Section XI Appendix G pressure-temperature (P-T) limits for pressurized water reactor (PWR) nozzles. The assessment is intended to generically address 10 CFR 50 Appendix G requirements recently clarified by the NRC in Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components." This RIS clarifies that 10 CFR 50 Appendix G requires that P-T limits sufficiently address all ferritic materials of the reactor vessel, including the impact of structural discontinuities such as nozzles. The purpose of the industry's activity is to develop a basis for generically addressing nozzles supporting P-T limit submittals and to justify the use of the reactor vessel shell region with the highest embrittlement as the limiting region to be used for P-T limits.

The NRC staff presented their plans to begin rulemaking for revising 10 CFR 50 Appendix H based on recent commission approval to do so. The planned revisions will include changes to permit the use of updated and modern revisions of ASTM standards and, in response to industry requests, to allow an increase in the required time to submit surveillance program test results. The longer reporting time is intended to mitigate scheduler complications that modern integrated surveillance programs have experienced related to the logistics of withdrawal, shipping, testing and reporting results of testing of surveillance capsules.

The NRC staff presented a status and the latest tentative schedule for issuing a draft Regulatory Guide (RG) describing guidance for implementation of the Alternate Pressurized Thermal Shock Rule, 10 CFR 50.61a. The RG was published in the Federal Register for a 60-day public comment period on March 13, 2015.
The NRC staff presented an ASME Code item related to pressure testing with the reactor vessel core critical. Although NRC regulations in 10 CFR 50 specifically prevent pressure

testing of the pressure vessel while the core is critical, ASME Section XI repair/replacement activities allow pressure testing of other non-RPV Class 1 components while the reactor vessel core is critical. The staff is considering additional actions to clarify the NRC position on this item.

The NRC staff discussed the Reactor Embrittlement Archive Project. This project catalogues historical records of surveillance data and research irradiation information.

Action items captured during the meeting were as follows:

1. NRC and industry will explore opportunities to share longitudinal and transverse Charpy data sources to facilitate data verification.
2. The NRC will evaluate the need to request and review the EPRI sponsored evaluation of BTP 5-3 conservatism.
3. The NRC will evaluate the need to request and review the PWROG report on MOTA.
4. The NRC and industry will evaluate further communication opportunities at ASME Code meetings.

A list of attendees is enclosed. The slide presentations presented by the NRC staff and the industry representatives can be found in the Agencywide Documents Access and Management System (ADAMS) at Accession Number ML15061A072.

Enclosure:
List of Attendees

The NRC staff presented an ASME Code item related to pressure testing with the reactor vessel core critical. Although NRC regulations in 10 CFR 50 specifically prevent pressure testing of the pressure vessel while the core is critical, ASME Section XI repair/replacement activities allow pressure testing of other non-RPV Class 1 components while the reactor vessel core is critical. The staff is considering additional actions to clarify the NRC position on this item.

The NRC staff discussed the Reactor Embrittlement Archive Project. This project catalogues historical records of surveillance data and research irradiation information.

Action items captured during the meeting were as follows:

5. NRC and industry will explore opportunities to share longitudinal and transverse Charpy data sources to facilitate data verification.
6. The NRC will evaluate the need to request and review the EPRI sponsored evaluation of BTP 5-3 conservatism.
7. The NRC will evaluate the need to request and review the PWROG report on MOTA.
8. The NRC and industry will evaluate further communication opportunities at ASME Code meetings.

A list of attendees is enclosed. The slide presentations presented by the NRC staff and the industry representatives can be found in the Agencywide Documents Access and Management System (ADAMS) at Accession Number ML15061A072.

Enclosure:
List of Attendees

DISTRIBUTION:

| | | | | | |
|---|---|---|---|---|---|
| PUBLIC | SRosenberg | GStevens | JPoehler | MKirk | SSheng |
| RidsResDe | RidsNrrDe | CFairbanks | CNove | DRudland | RTregoning |

EXTERNAL DISTRIBUTION:
ademma@epri.com, rdyle@epri.com, thardin@epri.com, bernie.rudell@cengllc.com
Andrew.odell@exeloncorp.com, Npalm@epri.com.

**ADAMS Accession Nos.:**

**Package: ML15061A072**

**Meeting Summary: ML15096A128 * By E-mail**

| | |
|---|---|
| **OFFICE** | NRR/DE * |
| **NAME** | RHardies |
| **DATE** | 04/06/2015 |

**OFFICIAL RECORD COPY**

**List of Attendees**

**Public Meeting with U.S. Nuclear Regulatory Commission (NRC) Staff to Discuss Reactor Pressure Vessel Issues**

**February 19, 2015**

| Name | Affiliation | Email |
|---|---|---|
| MARK KIRK | NRC/RES/DE/CIB | MARK.KIRK@NRC.GOV |
| Bryan Hall | Westinghouse | hallj;b@westinghouse.com |
| Dave Rudland | NRC/RES/DE/CIB | david.rudland@nrc.gov |
| DILIP DEDHIA | SIA - San Jose | DDEDHIA@STRUCTINT.COM |
| Tim Griesbach | SIA - San Jose | tgriesbach@structint.com |
| Anne Demma | EPRI | ademma@epri.com |
| Jeff Poehler | NRC/NRR/DE/EVIB | jeffrey.poehler@nrc.gov |
| Carolyn Fairbanks | NRC/NRR/DE/EVIB | carolyn.fairbanks@nrc.gov |
| Tim Wells | Southern Nuclear | tgwells@southernco.com |
| Bernie Rudell | Exelon Generation LLC | bernie.rudell@exeloncorp.com |
| Drew Odell | Exelon Generation | andrew.odell@exeloncorp.com |
| Christopher Koehler | Xcel Energy/PWROG | christopher.koehler@xenuclear.com |
| KENNETH R. BAKER | ENTERGY NUCLEAR | kbaker@entergy.com |
| STEVEN M. POPE | NuScale Power | spope@nuscalepower.com |
| Jim Andrachek | W | andracjd@westinghouse.com |
| Warren Bamford | W | bamforwh@westinghouse.com |
| Ashok Nana | AREVA | ashoknana@areva.com |
| Matthew DeVan | AREVA | matt.devan@areva.com |
| Simon Sheng | NRC/NRR/DE/EVIB | cfs@nrc.gov |
| Charles Tomes | Dominion | charles.a.tomes@dom.com |
| Allen Hiser | NRC/NRR/DLR | Allen.Hiser@NRC.GOV |
| Matthew A. Mitchell | NRC/NRO/DE | MAM4@NRC.GOV |
| DAN HEZIT | FIRST ENERGY | DHEZIT@FIRSTENERGYCORP.COM |
| GREG KAMMERDEINER | FIRST ENERGY | kammerdeinerg@firstenergycorp.com |
| Gary Stevens | NRC | gary.stevens@nrc.gov |
| Stacey Rosenberg | NRC | stacey.rosenberg@nrc.gov |
| BILL SERVER | ATI CONSULTING | wserver@ati-consulting.com |
| Tim HARDIN | EPRI | thardin@epri.com |
| Brian Frew | GE-Hitachi | brian.frew@ge.com |
| GEORGE DEPTA | GE HITACHI | GEORGE.DEPTA@GE.COM |
| Bob Hardies | NRC | roh@nrc.gov |

## List of Attendees (concluded)

**Public Meeting with U.S. Nuclear Regulatory Commission (NRC) Staff to Discuss Reactor Pressure Vessel Issues**

**February 19, 2015**

| Name | Affiliation | Email |
|---|---|---|
| Ron Gamble | Sartrex Corp | SARTREX@aol.com |
| Nathan Palm | EPRI | npalm@epri.com |
| Rob Tregoning | NRC | robert.tregoning@nrc.gov |
| NELSON AZEVEDO | ENTERGY | NAZEVED@ENTERGY.COM |
| RAY KUYLER | MORGAN LEWIS | rkuyler@morganlewis.com |
| David Whitaker | Duke Energy | David.Whitaker@duke-energy.com |
| ROBIN DYLE | EPRI | RDYLE@EPRI.COM |
| Elliot J. Long | Westinghouse | longej@westinghouse.com |
| Benjamin A. Rosier | Westinghouse | rosierba@westinghouse.com |
| FRANK GIFT | WESTINGHOUSE | giftfc@westinghouse.com |
| Brian Lusignan | State of New York | Brian.Lusignan@ag.ny.gov |
| MJ Ross-Lee | NRR/DE | maryjane.ross-lee@nrc.gov |
| Pat Purtscher | NRR/DE/EVIB | ptp1@nrc.gov |
| David Dijamco | NRR/DE/EVIB | David.Dijamco@nrc.gov |
| Austin Young | NRR/DE/EVIB | Austin.Young@NRC.GOV |
| KRISHAN GARG | PSEG NUCLEAR | Krishan.Garg@PSEG.com |
| Jim Cirilli | EXELON | James.Cirilli@EXELONCORP.COM |
| R. Scott Boggs | FPL/Nextera | scott.boggs@fpl.com |

ENCLOSURE

**Attachment 5**

**E-mail from Hearingdocket@nrc.gov to Brian Lusignan**
**Notification of the Non-public Filing of the Joint Brief of**
**Entergy and Westinghouse Re Proprietary Documents**
**June 4, 2015**

**From:** Hearingdocket@nrc.gov
**To:** pbessette@morganlewis.com; wglew@entergy.com; brian.harris@nrc.gov; hearingdocket@nrc.gov; rkuyler@morganlewis.com; Lisa S. Kwong; Brian Lusignan; lawrence.mcdade@nrc.gov; OCAAMAIL@nrc.gov; martin.oneill@morganlewis.com; drepka@winston.com; John J. Sipos; ksutton@morganlewis.com; sherwin.turk@nrc.gov; richard.wardwell@nrc.gov; alana.wase@nrc.gov; edward.williamson@nrc.gov
**Subject:** Re: NRC Proceeding "Indian Point 50-247-LR and 50-286-LR"
**Date:** Thursday, June 04, 2015 3:52:15 PM

---

MESSAGE FROM THE OFFICE OF THE SECRETARY, NUCLEAR REGULATORY COMMISSION REGARDING LIMITED ACCESS PROTECTED DOCUMENT

Re: NRC Proceeding "Indian Point 50-247-LR and 50-286-LR" The Office of the Secretary has received a limited access protected document entitled

"Joint Brief of Entergy and Westinghouse Re Proprietary Documents"

submitted by David A Repka who is affiliated with Winston and Strawn LLP.

It is intended for inclusion in the referenced proceeding and was submitted through the NRC Electronic Information Exchange (EIE) system. It arrived on 06/04/15 at 15:52 EDT.

The submitter has designated you as a hearing participant who is eligible to view this/these document(s). You are entitled to view and/or retrieve/save this/these document(s) by visiting the following web link(s):

Joint Brief of Entergy and Westinghouse re Proprietary Documents - https://eieprod.nrc.gov/EIE25L3/downloadAttachment.do?submissionID=50749&docID=1172 (1998 KB)

This/these document(s) is/are subject to special handling and limited distribution pursuant to an adjudicatory directive. The document(s) will remain available to you through the link(s) above for 14 days after which it/they will be removed from the E-Filing system. After 3 business days, you may also access the document(s), along with any other protected document(s) to which you have been authorized access, through the "Access Authorized Protected Documents" selection on the following page: https://eieprod.nrc.gov/EIE25/portal.do or by going directly to the Electronic Hearing Docket at: https://adams.nrc.gov/ehd

Receipt of this message constitutes completion of service of this filing.

**Attachment 6**

**E-mail from Brian Lusignan to attorneys for Westinghouse, Entergy and NRC Staff regarding the apparent failure of Westinghouse and Entergy to properly file the Joint Brief publicly**

**June 15, 2015**

| | |
|---|---|
| **From:** | Brian Lusignan |
| **To:** | "drepka@winston.com"; "Sutton, Kathryn M."; "Bessette, Paul M." |
| **Cc:** | John J. Sipos; Lisa S. Kwong; Turk, Sherwin |
| **Subject:** | Nonpublic Filing of Joint Brief |
| **Date:** | Monday, June 15, 2015 11:48:39 AM |

Counsel:

The State is in the process of preparing its response to the joint brief of Westinghouse and Entergy regarding the proprietary designation of certain documents in the Indian Point relicensing proceeding. Recently, we noticed that although the joint brief has not been designated as containing proprietary information pursuant to 10 C.F.R. section 2.390 (b) and the ASLB's September 4, 2009 Protective Order, paragraphs A and K, it was served via the NRC's non-public electronic information exchange (EIE) on a limited number of hearing participants. The Certificate of Service that accompanied the joint brief does not reference the non-public EIE exchange; instead, the Certificate of Service states that the brief was served on “those on the EIE Service List for the captioned proceeding,” indicating that the joint brief should have been served on all parties and interested governmental entities in the proceeding. The State respectfully requests that you re-file the joint brief on the public EIE exchange, so that all hearing participants may access it.

Very truly yours,

Brian Lusignan
Assistant Attorney General
NYS Office of the Attorney General
Environmental Protection Bureau
The Capitol
Albany NY 12224-0341
(518) 776-2399

**Please note the new phone number.**