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Environmentally Assisted Fatigue Screening Methods (Revision 1)

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ABSTRACT

This report provides the technical bases and processes for three environmentally assisted fatigue (EAF) screening methods for use in nuclear power plants. These screening methods are intended to identify appropriate limiting locations for systematic management of EAF effects on Class 1 reactor coolant pressure boundary components that are wetted with primary coolant. Use of any of these methods will ensure that the most limiting locations for EAF are determined on a consistent basis. Note, however, that there may also be other screening methods not covered in this report that may also be acceptable for use in determining plant-specific limiting locations to evaluate EAF.

The three methods documented in this report each provide guidance for the evaluation and relative ranking of estimated EAF cumulative usage factor, CUF_{en}, values for locations in components and systems where EAF must be addressed to minimize the number of locations required for explicit fatigue evaluation and/or management. The methods will enable plant owners to demonstrate knowledge of the locations in their plants that can serve as locations requiring EAF management to satisfy long-term operation EAF guidance. The EAF screening methods provide the rationale for selecting these locations. Plant owners can minimize costs by avoiding the necessity of formal fatigue analyses and reducing the overall number of component locations requiring EAF assessment while meeting the regulatory requirements to determine those EAF locations that are limiting in their plants. The EAF screening methods each provide a uniform approach for determining the limiting locations that require EAF assessment and management throughout extended plant operating periods.

This report is a public document available for reference by nuclear power plant owners pursuing extended plant operation. This report updates and supersedes EPRI report 1024995.

Keywords

Design basis
Environmentally assisted fatigue
Fatigue management
Fatigue usage
License renewal
Long-term operation
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PRIMARY AUDIENCE: Nuclear utility staff working on component fatigue evaluations

SECONDARY AUDIENCE: Nuclear utility staff working on long-term operation applications

KEY RESEARCH QUESTION

What are some environmentally assisted fatigue (EAF) screening processes that may be used to satisfy regulatory guidance to identify limiting plant-specific component locations in the reactor coolant pressure boundary?

RESEARCH OVERVIEW

Most regulators require nuclear power plant owners to demonstrate acceptable fatigue EAF cumulative usage factor (CUF_{en}) values for long-term plant operation for limiting reactor coolant pressure boundary components. Therefore, all U.S. nuclear plants that apply for 60- or 80-year extended operating licenses must address EAF consistent with U.S. Nuclear Regulatory Commission (NRC) Generic Aging Lessons Learned (GALL) or Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) guidance. Many non-U.S. nuclear plants must also address EAF for extended plant operation, and those plants use either GALL or similar guidance.

In August 2012, EPRI published an EAF screening process in report 1024995, *Environmentally Assisted Fatigue Screening: Process and Technical Basis for Identifying EAF Limiting Locations*. The EAF screening process in the EPRI report was developed to describe the technical basis for an EAF screening approach that satisfied GALL guidance to identify "additional plant-specific component locations in the reactor coolant pressure boundary if they may be more limiting than those considered in NUREG/CR–6260." This report updates and supersedes 1024995 and provides three EAF screening methods that may be used to screen critical plant components for EAF to identify the limiting plant locations to satisfy GALL and other regulatory guidance.

KEY FINDINGS

- This report adopts lessons learned from previous applications of the EPRI 2012 EAF Screening Report, identified in this report as the Structural Integrity Associates, Inc. (SIA) EAF Screening Method. This includes reducing unnecessary conservatisms when screening Class 1 piping component locations and including consideration of prior U.S. NRC requests for additional information.
- This report updates the SIA EAF Screening Method as necessary to address more recent SLR-related guidance.
- This report documents two other available EAF screening methods that have been developed and used by nuclear plants worldwide. These include the Westinghouse Electric Company LLC (WEC) EAF Screening Method and the Électricité de France (EDF) EAF Screening Method.

WHY THIS MATTERS

The EAF screening methods in this report will enable plant owners to demonstrate knowledge of the locations in their plants that can serve as locations requiring EAF management to satisfy long-term operation EAF guidance. The EAF screening methods provide the rationale for selecting these locations. Plant owners can minimize costs by avoiding the necessity of formal fatigue analyses and reducing the overall number of component locations requiring EAF assessment while meeting the regulatory requirements to determine EAF locations that are limiting in their plants. The EAF screening methods each provide a uniform approach for determining the limiting locations that require EAF assessment and management throughout plant extended operating periods.

HOW TO APPLY RESULTS

Any of the three EAF screening methods described in this report may be used to determine the Class 1 locations in nuclear power plants that require EAF assessment for long-term operation.

LEARNING AND ENGAGEMENT OPPORTUNITIES

- International codes and standards committees, such as the American Society of Mechanical Engineers, may find the report useful for possible future refinements to fatigue calculation methods.
- Materials Reliability Program (MRP) meetings.
- Boiling Water Reactor Vessel and Internals Project (BWRVIP) meetings.

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ACRONYMS AND VARIABLE DEFINITIONS

α	coefficient of thermal expansion
α_a, α_b	coefficient of thermal expansion on side *a* (or *b*) of a structural discontinuity as defined in ASME Code, Section III, Paragraph NB-3653
ANL	Argonne National Laboratory
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASN	French Safety Authority
Bi	Biot number
BWR	boiling water reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
C_1, C_2, C_3	piping stress indices
CFR	Code of Federal Regulations
CLB	current licensing basis
c_p	specific heat
CS	carbon steel
CUF	cumulative usage factor (same as *U*)
CUF_{en}	EAF cumulative usage factor (same as U_{en})
CVCS	chemical and volume control system
D	pipe inside diameter (same as *ID*)
D_o	pipe outside diameter
DO	dissolved oxygen
DSR	Design Stress Report
dtime	transient ramp time
ε	strain
ε_i, ε_{i-1}	strain at time intervals, *i* and *i-1*
ε_{max}, ε_{min}	maximum and minimum strain over a time interval

$\Delta\varepsilon_i$	change in strain for time interval, $i = \varepsilon_i - \varepsilon_{i-1}$
$\dot{\varepsilon}$	strain rate
$\dot{\varepsilon}_i$, $\dot{\varepsilon}_k$	strain rate for interval, *i* or *k*
$\dot{\varepsilon}^*$	transformed strain rate
E	Young's modulus of elasticity
E_{ab}	Young's modulus of elasticity for a structural discontinuity *a*/*b* as defined in ASME Code, Section III, Paragraph NB-3653
EAF	environmentally assisted fatigue
EDF	Électricité de France
EPRI	Electric Power Research Institute
F_{adj}	material fatigue curve environmental multiplier
FEA	finite element analysis
F_{en}	environmental fatigue multiplier
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
$F_{en\text{-}effective}$	effective environmental fatigue multiplier
$F_{en\text{-}integrated}$	F_{en} value that accounts for the effects of the PWR environment already covered in the fatigue curve
$F_{en\text{-}m,n}$	F_{en} calculated for the transient pair (*m*,*n*)
$F_{en\text{-}TOT}$	F_{en} calculated for the selected zone for all transient combinations
$F_{en,i}$, $F_{en,k}$	environmental fatigue multiplier for the i^{th} transient cycle pair or for time interval, *i* or *k*
FMP	fatigue management program
Fo	Fourier number
FROG	Framatome Reactor Owners Group
FSAR	Final Safety Analysis Report
FSRF	fatigue strength reduction factor
FU	a new cumulated usage factor incorporating EAF obtained by multiplying the partial usage factor by F_{en} (also called FU_{en})
$FU_{m,n}$	partial usage factor resulting from the combination of the transients *m* and *n*
GALL	Generic Aging Lessons Learned
GF	gradient factor

h	heat transfer coefficient
HHSI	high head safety injection
HWC	hydrogen water chemistry
I	piping moment of inertia
ID	inside diameter (same as D)
ISI	in-service inspection
k	thermal conductivity
K_1, K_2, K_3	piping stress indices
K_e	simplified elastic-plastic strain concentration factor
L	piping length
LAS	low alloy steel
LHSI	low head safety injection
LWR	light water reactor
m	material parameter for computing K_e
M	total number of sets of transient load set pairs
M_i	moment load for load pair, i
$M_{seismic}$	seismic moment load
M_{strat}	thermal stratification moment load
$M_{thermal}$	thermal moment load
MRP	Materials Reliability Program
v	kinematic viscosity
n	number of applied cycles or transients; or material parameter for computing K_e
n_{events}	design basis number of events
n_i	number of applied cycles for transient pair, $_i$
N_i	number of allowable cycles for transient pair, $_i$
$n_{projected}$	projected number of cycles
$N_{air,RT}$	number of stress cycles in air at room temperature
N_{allow}	number of allowable cycles
N_{water}	number of stress cycles in reactor water at the service temperature
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission

NWC	normal water chemistry
*O**	transformed dissolved oxygen content
OD	outside diameter
P_{max}	maximum pressure during a transient
P_{min}	minimum pressure during a transient
P_o	range of pressure for a transient pair
ΔP	transient pressure change
P&ID	piping and instrumentation diagram
ppb	parts per billion
ppm	parts per million
Pr	Prandtl number
PSA	periodic safety assessment
PWR	pressurized water reactor
Q	thermal hydraulic flow rate
R	mean pipe radius = (*OD* + *ID*) / 2
ρ	density
RCC-M	Règles de conception et de construction des matériels mécaniques des ilôts nucléaires REP (Design and Construction Rules for Mechanical Components of PWR Nuclear Islands)
RCL	reactor coolant loop
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
Re	Reynold's number
RG	Regulatory Guide
RHR	residual heat removal
RHX	regenerative heat exchanger
RWCU	reactor water cleanup
S	material sulfur content
*S**	transformed sulfur content
S_{alt}	alternating stress amplitude
S_{disc}	peak stress range from gross structural or material discontinuity

S_m	design stress intensity
S_{mom}	peak stress range from moments
S_n	primary plus secondary stress intensity range
S_{peak}	total peak stress range
S_{press}	peak stress range from pressure
S_{ts}	peak stress range from through-wall thermal gradient
SCF	stress concentration factor
SCL	stress cut line
SIA	Structural Integrity Associates, Inc.
SLR	subsequent license renewal
SRP	Standard Review Plan
σ_t, σ_{i-1}	stress at time intervals, *i* and *i-1*
t	wall thickness or transient time
t_h	wall thickness
t_i, t_{i-1}	transient time at intervals, *i* and *i-1*
Δt	ramp duration of the shock
Δt_i	change in time for time interval *i*, $= t_i - t_{i-1}$
Δt_{tot}	total time for the entire transient event to occur
t_0	transient time
t_a, t_b	average thickness on side *a* (or *b*) of a structural discontinuity as defined in ASME Code, Section III, Paragraph NB-3653
T	metal or fluid temperature
T^*	transformed temperature
T_a, T_b	average temperature on side *a* (or *b*) of a structural discontinuity as defined in ASME Code, Section III, Paragraph NB-3653
T_{ave}	average temperature during transient or of each load pair
T_{final}	final temperature during transient
T_i, T_o	inside and outside surface temperatures
T_{init}	initial temperature during transient
T_{max}	maximum temperature during transient
T_{min}	minimum temperature during transient

T_p	time constant of a shock
ΔT	transient temperature change
ΔT_1	absolute value of range of the temperature difference for each load set pair between the temperature of the outside surface, T_o, and the temperature of the inside surface, T_i, of the piping product assuming a moment generating equivalent linear temperature distribution or the temperature change for Transient 1 (EDF Method)
ΔT_2	absolute value of range for that portion of the nonlinear thermal gradient through the wall thickness not included in ΔT_1 or the temperature change for Transient 1 (EDF Method)
ΔT_o	transient temperature change $= (T_{init} - T_{final})$
TF	thickness factor
TLAA	time limited aging analysis
U	cumulative usage factor (same as CUF)
U^*	estimated CUF produced on a common basis
U_i	CUF for the i^{th} transient cycle pair or for time interval, i
U_{incr}	estimated incremental CUF for a load pair
U_{incr}^*	estimated incremental CUF produced on a common basis for a load pair
U_{6260}	CUF for a NUREG/CR-6260 location
U_{6260}^*	estimated CUF produced on a common basis for NUREG/CR-6260 location
U_{adj}	U with F_{adj} applied
U_{en}	EAF cumulative usage factor (same as CUF_{en})
U_{en}^*	estimated U_{en} produced on a common basis as $F_{en}^* \times U^*$
$U_{en\ max}^*$	maximum value of U_{en} and/or U_{en}^* in a set of locations
$U_{en\ 6260}$	U_{en} produced from the CUF for a NUREG/CR-6260 location as $F_{en} \times U_{6260}$
$U_{en\ 6260}^*$	estimated U_{en} produced for a NUREG/CR-6260 location on a common basis as $F_{en}^* \times U_{6260}^*$
$U_{en\ incr}$	incremental U_{en} for a load pair produced from U_{incr} and F_{en} values for each transient pair as $F_{en} \times U_{incr}$
$U_{en\ incr}^*$	estimated incremental U_{en}^* for a load pair on a common basis as $F_{en}^* \times U_{incr}^*$
V	fluid velocity
WEC	Westinghouse Electric Company LLC

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1 INTRODUCTION

1.1 Background

Chapter X.M1 of both NUREG-1801, Revision 2, the Generic Aging Lessons Learned (GALL) Report [1], and NUREG-2191, the Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report [2], provide U.S. Nuclear Regulatory Commission (NRC) guidance for identification of the limiting plant-specific component locations in the reactor coolant pressure boundary (RCPB) when considering environmentally assisted fatigue (EAF) effects on component fatigue life. Specifically, under Scope of Program, Chapter X.M1 of the GALL Report states:

> *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*** *[emphasis added].*

And, Chapter X.M1 of the GALL-SLR Report similarly states:

> *For the purposes of ascertaining the effects of the reactor water environment on fatigue, applicants include CUF_{en} calculations for a set of sample reactor coolant system components. This sample set includes 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*** *[emphasis added]. Plant-specific justification can be provided to demonstrate that calculations for the NUREG/CR–6260 locations do not need to be included.*

The above guidance is different than the guidance that was provided in earlier revisions to the GALL Report where EAF assessment was limited to the locations addressed for the specific plant design/vintage in NUREG/CR-6260 [3]. The current guidance, as indicated by the emphasized text in the above excerpts, identifies that other potentially more limiting locations than those in NUREG/CR-6260 must also be assessed.

The NRC requires applicants for extended plant operation to demonstrate acceptable fatigue cumulative usage factor (*CUF*) values for limiting plant Class 1 components that include the effects of a reactor water environment for the entire licensed period of operation. Therefore, all U.S. nuclear plants that apply for 60- or 80-year extended operating licenses must address EAF consistent with GALL or GALL-SLR guidance. Many non-U.S. nuclear plants must also address EAF for extended operation, and many of those plants use either the same or similar guidance.

In August 2012, EPRI published an EAF screening process in Technical Report 1024995 [4]. The EAF screening process in the EPRI report was developed to describe the technical basis for an EAF screening approach that satisfied GALL guidance to identify, "*additional plant-specific component locations in the reactor coolant pressure boundary if they may be more limiting than those considered in NUREG/CR–6260*." The EAF screening process in EPRI 1024995 was developed by Structural Integrity Associates, Inc. (SIA), and is herein referred to as the "SIA EAF Screening Method." The SIA EAF Screening Method considers the level of analysis rigor when assessing limiting Class 1 locations. Where Class 1 piping was designed to American National Standards Institute (ANSI)/American Society of Mechanical Engineers (ASME) B31.1 [5] and no design stress report exists, an estimated fatigue approach is defined that applies guidance from the EPRI Fatigue Management Handbook [6].

Generally, the EAF screening process requires an EAF evaluation and screening of Class 1 components, some of which do not have an explicit *CUF* calculation because it was not required by the design Code of record. Evaluating Class 1 components with a *CUF* calculation is generally straightforward and may only require a relatively simple evaluation process to estimate and apply environmental fatigue multipliers (F_{en}) to the existing *CUF* values to create CUF_{en} values (also referred to as U_{en} values in this report). In some cases, further refinement of *CUF* and F_{en} values are required to reduce CUF_{en} (or U_{en}) values to less than the allowed value of 1.0 to eliminate locations that would otherwise require corrective action. Class 1 components without *CUF* calculations (such as piping components designed to ANSI/ASME B31.1) may require a more extensive evaluation process to evaluate plant locations on a consistent basis because explicit *CUF* information is lacking.

The EPRI EAF Screening Report has not been submitted for formal regulatory review or approval as a topical report. However, several U.S. license renewal and SLR applicants have cited the EPRI report as a basis for the EAF screening applied to their plants as a part of their NRC license extension submittals. As a consequence, an SLR lessons-learned document issued in December 2017 [7] noted NRC issues with some aspects of EPRI 1024995.

This report represents a revision and update to EPRI 1024995 in its entirety that accomplishes the following:

- This report adopts lessons learned from previous applications of the SIA EAF Screening Method. This includes reducing unnecessary conservatisms when screening Class 1 piping component locations and including consideration of prior NRC requests for additional information.
- This report updates the SIA EAF Screening Method as necessary to address more recent SLR-related guidance.
- This report documents other available EAF screening methods that have been developed by others and used by nuclear plants worldwide. Included in this report revision are the following two additional methods:
    1. The EAF screening method developed and used by Westinghouse Electric Company LLC (WEC), herein referred to as the "WEC EAF Screening Method."
    2. The EAF screening method developed and used by Électricité de France (EDF), herein referred to as the "EDF EAF Screening Method."

1.2 Contents of this Report

The EAF screening methods in this report will enable plant owners to demonstrate knowledge of the locations in their plant that can serve as locations requiring EAF management to satisfy long term operation EAF guidance. The EAF screening methods provide the rationale for selecting these locations. Plant owners can minimize costs by avoiding the necessity of formal fatigue analyses and reducing the overall number of component locations requiring EAF assessment, while meeting the regulatory requirements to determine those EAF locations that are limiting in their plants. The EAF screening methods each provide a uniform approach for determining the limiting locations that require EAF assessment and management throughout extended plant operating periods.

The report is divided into the following sections:

- Section 2 provides some basics for fatigue and EAF, a definition of the basic fatigue terms used in this report, and background on fatigue analysis and EAF requirements for extended operation.
- Section 3 provides the SIA EAF Screening Method including the technical basis for the method.
- Section 4 provides the WEC EAF Screening Method including the technical basis for the method and an example application of the method.
- Section 5 provides the EDF EAF Screening Method including the technical basis for the method and an example application of the method. The EDF Screening Method is updated in this report from its initial publication in France to use the methods of NUREG/CR-6909, Revision 1 [8].
- Section 6 provides the description of a spreadsheet tool developed for estimating stainless steel F_{en} values. The spreadsheet tool was not developed under a quality assurance program and is intended for informational purposes only.
- Section 7 provides a summary of the overall conclusions of this report.
- Section 8 provides a list of all references cited in this report.
- Appendix A provides a summary of the available F_{en} formulations used by the three screening methods documented in this report.
- Appendix B provides a summary of the ASME Code rules for evaluation of Class 1 piping components.

2
FATIGUE AND EAF BASICS

This section provides definitions of a few basic fatigue terms used throughout this report, as well as background on fatigue analysis and EAF requirements for extended plant operation.

Most plants are qualified for fatigue via a design process defined in ASME Boiler and Pressure Vessel Code, Section III [9], ANSI/ASME B31.7 [10], or ANSI/ASME B31.1 [5]. For components subject to cyclic loadings, this generally includes a fatigue analysis. The ASME Code, Section III and ANSI/ASME B31.7 design codes provide rules for the explicit determination of *CUF*. The ANSI/ASME 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 generally straightforward and may require a relatively simple evaluation process to estimate and apply environmental fatigue multipliers (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.

2.1 Definition of Terms

Some basic fatigue terms are defined in this section. These terms are those that are commonly used in fatigue analyses, and they are common definitions for all sections of this report. Other, method-specific terms are included as necessary in each of the methodology sections of this report.

U: Design *CUF* (documented in the Design Stress Report (DSR) and design analyses).

Fatigue Table: The compilation of incremental *CUF* values 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.

F_{en}: A factor that represents an environmental multiplier in a fatigue life analysis to account for the effects of a water environment. This may apply to the entire *CUF* or on a Load Pair basis.

U_{en}: Usage factor that includes environmental effects equal to *CUF* multiplied by F_{en}. Also referred to as CUF_{en}.

Leading Transient: A thermal event that contributes significantly to the overall *CUF* of a component. An example is a temperature shock from starting and stopping fluid flow in nozzles.

Bundled Transients: Enveloping of multiple plant transient definitions by one conservative and bounding transient definition.

2.2 Basics of Fatigue Analysis

According to the ASME Code [9], 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 an acceptable design.

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

- Determine the stresses for all normal and upset 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 the extreme states of the service conditions.
- Determine the alternating stress amplitude (S_{alt}) for each stress cycle or load set 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 concentration factor K_e, which is calculated in accordance with Section III, Paragraph NB-3228.5[1] if the primary plus secondary stress range exceeds three times the design stress intensity ($3S_m$).
- Determine the allowable number of cycles (N_{allow}) for each transient pair, using a fatigue curve. A fatigue curve is a graph that shows the allowable cycles as a function of alternating stress intensity, or S_{alt}, values. Typical design fatigue curves in air reproduced from Appendix I of Section III of the ASME Code [9] and NUREG/CR-6909, Revision 1 [8] are shown in Figure 2-1 for carbon steel (also included in the figure is the EAF strain amplitude threshold, which is discussed in Sections 2.4 and 2.6). Prior to entering the fatigue curve 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 the modulus of elasticity used in the stress analysis.
- Determine fatigue usage contributions 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_{allow} (taken as either 10^6 or 10^{11} cycles depending upon material and edition of the ASME Code).

[1] Starting with the 2017 Edition, this content was relocated to Mandatory Appendix XIII.

The cumulative fatigue usage factor, U, or (CUF) is determined by the summation:

$$U = \sum_{i=1}^{M} \frac{n_i}{N_i} \qquad \textbf{Eq. 2-1}$$

where:

n_i = number of applied cycles for transient load set pair, i

N_i = number of allowable cycles for transient load set pair, i

M = total number of sets of transient load set 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.

Typically, design *CUF* values are available for ASME Code, Section III and ANSI/ASME B31.7 piping and ASME Code, Section III equipment locations. Many times, the degree of rigor used to generate the design *CUF* values is not consistent across component locations, or the design *CUF* value may be overly conservative. In situations such as these, the design *CUF* value may be reduced by:

- Revision of the CUF evaluation based upon load pairing by ungrouping or unbundling of transients.
- Recalculating CUF using actual plant operating transients and conditions rather than design basis transients and conditions (i.e., actual temperature and pressure fluctuations).
- Recalculating CUF using actual transient cycle accumulations and projections rather than conservative design basis transient cycles, as several design basis transients generally either do not occur or occur seldomly during plant operation. Many plants will not exceed the number of design transients even after 80 years of plant operation.
- Use of a later Code edition analysis (for example, in Subsection NB, later Code Editions reclassified the linear thermal gradient stress intensity from a secondary stress to a peak stress, thereby reducing K_e).
- Application of ASME Code Case N-779 [11] alternative rules for calculation of K_e such that K_e is reduced.
- Calculation of Thickness Factor (*TF*) and Gradient Factor (*GF*) per ASME Code Case N-902 [12].
- Application of ASME Code Case N-904 [13] alternative rules for calculation of K_e such that K_e is reduced.

Other methods may be used to reduce the design *CUF* value; details on the other methods for reducing the *CUF* value are not included in this report. Similarly, reductions in CUF_{en} can be achieved after reducing the *CUF* by recalculating F_{en} values with various approaches, e.g., using an average temperature, using actual plant transient data, etc.

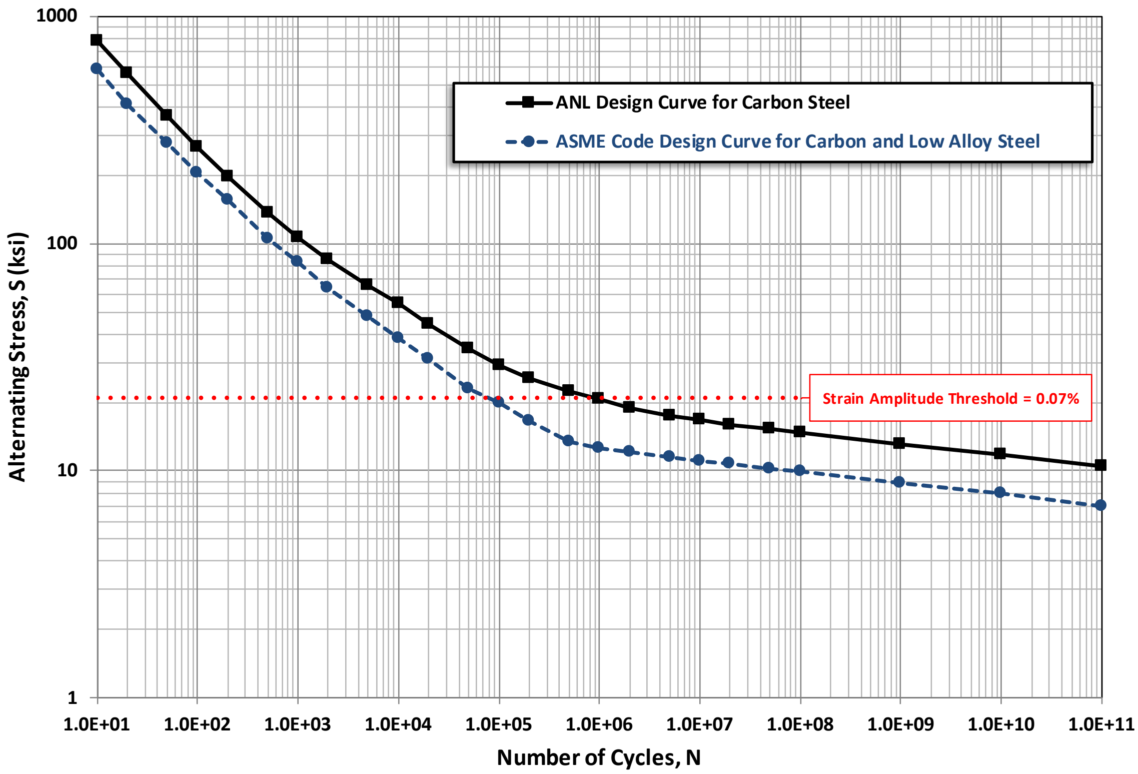


Figure 2-1
Typical design fatigue curve in air for carbon steel

2.3 Background on EAF Calculations

Over the past several 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. 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, low alloy, austenitic stainless and nickel alloy steels used in nuclear plants [8, 14, 15, 16]. With respect to these parameters, an environmental fatigue multiplier (F_{en}) approach was developed to incorporate the effects of LWR environments into ASME Code, Section III fatigue evaluations.

NUREG-1801, Revision 2 [1] and NUREG-2191 [2] identify acceptable aging management programs for fatigue and cyclic operation for the period of extended operation. They describe 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 [3].

NRC guidance in NUREG/CR-5704 [15], NUREG/CR-6583 [16] and NUREG/CR-6909 [8, 14] defines the F_{en} multiplier 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 ($1/N_{allow}$), thus:

$$F_{en} = N_{air,RT}/N_{water} \quad \textbf{Eq. 2-2}$$

Environmental effects are incorporated into ASME Code, Subsections NB-3200 or NB-3600 [9] fatigue analyses by multiplying the partial ASME usage factors for each stress cycle by the F_{en} multiplier computed for that stress cycle. For example, referring to Eqn. 2-1 for *M* different transient cycle pairs, the cumulative environmental fatigue usage is:

$$U_{en} = U_1 \cdot F_{en,1} + U_2 \cdot F_{en,2} + U_3 \cdot F_{en,3} + U_i \cdot F_{en,i} \ldots + U_M \cdot F_{en,M} \quad \textbf{Eq. 2-3}$$

where:

U_i = computed fatigue usage using the air fatigue curve for the i^{th} transient cycle pair

$F_{en,i}$ = computed F_{en} for the i^{th} transient cycle pair

An effective F_{en} factor, $F_{en\text{-}effective}$, is computed as follows:

$$F_{en-effective} = U_{en}/U \quad \textbf{Eq. 2-4}$$

Where *U* is calculated per Eqn. 2-1.

2.4 F_{en} Formulations

Per NUREG-1801 [1], plants have the option of computing U_{en} in accordance with guidance from EITHER NUREG/CR-5704 [15] (for austenitic stainless steels), NUREG/CR-6583 [16] (for carbon and low alloy steels) and NUREG/CR-6909, Revision 0 [14] (for nickel alloy steels), OR NUREG/CR-6909, Revision 0 [14] (for all materials).

Per the GALL-SLR Report [2], plants have the option of computing U_{en} in accordance with guidance from EITHER NUREG/CR-6909, Revision 0 [14] (with "average temperature" used consistent with the clarification that was added to NUREG/CR-6909, Revision 1), OR NUREG/CR-6909, Revision 1 [8] for all materials.

In the case of NUREG/CR-5704 and NUREG/CR-6583 for austenitic stainless and carbon/low alloy steels, 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 curves provided in Appendix A of both revisions of NUREG/CR-6909 [8, 14] must be used for austenitic stainless and nickel alloy steels, and optionally for carbon and low alloy steels (the ASME curve is more conservative for carbon and low alloy steels steels).

However, it is worth noting that different ASME Code editions are referenced by each of the F_{en} NUREG reports, and the ASME Code fatigue curves have changed over the time span of those reports. For example, the stainless steel fatigue curve was revised in the 2009 Addenda of Section III to be consistent with NUREG/CR-6909, but NUREG/CR-5704 and NUREG/CR-6583 reference earlier editions of the ASME Code. Therefore, users should be careful that *CUF* values are computed consistent with the fatigue curves required for the particular F_{en} method they are using.

Parameters used in equations for each of the above NUREGs are discussed below and are provided in detail in Appendix A.

*S**	=	a transformed factor based on the *Sulfur Content, S,* of the steel (for carbon and low alloy steels only).
*T**	=	a transformed factor based on the *Service Temperature*.
*O**	=	a transformed factor based on the *Dissolved Oxygen (DO)*.
$\dot{\varepsilon}^*$	=	a transformed factor based on the *Strain Rate*.

Sulfur Content: The weight percent content of sulfur in the steel.

Service Temperature: The temperature of the metal in contact with the primary fluid "environment." For all materials, higher temperature causes a higher F_{en} (up to a maximum value).

Dissolved Oxygen: The level of *DO* in the primary fluid.

Strain Rate: The rate of strain in the metal during the increasingly tensile time periods for each transient cycle pair. Strain rate is the parameter that has the largest effect on the value of F_{en} for all materials when all other parameters are above threshold values.

Strain Amplitude Threshold: A minimum strain amplitude below which LWR environmental effects are considered insignificant on the fatigue life of steels. The strain amplitude threshold from NUREG/CR-6909, Revision 1 [8] for carbon steel is shown in Figure 2-1.

Temperature Threshold: A minimum temperature value below which LWR environmental effects are considered insignificant on the fatigue life of steels. This threshold is used as the lower value when determining an "average temperature" to compute F_{en} if the minimum transient temperature is below the temperature threshold.

2.5 Maximum F_{en} Values

In cases where parameter values are uncertain, the following parameter values may be used to calculate a maximum F_{en} value for a conservative assessment of EAF:

<u>NUREG/CR-6909, Revision 0 [14]</u>

For carbon steel and low alloy steels:

S	=	0.015 weight percent (wt. %) or higher
DO	=	0.5 parts per million (ppm) or higher
T	=	350°C (617°F) or higher
$\dot{\varepsilon}$	=	0.001%/sec or lower

For austenitic stainless and nickel alloy steels:

T	=	325°C (617°F) or higher
$\dot{\varepsilon}$	=	0.0004%/sec or lower

NUREG/CR-5704 [15]

For austenitic stainless steels:

T = 200°C (392°F) or higher

DO = 0.05 ppm or higher

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

NUREG/CR-6583 [16]

For carbon steel and low alloy steels:

S = 0.015 weight percent or higher

DO = 0.5 ppm or higher

T = 350°C (662°F) or higher

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

NUREG/CR-6909, Revision 1 [8]

For carbon steel and low alloy steels:

S = 0.015 weight percent or higher

DO = 0.5 ppm or higher

T = 325°C (617°F) or higher

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

For wrought and cast austenitic stainless and nickel alloy steels:

T = 325°C (617°F) or higher

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

For nickel alloy steels (except Inconel 718):

T = 325°C (617°F) or higher

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

2.6 Modified Rate Approach

A modified rate approach can also be used to incorporate transient-specific results into the F_{en} equation. The modified rate approach integrates F_{en} over the positive strain rate portion of the transient cycle pair. The modified rate approach is illustrated by the following equation:

$$F_{en} = \sum_{i=1}^{n} F_{en,i} \frac{\Delta\varepsilon_i}{(\varepsilon_{max} - \varepsilon_{min})} \qquad \textbf{Eq. 2-5}$$

where:

n	=	number of intervals from the stress valley to the stress peak for the tensile strain portion of the transient cycle pair
$F_{en,i}$	=	F_{en} computed for time interval, i, based on the strain rate, $\dot{\varepsilon}_i = 100\Delta\varepsilon_i/\Delta t_i$ (%/sec.) and transformed parameters T^*, $\dot{\varepsilon}^*$, and O^* computed for the interval
$\Delta\varepsilon_i$	=	change in strain for time interval i
	=	$(\sigma_i - \sigma_{i-1}) / E$
σ_i	=	stress intensity for time i
σ_{i-1}	=	stress intensity for time $i-1$
ε_{max}	=	maximum strain for the increasing strain transient
ε_{min}	=	minimum strain for the increasing strain transient
Δt_i	=	change in time for time interval i, $\Delta t = t_i - t_{i-1}$
E	=	Young's modulus

Per both revisions of NUREG/CR-6909 [8, 14], the strain amplitude threshold is defined as 0.07% for carbon and low alloy steels, or a stress amplitude of 21 ksi (as shown in Figure 2-1); and 0.10% for austenitic stainless and nickel alloy steels, or a stress amplitude of 28.3 ksi. If the strain amplitude for a transient cycle pair is less than the strain amplitude threshold, then F_{en} is set to 1.0 for that transient cycle pair.

2.7 U.S. Requirements for License Renewal and Subsequent License Renewal

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

NUREG/CR-6260 [3], published in 1995, established a set of six locations (by plant design and vintage) that were expected to be representative of components that had higher *CUF* values and/or were important from a risk perspective. NUREG/CR-6260 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 fatigue sensitive locations in the plant.

The NRC has questioned whether the NUREG/CR-6260 locations effectively cover all locations in the plant (see GALL Revision 2 [1] and GALL-SLR [2]). 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 locations that may be more limiting for EAF than the NUREG/CR-6260 locations on a plant-specific basis.

Determination of this list is not as simple 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 higher *CUF* values for a component than analysis of actual severity plant transients.
  - Analyses using "bundled transients" yield significantly higher *CUF* values than analyses of the same component with "unbundled" transients.
  - Design by rule (ASME Code, Section III, Subarticle NB-3600 or ANSI/ASME B31.7) typically, for limiting locations, produces higher *CUF* values than design by analysis (ASME Code, Section III, Subarticle NB-3200).
- For a given plant transient, F_{en} factors often trend counter to the computed *CUF* values, thus potentially complicating the ranking of the U_{en} values for a component.
  - Faster rise times for thermal transients cause the highest stresses and larger *CUF* values, but they produce lower F_{en} factors. Because $U_{en} = F_{en} \times U$, the largest 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 materials of construction.
  - *DO* content affects materials of construction differently:
    - For some F_{en} methods, higher *DO* yields lower F_{en} values for austenitic stainless steels [15].
    - For some F_{en} methods, there is a constant effect for all *DO* contents for austenitic stainless steels [14].
    - For some F_{en} methods, there is a constant effect for all *DO* contents for austenitic stainless steels for pressurized water reactors (PWRs) and under hydrogen water chemistry (HWC) conditions for boiling water reactors (BWRs), and a constant effect for all *DO* content for austenitic stainless steels for normal water chemistry (NWC) in BWRs except for sensitized high-carbon stainless steels [8].

- o Higher DO yields higher F_{en} values for carbon and low alloy steels [8, 14, 16].
- o There is a constant effect for all *DO* contents for nickel alloy steels for PWRs, during HWC in BWRs, and for nickel alloy steels during NWC in BWRs [8, 14].

Further factors that influence the evaluations are:

- Use of NUREG/CR-5704 [15] (for austenitic stainless steels) and NUREG/CR-6583 [16] (for carbon and low alloy steels) will produce different values of F_{en} compared to the newer rules of NUREG/CR-6909 [8, 14] 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 [17] where it was determined that:
  - Although ANSI/ASME B31.1 [5] and ASME Code, Section III [9] and ANSI/ASME B31.7 [10] Class 1 piping rules are fundamentally different, experience in operating plants has shown that piping systems designed to ANSI/ASME B31.1 are adequate.
  - The operation of ANSI/ASME B31.1 plants is also not different from that of plants designed to ASME Code, Section III.
  - Section 2.8 describes practices for addressing EAF for ANSI/ASME B31.1 components.

Providing a robust solution without resorting to a complete reanalysis requires a new approach. EDF, SIA, and WEC have developed methods for screening the primary coolant-wetted Class 1 RCPB fatigue-sensitive components in a plant by ranking them in terms of U_{en} and then determining a set of leading locations, or *Sentinel Locations*, such that every plant component is covered by one or more Sentinel Locations. The basis for this approach is to first group components into common thermal zones, or transient sections. CUF_{en} values for the sampled locations in NUREG/CR-6260 can be compared to Sentinel Locations within the same transient section or thermal zone. Sentinel Location CUF_{en} values in thermal zones without corresponding locations from NUREG/CR-6260 can be screened relative to whether they are greater than or less than unity. Thus:

- A *Sentinel Location* is a specific location in a piping system or component that serves as a leading indicator for EAF damage accumulation. In this context, a Sentinel Location is a location in a plant system or Transient Section or Thermal Zone that is expected to accumulate more EAF usage than other locations in that system. These Sentinel Locations are expected to remain bounding as plant transients accumulate during plant operation. Thus, management of Sentinel Locations maintains 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.
- A *Transient Section* or *Thermal Zone* is defined and discussed in further detail in Sections 3, 4 and 5 for each of the EAF screening methods.

EPRI Report 1022873 [18] concludes that CUF_{en} (or U_{en}) 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 CUF_{en} (or U_{en}) value of less than 1.0 has an insignificant safety impact, since the estimated core damage frequency was in all cases below $1x10^{-6}$ failures/per reactor year. Thus, the consideration of a special EAF limit for the purposes of screening or ranking of postulated intermediate pipe break locations is not considered in this report.

2.8 Addressing EAF for ANSI/ASME B31.1 Components

ANSI/ASME B31.1 does not require explicit calculation of CUF. As a result, time limited aging analyses (TLAAs) defined under 10 CFR 54.3 for piping components designed in accordance with ANSI/ASME B31.1are addressed differently than piping components that have a CUF calculation as part of their current licensing basis (CLB). Piping components designed in accordance with the ANSI B31.1 design rules do not have an explicit CUF calculation; rather, cyclic loading is considered in an implicit, simplified manner in the design process using a stress range reduction factor based on the number of thermal and pressure cycles expected during the component operating lifetime. If the total number of fatigue cycles is expected to be 7,000 or less, the stress range reduction factor is 1.0. For higher numbers of fatigue cycles, a stress range reduction factor of less than 1.0 is used in the assessment for the piping component, which reduces the allowable stress range. As a result, there is no CUF available for B31.1 piping locations for which to apply Fen multipliers and address EAF.

In accordance with 10 CFR 54.21(c)(1), the TLAA evaluations for B31.1 piping components are typically shown to experience actual fatigue cycles during the extended licensed operating time period that are significantly less than 7,000 cycles. Therefore, the CLB analyses remain valid for the period of extended operation and further assessment, including for EAF, is not required, provided the transient pressure and thermal cycles are monitored and verified to stay within the CLB. This is supported by the fatigue monitoring program guidance in the Chapter X.M1 of the GALL SLR Report [2], which states:

> *Some of the design fatigue analyses are implicit evaluations or fatigue waivers. Both of these analyses provide the basis for not requiring detailed fatigue analyses (e.g., CUF, CUF_{en}). Implicit evaluations specify allowable stress levels based on the number of anticipated full thermal range transient cycles. As an example, piping components designed to USAS American National Standards Institute (ANSI) B31.1 requirements and ASME Code Class 2 and 3 components designed to ASME Code Section III design requirements include implicit cycle-based maximum allowable stress range calculations. Fatigue waivers are based on transient cycle limits. Fatigue waivers may have been permitted such that a detailed fatigue calculation was not required if a component conformed to certain criteria, such as those established in ASME Code, Section III, NB-3222.4(d). The AMP monitors and tracks the number of critical thermal and pressure transient occurrences for the selected components and verifies that the severity of the monitored transients is bounded by the design transient definitions in order to ensure these implicit fatigue evaluations or fatigue waivers remain valid.*

The above is consistent with the NRC's findings associated with their Fatigue Action Plan investigations, as discussed in Section 1.4 of NUREG/CR-6909, Revision 1 [8]. In addition, RG 1.207, Revision 1 [19] defines the scope of EAF as follows:

> *This regulatory guide (RG) describes methods and procedures that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in determining the acceptable fatigue lives of components evaluated by a cumulative usage factor (CUF) calculation in accordance with the fatigue design rules in Section III, "Rules for Construction of Nuclear Power Plant Components," of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereinafter "Code") (Ref. 1), to account for the effects of light-water reactor (LWR) water environments.*

The WEC and SIA Screening Methods in Sections 3 and 4, of this report include processes for addressing EAF in B31.1 piping locations in the limited cases where EAF assessment is necessary.

2.9 EAF for Reactor Internals Components

Generally, EAF assessment is not performed for reactor internals because internals components do not typically possess a CUF calculation. In these cases, monitoring of the transient pressure and thermal cycles and verifying that they stay within the CLB is sufficient. This practice is consistent with GALL-SLR Report [2] guidance, as stated under the "Scope of Program" in Chapter X.M1, "Fatigue Monitoring":

> *For the purposes of ascertaining the effects of the reactor water environment on fatigue, applicants include CUF_{en} calculations for a set of sample reactor coolant system components. This sample set includes 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.*

This is also consistent with the RG 1.207, Revision 1 [19] guidance quoted in Section 2.8 for B31.1 piping locations.

Some newer vintage plants have CUF calculations for internals components because Subsection NG, "Core Support Structures," of Section III of the ASME Code, or similar requirements, was used for the design of the reactor internals. In such cases, EAF assessment may be appropriate; however, BWR and PWR internals are subject to inspection programs, which may suffice for addressing the fatigue management needs for reactor internals for long term operation. For example, Appendix D of BWRVIP-315 [20] provides an aging management program-based approach for disposition of cracking due to low-cycle EAF. That appendix demonstrates how the existing guidance provided by the BWR Vessel and Internals Project (BWRVIP) inspection-based reactor internals program adequately fulfills the guidance currently defined in the GALL-SLR Report for managing EAF, such that relying on updated CUF_{en} calculations to disposition fatigue TLAAs are not necessary. PWR plants inspecting reactor internals in accordance with MRP-227 [21] are addressed because that report includes EAF screening for generic applicability.

Therefore, the need to address EAF for reactor internals is covered for most plants under their internals inspection programs.

3 SIA EAF SCREENING METHOD

The purpose of this section is to describe the technical bases of the SIA EAF screening method. The SIA EAF screening method defines a process that may be used for EAF screening and ranking of components in nuclear power plant Class 1 systems. This process allows for consistent application across multiple components with differing *CUF* assessments, but it is not the only acceptable way to identify limiting locations. This process is intended to be effective for both PWRs and BWRs, and with both ASME Code, Section III [9] / ANSI/ASME B31.7 [10] and ANSI/ASME B31.1 [5] piping methods. This process can be used to screen plant locations in order to rank EAF values on a consistent basis. Industry experience has shown that while *CUF* values may exist for ASME Code, Section III and ANSI/ASME B31.7 piping, sometimes the degree of rigor used to create the *CUF* values is inconsistent or the design *CUF* values may be overly conservative. When situations such as these exist, additional efforts may be used to reduce the *CUF* values in order to more carefully compare these locations to those identified in NUREG/CR-6260 [3]. This report discusses four methods for refining *CUF* values. These ranked locations can then be compared to the NUREG/CR-6260 sample locations and may augment a plant's Fatigue Management Program (FMP).

The desired outcome of the SIA EAF screening method process is to determine plant locations that require EAF management throughout the extended operating period. Typically, EAF management is conducted through a combination of transient cycle counting, fatigue monitoring, or periodic inservice inspection (ISI) coupled with a flaw tolerance evaluation.

The SIA EAF screening method procedure developed for this report has the following properties:

- No need for new formal stress or fatigue analysis.
- Includes guidance for methods that can be applied to refine EAF.
- Includes procedures that are practical to use, with readily available design input.
- Provides appropriate relative EAF rankings of components.
- Provides information regarding the use of either NUREG/CR-5704 [15] (for stainless steels), NUREG/CR-6583 [16] (for carbon and low alloy steels), both revisions of NUREG/CR-6909 [8, 14] (for nickel alloys), or both revisions of NUREG/CR-6909 [8, 14] for all materials.

In addition to the definitions provided in Section 2.1, the following terms specific to the SIA EAF screening method are used in this section:

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} in the Common Basis Stress Evaluation Procedure.

$U_{en\ incr}$: Incremental U_{en} for a load pair produced from U_{incr} and F_{en} values for each transient pair as $U_{en\ incr} = F_{en} \times U_{incr}$.

$U_{en\ incr}$*: 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_{6260}: The *CUF* for a NUREG/CR-6260 location.

$U_{en\ 6260}$: U_{en} produced from the *CUF* for a NUREG/CR-6260 location. $U_{en\ 6260} = F_{en} \times U_{6260}$.

$U_{en\ 6260}$*: 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} and/or U_{en}* in a set of locations.

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

3.1 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* (if appropriate) 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 [3]. 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 or that a Sentinel Location is not needed for a particular Thermal Zone when the CUF_{en} has been conservatively demonstrated to be less than unity.

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.

3.1.1 Outline of Process Steps

The screening process consists of the following elements, which are summarized below and described in more detail in Section 3.2:

- Gathering of Required Inputs (i.e., data collection)
- Determination of Thermal Zones
- Identification of Materials and Candidate Locations
- Establishing F_{en} and U_{en} values within each Thermal Zone
- Ranking U_{en} values by material within each Thermal Zone
- Identification of Sentinel Locations

1. **Data Collection**

   Data collection is necessary to equip the user to perform screening evaluations.

   In general, this is limited to Class 1 locations analyzed to ASME Code, Section III [9] and / or ANSI/ASME B31.7 [10]. This data collection is accomplished by obtaining the fatigue analyses, which contain input data such as component geometry and material properties, plant transient characteristics and design cycles.

   In some instances, for Class 1 locations analyzed to ANSI/ASME B31.1 [5] that cannot be dispositioned as described in Section 2.8 and require EAF assessment, input data such as component geometry and material properties and plant transient characteristics are required to compute relative stress and estimate *CUF* values for evaluated components.

   For all Class 1 locations needing to be evaluated, projections of plant transients for the licensed operating period and the extended operating period based on accumulated cycles to date are useful in computing projected *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. The differences in stresses experienced by each component in a Thermal Zone are generally the result of material and geometry differences when stress evaluation methods are similar. Components are assigned to appropriate Thermal Zones and evaluated as a group. This allows comparative rankings to be determined. For BWRs, plant-specific transients are provided in Thermal Cycle Diagrams for the reactor vessel and reactor vessel nozzles and are usually listed in the plant Final Safety Analysis Report (FSAR). For PWRs, transients are typically provided in NSSS System Standards Specifications and most are also usually listed in the plant FSAR. EPRI has compiled sample transient definitions for operating plants in MRP-393 [22].

3. **Identification of Materials and Candidate Locations**

   Locations in plant components are evaluated to establish *relative CUF* and CUF_{en} values keeping in mind the differences that can result depending on the level of rigor used in the analysis (as noted above regarding complicating factors).

   Locations in each Thermal Zone are then evaluated on a consistent analytical basis. This consistent approach mitigates the skewing effects of refined analyses (such as elastic-plastic analysis) for selected components. The purpose of ranking on a consistent 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 for each material. Once the leading location or locations have been identified for a particular material in a Thermal Zone, then the *CUF* value for that or those leading locations may be reduced by reanalysis to determine if the CUF_{en} is less than unity or less than the NUREG/CR-6260 location.

   For Class 1 locations analyzed to ASME Code, Section III [9] and/or ANSI/ASME B31.7 [10], this evaluation is accomplished by reviewing the fatigue analyses classified as Time-Limited Aging Analyses per 10 CFR 54.3. *CUF* values are obtained and grouped by Thermal Zone and material type. The level of rigor applied in the analysis (e.g., design by rule, design by analysis) is also noted, if known. This grouping is done to establish a consistent basis for ranking locations within each Thermal Zone.

   Furthermore, if the consistent basis CUF values are relatively large, the *CUF* value of ASME Code, Section III or ANSI/ASME B31.7 piping may be reduced if desired through a consistent use of one or more of the following methods:

   i. Revision of the *CUF* evaluation based upon load pairing.

   ii. Recalculating *CUF* using projected cycles for the extended operating period instead of design cycles.

   iii. Use of a later Code edition analysis (for example, later Code Editions of the Subsection NB Code reclassified the linear thermal gradient stress intensity from a secondary stress to a peak stress, thereby reducing K_e).

   iv. Application of ASME Code Case N-779 [11] alternative rules for calculation of K_e such that K_e is reduced.

   v. Calculation of *TF* and *GF* per ASME Code Case N-902 [12].

   vi. Application of ASME Code Case N-904 [13] alternative rules for calculation of K_e such that K_e is reduced.

In addition, F_{en} *Estimation Evaluations* may be applied for all components. The F_{en} *Evaluation Procedure* is developed to calculate or estimate F_{en} for locations in plant components on the basis of the relevant parameters – *DO*, maximum temperature and estimated tensile strain rate – for the leading transient(s). This procedure can be used as a source of F_{en} values for plants both with and without fatigue design analyses. A detailed description of the procedure is provided in Section 3.2.1.5. As noted in Section 1.1, the GALL [1], Revision 2 and GALL-SLR [2] state that additional plant-specific component locations in the reactor coolant pressure boundary that may be more limiting than those considered in NUREG/CR-6260 [3] may need to be managed. Therefore, for Class 1 piping locations analyzed to ANSI/ASME B31.1 [5] that cannot be dispositioned as described in Section 2.8 and require EAF assessment, a simpler and less resource-intensive method can be applied to provide reasonable assurance that the common basis evaluation methods described below.

In the case where one or more plants of similar design having Class 1 locations exist that were analyzed to ASME Code, Section III [9] or ANSI/ASME B31.7 [10], an engineering review can be performed to determine whether ANSI/ASME B31.1 locations in addition to those considered in NUREG/CR-6260 [3] required EAF management. As part of this review, a comparison should be performed to assess whether there are significant differences in plant operation and cycle accumulation rates that would suggest these ANSI/ASME B31.1 locations warrant management or not. In the case where plants of similar design do not have Class 1 locations that were analyzed to ASME Code, Section III [9] or ANSI/ASME B31.7 [10], an evaluation could be performed to assess similarities in system design configuration, materials of construction and operating parameters.

In the event that the aforementioned engineering review is unsuccessful and in keeping with the principle that no formal fatigue analyses are required to perform this screening for Class 1 locations analyzed to ANSI/ASME B31.1 [5], locations in each Thermal Zone may be evaluated with a *Common Basis Stress Evaluation* approach (discussed in Section 3.2.1.6). 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.

The Procedures for determining candidate locations are developed to support:

- 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, 80-year) projected numbers of transients.

4. **Ranking and Identification of Sentinel Locations**

For Class 1 locations analyzed to ASME Code, Section III or ANSI/ASME B31.7, U_{en} is determined by multiplying the design *CUF* by the calculated F_{en}.

For Class 1 locations analyzed to ANSI/ASME B31.1 for which comparison to a similar plant design is used, an estimated U_{en} (U_{en}*) is determined by comparing relevant parameters between the plants. In most cases, the estimated U_{en} will be less than 1.0 or less than the NUREG/CR-6260 location for the relevant Thermal Zone and will not be retained as a Sentinel Location.

If it becomes necessary to include Class 1 locations analyzed to ANSI/ASME B31.1 for which a similar plant analysis is not available, the *Common Basis Stress Evaluation* procedure can be used to obtain an estimated U_{en} (U_{en}*) by multiplying the common basis *CUF* by the estimated F_{en}.

Those locations within each group with the highest U_{en} or 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.

Note: Once initial Sentinel Locations are identified using the above approach, further refinement of the Sentinel Locations may be performed for these locations. Where it can be demonstrated that a reasonably conservative U_{en} value of less than 1.0 is achieved at the end of the extended operating period, the location can be removed from the list of Sentinel Locations requiring management. This can result in a situation where not every Thermal Zone has a Sentinel Location. Consideration must be given to ensuring that assumptions used in obtaining U_{en} values of less than 1.0 (e.g., *DO*, transient severity) are periodically validated during the extended operating period.

Note: Chapter X.M1 of the GALL-SLR Report [2] states that plant-specific justification can be provided to demonstrate that calculations for the NUREG/CR-6260 locations do not need to be included.

The result of this screening process is a listing of fatigue-sensitive plant components, organized into groups, ranked by U_{en} or U_{en}* severity. Thus, a Sentinel Location may represent several other locations. This information can be used to identify that bounding EAF locations in each Thermal Zone are monitored by the plant FMP, and that such locations can serve as early warning beacons and action triggers for components which might approach U_{en} = 1.0 based on cycles projected to the end of the licensed operating period.

3.1.2 Process Development Assumptions and Characteristics

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

- Screening U_{en} values can significantly exceed a value of 1.0 due to the considerable conservatisms in the process (e.g., use of bounding F_{en} values).
- The screening process is not intended to qualify components.
- Estimated F_{en} method is sufficient for a screening process.
- Except for relatively large diameter components, pressure transients are insignificant contributors to fatigue usage compared to temperature transients.
- Components which are exposed to reactor water, but do not form part of the RCPB (e.g., reactor internals not connected to the reactor vessel but having fatigue calculations) do not require screening.
- The F_{en} factor need only be applied for increasingly tensile portions of transients, based on the guidance of MRP-47, Revision 1 [23] and the modified strain rate approach of NUREG/CR-6909, Revision 1 [8].
- 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 ASME Code Case N-792-1 [24].
- Design severity transients (can use actual severity, if available and consistently applied) are used.
- Components of differing materials of construction within each Thermal Zone are evaluated with the appropriate material-unique F_{en} values.
- Common basis stress evaluation is only used for ANSI/ASME B31.1 [5] piping without an explicit fatigue analysis or a piping segment or system for which a comparable fatigue analysis of a similar piping segment or system is not available for comparison.
- Several characteristics of the *Common Basis Stress Evaluation* 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.
  - Geometric factors are applied to stress terms.
  - 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.

3.1.3 Determining Thermal Zones

For the purpose of this process, a *Thermal Zone* is defined as a collection of piping and/or vessel components which are evaluated to essentially the same group of thermal transients during plant operations. The idea of Thermal Zones is analogous to the way that design documents 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 based on the design set of transients or, if applied consistently, the actual transients that occur in service.

An important step in this process is dividing plant systems into Thermal Zones. For each major component or system, one or more Thermal Zones must be determined based on similar thermal transients. Where thermal cycle diagrams or histograms are unavailable for Class 1 piping analyzed to ANSI/ASME B31.1, operating procedures, design specifications and piping isometric drawings are used to determine which components undergo essentially the same set of 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. This step requires a detailed understanding of the transients and/or supporting evaluations. Some components may be considered to be part of two adjacent Thermal Zones.

For instance, the Class 1 portions of the chemical and volume control system (CVCS) in a PWR are 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 or crossover 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 3.2.1.2 for more detail of the process).

3.1.4 Technical Basis for F_{en} Evaluation

This section describes the technical basis and procedure developed to compute F_{en}, estimated F_{en} (F_{en}*), U_{en} and estimated U_{en} (U_{en}*) values for locations in individual plant components.

Note: It is preferable to calculate F_{en} and U_{en} instead of F_{en}* and U_{en}* where there is a DSR or plant data that provides adequate information to calculate these values with reasonable accuracy.

For the purpose of this screening, the rules for calculating F_{en} values may either be taken from (1) NUREG/CR-5704 [15] for stainless steel material, NUREG/CR-6583 [16] for carbon/low alloy steel material and NUREG/CR-6909 [8, 14] for nickel alloy material, or (2) from NUREG/CR-6909 [8, 14] for all materials.

These rules allow calculation of F_{en} factors based on the material at a given location and the following environmental parameters:

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

Note: Sulfur content of the metal, *S*, is also a factor for carbon and low alloy steels. However, this procedure will conservatively assume all carbon and low alloy steel components have the worst possible sulfur content.

F_{en} 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 or based on plant-specific operating parameters. Specifically:

- Any components which have no exposure to the "environment" (i.e., primary 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 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, consistent with the strain rate threshold values in the applicable NUREGs.
- 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 the revision to Code Case N-792 that determined it is acceptable to include the stress due to K_e in the strain rate calculation.
- 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 *DO* is typically maintained below 0.04 ppm when water temperature is ≥ 150°C (302°F), with rare exceptions. 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, Revision 0 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.

- Where "average temperature" is used, the higher of the threshold temperature from the applicable NUREG for the material under consideration or the minimum transient temperature will be used for the lower temperature to be averaged, consistent with the guidance in NUREG/CR-6909, Revision 1.

Where information is insufficient to calculate a reasonably accurate F_{en} value 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 estimated F_{en} for each component. This two-part estimated 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 a reasonable approach, but other methods for assigning F_{en} values may be used, as justified by the user.

Table 3-1
Strain rate categories

Strain Rate Category	Estimated $\dot{\varepsilon}$ (%/sec)
Extreme	≥ 5.0
Very High	~ 1.3
High	~ 0.33
Mid-High	~ 0.087
Medium	~ 0.023
Low-Mid	~ 0.0059
Slow	~ 0.0015
Very 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.

3.1.5 Technical Basis for ANSI/ASME B31.1 Piping Common Basis Stress Evaluation

If needed, the *Common Basis Stress Evaluation Procedure* may be used for Class 1 piping analyzed to ANSI/ASME B31.1 [5] that cannot be dispositioned as described in Section 2.8 and require EAF assessment to compute an estimated *CUF* (U^*), estimated F_{en} (F_{en}^*) and estimated U_{en} (U_{en}^*) values.

3.1.5.1 Rationale for the Common Basis Stress Evaluation Procedure

The evaluation incorporated in the *Common Basis Stress Evaluation Procedure* for components analyzed to ANSI/ASME B31.1 that cannot be dispositioned as described in Section 2.8 and require EAF assessment is based on the underlying rules of ASME Code, Section III NB-3600 / ANSI/ASME B31.7 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 / ANSI/ASME B31.7 are appropriate.
- The NB-3600 / ANSI/ASME B31.7 equations are explicitly defined and require minimal analyst interpretation so that they can be easily included in a spreadsheet.
- The NB-3600 / ANSI/ASME B31.7 rules are representative of the more general rules of ASME Code, Section III, NB-3200 design by analysis, which are appropriate for all plant components or for more complex geometries with a complex thermal response that require more detailed treatment.

For cases where a component or location has no explicit design fatigue analysis available (e.g., ANSI/ASME B31.1 piping) and EAF cannot be dispositioned as described in Section 2.8, the user may need to assess EAF. Furthermore, for such situations, it may not be possible to demonstrate through comparison that the location is less limiting than the corresponding location for the similar plant design evaluated in NUREG/CR-6260. In such cases, it may be necessary 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* for Class 1 components for such situations:

- 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 stress concentration factors (SCFs) or fatigue strength reduction factors (FSRFs) taken from the ASME Code.

Taking guidance from the EPRI Fatigue Management Handbook [6], 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, so they may be combined using linear superposition. Non-linear plasticity effects are accounted for using K_e in accordance with ASME Code Subarticles NB-3200, NB-3600 [9] or ANSI/ASME B31.7 [10]. 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 but a less strenuous level of analysis.

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 [25]. 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. The user should confirm this before finalizing the evaluation.
- 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 applied to piping without an explicit fatigue analysis to determine approximate stress ranges arising from pairs of selected significant transients, compute alternating stress values including 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.

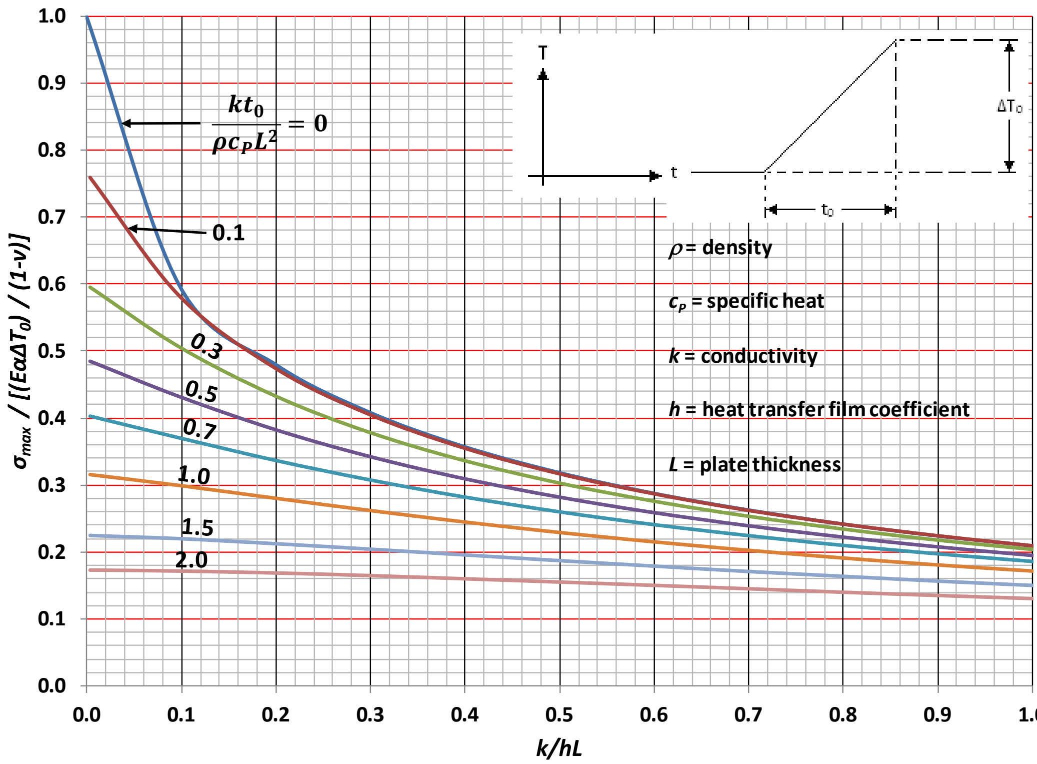


Figure 3-1
Determination of transient stresses for ramp transients [25]

3.1.5.2 Formulas and Equations

The evaluation addresses the following equation.

$$S_{peak} = S_{press} + S_{mom} + S_{disc} + S_{ts} \quad \textbf{Eq. 3-1}$$

(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_{disc} = peak stress range 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_{disc}: 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: Pipe inside 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 (ΔT), transient time (t_0), transient pressure change (ΔP).
- Computed values are heat transfer coefficient (h), k/ht, $kt_0/\rho c_p t^2$, normalized thermal stress (σ_{max} / $[(E\alpha\Delta T)/(1-\nu)]$) 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:

$$h = 0.023\left(\frac{\rho V D}{\nu}\right)^{0.8} Pr^{0.4}(k/D) \qquad \textbf{Eq. 3-2}$$

(valid for turbulent flow conditions, Reynolds No., $Re > 2000$)

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

$$h = 12.92874\left(\frac{Pr^{0.4}k}{\nu^{0.8}}\right)\left(\frac{Q^{0.8}}{D^{1.8}}\right) \qquad \textbf{Eq. 3-3}$$

$$h = \Phi\left(\frac{Q^{0.8}}{D^{1.8}}\right) \qquad \textbf{Eq. 3-4}$$

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

$$\Phi = 56.45 + 1.270T - 2.927x10^{-3}T^2 + 3.952x10^{-6}T^3 - 2.654x10^{-9}T^4 \quad \textbf{Eq. 3-5}$$

where:

h	=	heat transfer coefficient (Btu/hr-ft^2-°F)
ρ	=	density (lbf/ft^3)
V	=	velocity (ft/sec)
D	=	pipe inside diameter (inches)
v	=	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.

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

3.1.5.3 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_{alt} from S_{peak}, 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, however, the recommended approach for ranking is to use the design basis number of events. If the projected number of events is used, the new cycle limit may need to be reflected in the plant FMP.

- Compute transient pair $U_{en\ incr}$* from U_{incr}*and F_{en}*.
- Sum all transient pair $U_{en\ incr}$* into component U_{en}*.

3.1.5.4 Comparison of ANSI/ASME B31.1 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 ANSI/ASME B31.1 Class 1 piping components that cannot be dispositioned as described in Section 2.8 and require EAF assessment is made to compare the elements of the screening process to the ASME rules. NB-3600 uses a similar approach as ANSI/ASME B31.7, which was a predecessor to NB-3600. All of the NB-3600 equations are provided for reference in Appendix B.

A comparison of the Common Basis Stress Evaluation process is made to the basis equations of NB-3600 (provided in Appendix B). 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 has been addressed in the component design and will not be evaluated for screening.

NB-3653.1 Equation (10) [9] is accommodated in the screening by calculating the secondary stress range used to compute the elastic-plastic penalty factor, K_e, from Equation (14) of NB-3653.1 [9].

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 NB-3653.1 Equation (11):

$$S_p = K_1 C_1 \frac{P_O D_O}{2t} + K_2 C_2 \frac{D_O}{2I} M_i + K_3 C_3 E_{ab} |\alpha_a T_a - \alpha_b T_b| + \frac{1}{2(1-\upsilon)} K_3 E\alpha |\Delta T_1| + \frac{1}{1-\upsilon} E\alpha |\Delta T_2| \quad \textbf{Eq. 3-6}$$

The first three terms of Equation (3-6) are represented by the first three terms of Equation (3-1). 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 (3-6) is represented by the last term of Equation (3-1). The fifth term of Equation (3-6) 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, the fifth term can be ignored.

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_{disc}: ignore when calculating S_p, although it's included in establishing S_n and K_e
- For S_{ts}: evaluate by determining the maximum through-wall thermal stress using the graphical relationship in Figure 3-1.

Equations (12) and (13) of NB-3653.1 [9] 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.

Equation (14) of NB-3653.1 [9] 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.

3.1.5.5 Limitations and Assumptions of the Common Basis Stress Evaluation Process

Application of this process is generally limited to piping lacking explicit fatigue design calculations. This is because piping with explicit fatigue design calculations generally constitute the fatigue analysis of record for the plant and is not limited by some of the inherent simplifications of this process noted in the paragraphs that follow.

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 NUREG/CR-6260 locations at a minimum).

3.2 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 FMP. 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).

3.2.1 Detailed Screening Procedure

3.2.1.1 Gather Required Inputs for all Systems Containing Class 1 RCPB Components

For Class 1 locations analyzed to ASME Code, Section III [9] or ANSI/ASME B31.7 [10], obtain a copy of the fatigue analyses. These contain information regarding *CUF* based on fatigue tables that contain information on material properties, plant transient characteristics and design cycles.

For all Class 1 locations, obtain the following:

- Materials (austenitic stainless steel, carbon steel, low alloy steel or nickel alloy)
- DO history (for the contained fluid) from plant chemistry data
- List of thermal transients
- Projections of plant transients for the licensed operating period and the extended operating period based on accumulated cycles to date

For all Class 1 piping locations, obtain the following:

- Layout (connectivity and flow paths)

For Class 1 locations analyzed to ANSI/ASME B31.1 [5] where the *Common Basis Stress Evaluation* process is applied, determine the following data for the components in each system:

- Geometry (*ID*, *OD*, material/geometric discontinuities)
- Fatigue Strength Reduction Factor for vessels or piping stress indices (e.g., K_2 and K_3)
- List of thermal and pressure transients
- For the leading transients (available from stress report or transient monitoring program)
  - T_{max}, T_{min} and T_{ave} (determined based on T_{max} and lower threshold temperature from applicable NUREG or T_{min}, whichever is larger)
  - Material properties (S_m, k, E, α, ρ, c_p) as a function of 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 piping and instrumentation diagrams (P&IDs) 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 DSR:
  - Material properties as a function of temperature
  - Properties of each leading transient (see above)
- Either DSR or FSAR or transient monitoring system:
  - Moments, stresses, and loadings (see above)

3.2.1.2 Determine Thermal Zones

For each major component / system, Thermal Zones must be determined based on similar transients. In some cases, there may be multiple Thermal Zones and in other cases, they may be combined so that the more severe Thermal Zones encompass locations that experience less severe transients. For Class 1 locations analyzed to ASME Code, Section III or ANSI/ASME B31.7, thermal cycle diagrams / transient histograms or similar are available. For Class 1 locations analyzed to ANSI/ASME B31.1 that cannot be dispositioned as described in Section 2.8 and require EAF assessment, operating procedures, design specifications and piping isometric drawings are used to determine which components undergo essentially the same set of 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 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 the components within the boundaries.

A BWR example of a thermal zone boundary would be at the interface between the reactor recirculation piping and the reactor water cleanup (RWCU) piping or the residual heat removal (RHR) piping where there are localized thermal transient effects caused by isolation and restoration of RWCU flow or initiation and isolation of RHR flow for shutdown cooling.

A PWR example of a thermal zone boundary would be at a pressurizer and reactor coolant system (RCS) hot leg where there are localized effects due to insurge / outsurge and thermal stratification in addition to RCS hot leg transients.

For another PWR example of a thermal zone boundary, consider the CVCS. It is comprised of piping that connects the RHX to the cold leg charging nozzle(s) and the auxiliary spray tee on the pressurizer spray system and from the cold or crossover 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, noting that the user should consider the effects of valve alignments on section determination. 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 transients 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 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. It may also 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.

3.2.1.3 Identify Materials and Candidate Locations

For each Thermal Zone, select candidate locations, including each material associated with component locations.

In general, candidate locations are logically established based on major components (e.g., reactor vessel, pressurizer) and by system function.

When using the Common Basis Stress Evaluation procedure, determine candidate locations by examining nozzles and wall thickness transitions and piping nozzles, elbows, tees and other geometric 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).

3.2.1.4 Calculate U_{en} or 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 the *CUF* values fit a consistent basis model?
    - If YES, go to U_{en} Evaluation Procedure (Part A).
    - If NO, indicate what method was used (e.g., NB-3200 or NB-3600 / ANSI/ASME B31.7) and whether lumping of transients was performed, then go to U_{en} Evaluation Procedure (Part A).

- If NO:
  - Are CUF values and other information are available to assess consistency and method used?
  - If Yes, go to U_{en} Evaluation Procedure (Part A).
  - If No, go to Common Basis Stress Evaluation Procedure (Part B).

The common basis / consistency 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. In determining leading EAF locations, the level of rigor used in CUF calculations across all locations should be understood to assure proper relative ranking of fatigue results.

For CUF locations, a consistent 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 U_{en} evaluation procedure (Part A) below can be used. Otherwise, a common basis stress evaluation CUF may need to be calculated (Part B).

3.2.1.5 U_{en} Evaluation Procedure

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

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

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

F_{en} Formulations

The F_{en} *Evaluation Procedure* allows the user to select from the F_{en} formulations in NUREG/CR-5704 [15] and NUREG/CR-6583 [16], or NUREG/CR-6909 [8, 14]. The details of these formulations are given in Appendix A. The user must select whether to use the older guidance [15, 16] or the newer guidance [8, 14]. Use of the newer guidance is mandatory for SLR [2].

Determine Input Values

For each plant location, determine the following five variables:

- **CUF:** The consistent or common basis *CUF*.
- **Material Type:** Either stainless, low alloy, carbon, or nickel alloy steel.
- **DO**
- **T_{max}:** The maximum fluid temp to which the component is exposed (°F), which must be converted to °C in the F_{en} calculation. The value of the average temperature of each load pair (T_{ave}) may be used instead of T_{max}, keeping in mind the "average temperature" used must be consistent with the clarification that was added to NUREG/CR-6909, Revision 1 [8].
- **Strain Rate:** A calculated value or qualitative estimate of the strain rate ($\dot{\varepsilon}$) for each transient pair or the typical or controlling fatigue transient(s), based on knowledge of the corresponding plant system. If not calculated, the strain rate must be specified as one of the categories listed in Table 3-1. The "Expected F_{en}" is obtained by applying one of these strain rates to calculate F_{en}.

***F_{en}* Evaluation Procedure**

Note: When this procedure is applied to Part A, the strain rate defined above is used. This may involve only one F_{en}/F_{en}* for the location or one for each transient pair. 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}*.

1. If the strain rate is unable to be calculated from the information in the DSR and must be estimated, Compute F_{en}* for each location according to the specific NUREG guidance document; see Appendix A for details.
2. Compute Maximum F_{en} for each location. Calculate as above using 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: $F_{en}^* = (\text{Expected } F_{en} + \text{Maximum } F_{en})/2$.

Final Step for *U_{en}* Evaluation Procedure

Compute U_{en} for each location as: $U_{en} = F_{en} \times CUF$ or, where it must be estimated, $U_{en}^* = F_{en}^* \times CUF$.

3.2.1.6 Common Basis Stress Evaluation Procedure (if needed)

Where an engineering review was unable to determine that other ANSI/ASME B31.1 locations in addition to those considered in NUREG/CR-6260 [3] required EAF management, the *Common Basis Stress Evaluation Procedure* may be used to calculate an estimated (pseudo) *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 stainless, low alloy, carbon, or nickel alloy steel.
   - Maximum *DO* level during normal operating conditions.
   - M_i, I, (T_a-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, including T_{init}, T_{final}, t_0, P_{min}, P_{max}, and $n_{projected}$.
2. Calculate several dependent parameters:
   - h = convective heat transfer coefficient (use the formulas in Section 3.1.5.2).
   - t = wall thickness = $(OD - ID)/2$
   - ΔT_o = temperature change = $(T_{init} - T_{final})$
   - $\dot{\varepsilon}$ = estimated strain rate = $\Delta T_o / t_0$ (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 is 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 defined as 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 $\sigma_{max}/(E\alpha\Delta T_o/(1-\nu))$. Note: The equations in Figure 3-1 use L for wall thickness, instead of t. Multiply that value by $E\alpha\Delta T_o/(1-\nu)$ 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_{peak} for each transient pair as: $S_{peak} = K_1S_{press} + K_2S_{mom} + K_3(S_{ts,max} - S_{ts,min})$.
3. Compute S_n for each transient pair as: $S_n = C_1(S_{press,max} - S_{press,min}) + C_2(S_{mom,max} - S_{mom,min}) + C_3(S_{disc,max} - S_{disc,min})$.
4. Compute K_e for each transient pair using the rules in NB-3653.6, with $K_e = 1.0 + [(1-n)/n(m-1)]\ (S_n/3S_m - 1)$ within the bounds of $3S_m < S_n < 3mS_m$.
5. Compute the alternating stress amplitude for each pair as $S_{alt} = S_{peak}/2 \times K_e$.
6. Compute N_{allow} for each pair using the appropriate fatigue curve from either NUREG/CR-6909 or the ASME Code.

Perform Pseudo 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, 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 the 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.

3.2.1.7 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} or $U_{en}*$ values as $U_{en\ 6260}$ or $U_{en\ 6260}*$. Sort all remaining locations by U_{en} or $U_{en}*$ from highest to lowest value.
2. Identify candidate Sentinel Locations for final consideration.
   - Is $U_{en\ max}$ or $U_{en\ max}* \geq 1.0$
     - If NO, evaluate candidate Sentinel Location $U_{en\ max}$ or $U_{en\ max}*$ 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 2 within 25%?
    - If NO, evaluate 1 candidate Sentinel Location in Step 3, remove all other locations from further consideration.
    - If YES, evaluate 2 candidate Sentinel Locations in Step 3, remove all other locations from further consideration.

3. Consolidate candidate Sentinel Locations with any resident NUREG/CR-6260 U_{en} or U_{en}* location(s)
   - Does the Thermal Zone include a NUREG/CR-6260 location?
     - If YES:
       - Is candidate Sentinel Location U_{en}* > 1.0 and ≥ highest $U_{en\ 6260}$* location?
         - If YES, the candidate Sentinel Location is promoted to an official Sentinel Location.
         - If NO, remove from candidate Sentinel Location list.
     - If NO and U_{en}* > 1.0, 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 Major Component/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 FMP.

3.2.1.8 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 significantly greater than 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.

- *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 [1], 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…"*

  This is allowed within the guidelines for NRC review of GALL-SLR [2], which states on page X.M1-3: *"Plant-specific justification can be provided to demonstrate that calculations for the NUREG/CR–6260 locations do not need to be included."*

- *A location with U_{en} or U_{en}* < 1.0 may be removed from the Sentinel Location list*

  If the Sentinel Location U_{en} or U_{en}* for the projected number of design cycles is below 1.0, that Sentinel Location may be removed from the final list if there is reasonable assurance the value is obtained conservatively (i.e., inputs used for calculating U or F_{en} can be shown to be conservative relative to expected or actual values).

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.

4
WEC EAF SCREENING METHOD

The purpose of this section of the report is to describe the WEC EAF screening method along with the corresponding technical bases for the approach. The approach may be used to establish a comprehensive list of lead indicators for assessing the effect of EAF in the Nuclear Class 1 reactor coolant pressure boundary components. This list of lead indicators, also referred to as Sentinel Locations, supplements the locations previously identified in NUREG/CR-6260 and is determined by performing a review and comparison of locations susceptible to the effects of the reactor water environment. The list of Sentinel Locations determined with this approach can be used by a plant to support their fatigue management and monitoring programs.

The methodology presented in this section is applicable to all RCPB components, which consist of both equipment and piping locations, evaluated to the guidance in ASME Code, Section III [9], ANSI/ASME B31.7 [10], or ANSI/ASME B31.1 [5] that require EAF assessment. This approach was originally developed for application to PWR plants but may also be applied to BWR plants. A plant may utilize any of the industry documents on EAF evaluations discussed in Appendix A of this report with this approach. To facilitate the implementation of recent industry information on the topic discussed in Section 2.4, this methodology includes consideration of the potential differences in design fatigue curves between the fatigue analysis of record and the fatigue curves prescribed in NUREG/CR-6909, Revision 0 [14] and Revision 1 [8], when applicable.

The WEC EAF screening approach utilizes extensive design and analysis information and experience to consistently compare plant component locations without performing new analyses or extensive calculations in the screening process. The method is supported by comprehensive definitions of regions exposed to similar transients, referred to as Transient Sections. The Transient Sections definition provides a framework to compare locations based on relative effects of the reactor water environment using F_{en} screening assignments. It further supports comparison of U values based on a ranking system for the complexity of the analysis relative to the other locations. This Stress Basis Comparison ranking system provides a consistent characterization of the level of conservatism in the fatigue analysis, which can be used to differentiate between, for example, a simple analysis with a large U and a complex analysis with a moderate U value. Using the combined levels of F_{en} and Stress Basis Comparison ranking, the comparison and selection process identifies potential Sentinel Locations, first within Transient Sections, then between Transient Sections in a system, and ultimately between systems, to minimize the number of EAF locations for consideration. Guidelines are also provided for refined evaluation methods to assist in further reductions of both the initial list of Sentinel Locations and corresponding screening U_{en} values.

4.1 Technical Basis

EAF is quantified by a U_{en} value, which is the product of the component U determined using an air fatigue curve and application of an F_{en} multiplication factor to account for the LWR environment. The value of U for a component is a function of the component stress variations and cycles. The stress variations are influenced by the component material, geometry, and transient loadings. The transient loadings are typically influenced by variations in temperature, pressure, and force and/or moment loadings. Experience has shown that fatigue and EAF in Nuclear Class 1 components are most limiting in components subjected to relatively severe thermal transients. Stress ranges are typically dominated by the effects of temperature shocks. Ranges in stress due to forces and moments are generally related to the ranges of the thermal transient temperatures. In the component fatigue analyses, the material, geometry, and fundamental transient loadings (i.e., design transients) are fixed. The variable aspects of the fatigue analysis include modeling and stress calculation methods, simplification of loading applications by bounding or grouping transients or their effects, and conservatism in assumptions that influence various factors in the stress calculation process (such as the elastic-plastic penalty factor, K_e). The F_{en} multiplication factor is influenced by component material, temperature, strain rate, and *DO* content in the reactor water. The temperature and strain rate are also influenced by the transients and stress calculation methods discussed above. Therefore, consistent comparison of components that are influenced by these factors needs to address the relative effects of the variable aspects of the evaluation. The bases for assessing these in the comparison process are discussed in the following subsections.

4.1.1 Transient Section Technical Basis

A Transient Section is defined as a group of sub-components/locations that experience the same transients (i.e., thermal and related loadings). The concept of Transient Sections is typically used in design fatigue evaluations of system components for efficiency of application and is also effective for the screening process. Components that reside in the same Transient Section can be first compared with each other to determine the most limiting component (or Sentinel Location). For locations within a given Transient Section evaluated with common stress analysis methods, the differences in stresses experienced by each component are generally the result of the material and geometry differences and can be quantified. A typical piping system or major equipment will be divided into several Transient Sections. Often, it is the section 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.

The Transient Sections are developed based on knowledge of the system function in relation to plant transients, system layouts and flow paths, and/or equipment configurations. This is typically determined from the fatigue analysis of record, since common transient local effects required for the analysis are defined for various groups of components. For systems without an explicit fatigue analysis (e.g., B31.1 piping), the Transient Sections can typically be defined using system descriptions or specifications if available, or more directly by comparison to similar systems where a fatigue analysis was required and performed.

Components that reside in different Transient Sections but are within a common system or piece of major equipment can also be compared to determine the leading locations to represent their respective system/equipment. However, the comparison of components in different Transient Sections must be done after the appropriate F_{en} multiplication factors are applied to the component usage factors. This is because the F_{en} multiplication factor is dependent on temperature and strain rate and, therefore, can also vary for each Transient Section. Therefore, Transient Sections provide a consistent framework for comparison based on stress and fatigue as well as environmental effects.

Consider a PWR pressurizer for an example of a Transient Section definition. During operation, the pressurizer upper head is subject to transients that are greatly influenced by the pressurizer steam temperature as well as the spray operation and temperature. The pressurizer lower head region is influenced by pressurizer liquid temperatures as well as surge line fluid temperatures and flows. This difference in localized transient behavior within a single piece of equipment will induce differences in calculated fatigue usages; therefore, the uniqueness of these regions would constitute two Transient Sections.

A common example of a Transient Section for PWR Class 1 piping is the accumulator safety injection system. For certain PWR configurations the accumulator line reactor coolant loop nozzle and interconnected piping systems are used for accumulator injection, high head safety injection (HHSI) and low head safety injection (LHSI)/RHR, as illustrated in Figure 4-1 in Section 4.4.2. Each of these systems is separated from the primary flow path through the accumulator line to the cold leg by a series of valves. Therefore, the Transient Sections for the accumulator safety injection systems are typically defined by the isolated and combined flow paths for the given system functions. In this example, the section from the cold leg to the first check valve is defined as the first section due to the combination of the reactor coolant system transients and the local injection-related transients. The second Transient Section is upstream of the first check valve to the common header for the separate systems and is intended to address the combination of various local injection-related transients. Finally, the remaining three Transient Sections are defined based on the local transients associated with the specific system operation in each section. These differences in localized transient behavior within a single system, along with geometric differences in the piping layout, will induce non-trivial differences in calculated fatigue usages; therefore, these differences constitute the definition of multiple Transient Sections for the single piping system.

4.1.2 Stress Basis Comparison Technical Basis

A major consideration in the comparison process for a comprehensive screening assessment is the fact that different stress analysis techniques may have been used for each component usage factor calculation. For example, presume there is a component that was analyzed using simplified analytical methods and yielded a usage factor of 0.8. Also, presume there is another component in the same Transient Section that had a usage factor of 0.8, but was qualified using plastic analysis methods. Although both 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 comparison must consider the various stress analysis methods and techniques (i.e., technical rigor) that were used in the usage factor evaluation to provide a consistent basis of comparison.

When performing such an assessment, the technical rigor characteristics considered in determining the limiting locations within a given Transient Section include:

1. Qualification Criteria
2. Stress Analysis Method
    a. Simplified or One-Dimensional Analysis
    b. Finite Element Analysis
        i. Thermal
        ii. Mechanical
    c. Elastic/Plastic Analysis

To perform these comparisons and define the relative complexity of the various methods, a hierarchy of technical rigor was presented in PVP2014-29093 [26] based on a significant database of Class 1 component fatigue analyses and analytical experience. The hierarchy used is presented below, ordered from the least complex to the most complex methods within typical ASME Code, Section III, Subarticles NB-3200 and NB-3600 [9] analyses. In general, fatigue analysis performed to NB-3200 criteria are regarded as more complex than those performed to NB-3600 criteria. Combinations of the various methods are assessed on a case by case basis. Note that for primary equipment, which are typically evaluated to NB-3200 criteria, the ranking ranges are typically isolated to the subcategories of Category 5. PVP2020-21678 [27] presents a shorthand notation from one (most conservative) to three (least conservative), which may be used to replace 5a to 5c. An example of the use of this shorthand notation is shown in the example in Section 4.4.1.

1. Standard NB-3600 analysis
2. NB-3600 with mechanical FE stress quantities substituted in stress formulas
3. NB-3600 with thermal FE stress quantities substituted in stress formulas
4. Combination of 2) and 3)
5. NB-3200 fatigue analysis:
    a. NB-3200 with interaction analysis or stresses derived with closed-form solutions
    b. NB-3200 with elastic FE analysis
    c. NB-3228 plastic analysis

In applying the Stress Basis Comparison, elimination of the location with the lower final screening U_{en} value and analysis method ranking is justified, since, if it were analyzed with the same rigor as the retained location, its U would be even lower and result in a lower U_{en}. For cases where the comparison of the U_{en} and Stress Basis Comparison rank combination is inconclusive, multiple Sentinel Locations may be initially retained, and additional evaluation or scoping analyses may be used to determine the most limiting component for a given Transient Section or system.

4.1.3 Similarity Comparison Technical Basis

Some applications of the EAF screening process will include Class 1 piping systems designed to the ANSI/ASME B31.1 Code, which does not include requirements for explicit calculation of fatigue usage factors. Instead, cyclic loads in B31.1 piping systems are evaluated by applying stress intensification factors on the thermal expansion stress range equation that compares to an allowable stress value. The stress intensification values are based on fatigue tests of piping components and include consideration of stress cycles. In spite of this design methodology difference, system functions, flow paths, and typical components in Class 1 piping systems designed to B31.1 are essentially the same as Class 1 piping designed to ASME Code, Section III, or other standards that require explicit fatigue usage calculation (e.g., ANSI B31.7). Therefore, it is valid to generally conclude that a piping system designed to the B31.1 rules would meet the fundamental intent of the ASME Code, Section III rules if reanalyzed. This rationale is discussed in detail in EPRI TR-102901 [17]. Rather than the significant effort of reanalyzing B31.1 piping system components to a standard such as ASME Code, Section III to obtain fatigue usage values, it is valid for screening purposes for B31.1 locations that require EAF assessment to demonstrate similarity to a piping system with existing usage factors and to use the applicable values from that "reference" piping system to perform EAF screening.

Since the EAF screening process is a comparison based on relative fatigue sensitivity, the fatigue results from a similar reference piping system will appropriately identify the limiting locations corresponding to the B31.1 system components. If the B31.1 piping were reanalyzed to a standard like ASME Code, Section III, the analysis would employ the same methods to define Transient Sections, compute loadings, and calculate stress and fatigue usage. Therefore, B31.1 piping systems can be systematically compared to similar piping systems designed to ASME Code, Section III or similar standards (reference system analysis), based on the considerations listed below:

1. **Piping system functions and flow paths comparison to the reference system**: Demonstrate that the plant B31.1 piping system performs similar functions to the reference piping system and identify flow paths for each of the functions, especially where multiple functions are performed (e.g., some PWR safety injection and RHR piping).
2. **Transient Section definition consistent with reference analysis**: Based on functions and flow paths, the plant system is divided into Transient Sections corresponding to those in the reference system analyses, as described in Section 4.1.1. This comparison step should include consideration of applicable thermal loading conditions that are included in the current licensing basis analysis.
3. **Identification of components within each Transient Section**: Compile the individual piping components in each Transient Section of the plant system.
4. **Mapping and reconciliation of piping components in each Transient Section to the reference analysis** considering:
    a. **Component type:** Piping component types are assigned generally based on the component types such as those defined in ASME Code, Section III, Subsection NB-3680. Some component characteristics that influence the stress calculations should also be considered and reconciled. Examples include curved pipe bend radius, tapered transition angles, branch connection types and dimensions, etc.

b. **Material:** Material types are compared with respect to their effect on stress calculations and allowable values. Where material grades are different, application of the reference analysis may be justified by demonstrating conservatism or introducing appropriate scaling factors for comparison.

c. Geometry

   i. **Nominal thickness/schedule:** These parameters are compared with respect to their influence on the piping component stresses, including pressure, moment, and through-wall thermal transient stress.

   ii. **Discontinuity:** For components with structural discontinuity, characterized in NB-3653 by parameters t_a and t_b, the relative impact of these parameters on the comparative approach to the screening process is assessed.

d. **Moment load stress:** For the purposes of the comparative screening process, it is appropriate to assume the adequacy of the reference analysis with respect to moment stress based on the following:

   i. Piping expansion moment ranges are generally related to the thermal transient load ranges in the piping system Transient Sections.

   ii. For fatigue significant piping components, the moment stresses alone are not typically controlling, independent of the thermal transient stresses.

   After the screening process has identified the initial set of limiting locations, final detailed comparisons of remaining potential Sentinel Locations, performed to reduce the final number of locations, should consider the effects of the plant system moment loads. Additionally, detailed EAF evaluations of the final identified Sentinel Locations should consider plant specific moment loads.

e. **Applicability of any unique inputs in reference analysis:** If the reference analysis included unique inputs relative to the component, such as non-standard stress indices, or use of refined stress analysis techniques (e.g., finite element analysis, or FEA), the applicability of these is accounted for in the comparison.

The system comparison demonstrates the similarities between the piping system of interest and the reference analysis, such that fatigue results for the reference system can be used with the EAF screening process to determine the leading locations. The similarity reconciliation may require adjustments for the reference analysis usage factors to be applied to plant specific components for the EAF screening process. The bases for the similarity reconciliation are maintained for re-consideration in any subsequent refined screening comparisons of Sentinel Locations.

4.1.4 F_{en} Technical Basis

When performing an EAF screening evaluation, the applicable F_{en} values are a critical input to the analysis and the derivation of the U_{en} values. Consistency in the F_{en} calculations and related inputs supports a consistent comparison of locations. For screening, the F_{en} values are not determined by detailed integrated methods, such as the modified rate approach described in NUREG/CR-6909 Revisions 0 and 1, but instead use more general inputs and assumptions to provide a consistent basis for F_{en} assessment in the comparison of locations. The F_{en} values employed in the screening process are based on the inputs to the F_{en} formulas:

- Material
- Strain rate
- Temperature
- Dissolved oxygen

The four generic input parameters to the F_{en} equations are related to the applicable industry documents, typically NUREG/CR-5704 [15] for austenitic stainless steels, NUREG/CR-6583 [16] for carbon and low alloy steels, and NUREG/CR-6909, Revision 0 [14] for nickel alloy steels, or NUREG/CR-6909, Revision 1 [8] for all materials. For the carbon and low alloy materials, a conservative assumption is sufficient to address the sulfur content in an effort to reduce the amount of research required into plant records. The remaining parameters of interest are established with the intent to develop a conservative and consistent set of F_{en} values for screening comparisons.

The estimated strain rate value chosen is intended to maximize the transformed strain rate term, which has a significant impact on the magnitude of the F_{en} values. This assumption is employed because a generically established strain rate can be determined for a given Transient Section; however, the local effects of the components within the Transient Section would be required to determine if a given component is more limiting based on the contribution of the strain rate to the derivation of the F_{en} values. Furthermore, design transients with fast temperature changes were employed to maximize the through-wall temperature gradient and corresponding stresses. The operation at a given plant may induce a less severe transient, however, the reduced rate of temperature change in the transients will have offsetting effects in the derivation of the F_{en}, as noted in PVP2020-21545 [28]. Therefore, for the purposes of the initial EAF screening comparison, the use of a strain rate value to maximize the transformed strain rate is appropriate.

The metal temperatures associated with components are established by reviewing the critical transients defined for a given Transient Section and selecting an upper bound temperature for each Transient Section. This can have a significant effect in screening out systems or locations with lower service temperatures that are less susceptible to the effects of EAF.

To avoid artificially skewing the EAF sensitivity of a given material type, the DO content is determined using a slightly more detailed approach to establish a reasonable, conservative value for screening. For example, for a PWR, a value of 0.005 ppm is normally used for the DO content, which is typical of the PWR environment based on the EPRI published primary water chemistry guidelines [29] for use by the PWR fleet. For PWRs, DO is generally well below 0.005 ppm during normal operation except possibly during early heatup and late cooldown conditions, which could be confirmed by a review of plant specific data or procedures.

Furthermore, an approach based on plant operation can be used to justify critical inputs to the F_{en} assessment, such as reviewing the product of the transformed *DO* level (O^*) and the transformed material temperature (T^*) to illustrate that the combination of parameters results in a set of conservative F_{en} values for the plant specific application. A similar approach for BWR chemistry could be used.

4.2 Screening F_{en} Development

F_{en} factors are developed for each component so that U_{en} can be calculated. Note that the F_{en} multiplication factors calculated herein are conservatively calculated for screening purposes. Four different industry documents on EAF have been issued from 1998 to 2018 to establish a consistent process and requirements for analyzing environmental fatigue of materials exposed to light water reactor environments. Due to this variation, Table 4-1 presents representative maximum F_{en} values for each applicable industry document on EAF to highlight the potential variability for a PWR environment. As previously noted, the formulas outlined in NUREG/CR-6909 can be used for stainless steels, carbon and low-alloy steels, and Ni-Cr-Fe alloys. The formulas outlined in NUREG/CR-5704 can be used for stainless steels. The formulas outlined in NUREG/CR-6583 can be used for carbon and low-alloy steels. The resulting maximum F_{en} factors per NUREG/CR-5704, NUREG/CR-6583, and NUREG/CR-6909 are shown.

Table 4-1
Maximum F_{en} values by material and NUREG for PWRs

Material	Parameter	NUREG/CR-5704	NUREG/CR-6583	NUREG/CR-6909 Revision 0	NUREG/CR-6909 Revision 1
Stainless Steel	**Maximum F_{en}**	15.348	N/A	14.514	12.81
	Strain Rate, $\dot{\varepsilon}$ < (%/sec)	0.0004	N/A	0.0004	0.0004
	Dissolved Oxygen Content < (ppm)	0.005	N/A	0.005	0.005
	Service Temperature	200°C (392°F)	N/A	325°C (617°F)	325°C (617°F)
	Maximum F_{en} at low temperatures	2.547 below 200°C (392°F)	N/A	2.083 below 150°C (302°F)	1.00 below 100°C (212°F)
Carbon (CS) and Low Alloy Steels (LAS)	**Maximum F_{en} =**	N/A	2.532(LAS) 1.740 (CS)	2.018(LAS) 1.881 (CS)	6.276
	Sulfur Content > (weight %)	N/A	0.015	0.015	0.015
	Strain Rate, $\dot{\varepsilon}$ < (%/sec)	N/A	0.001	0.0004	0.0004
	Dissolved Oxygen Content < (ppm)	N/A	0.05	0.04	0.04
	Service Temperature	N/A	350°C (662°F)	350°C (662°F)	325°C (617°F)
	Maximum F_{en} below 150°C (302°F)	N/A	2.532(LAS) 1.740 (CS)	2.018(LAS) 1.881 (CS)	1.736
Ni-Cr-Fe- Alloys	**Maximum F_{en} =**	N/A	N/A	4.524	3.746
	Strain Rate, $\dot{\varepsilon}$ < (%/sec)	N/A	N/A	0.0004	0.0004
	Dissolved Oxygen Content < (ppm)	N/A	N/A	0.005	0.005
	Service Temperature	N/A	N/A	325°C (617°F)	325°C (617°F)
	Maximum F_{en} at Temp. < Service Temp.	N/A	N/A	Calculated at Max Temp	1.00 below 50°C (122°F)

4.3 EAF Screening Process Steps

The steps described below present the process elements of the WEC EAF screening method that is used to determine the list of Sentinel Locations.

1. **Data Collection:** All of the pertinent inputs, including information on the applicable locations identified in NUREG/CR-6260, must be collected. This includes all of the materials, drawings, and CLB fatigue evaluations, if they exist. Any location that was not part of the Class 1 RCPB should be removed from consideration. If the results of this task indicate design differences between comparable components within a unit, the information pertaining to the design differences is evaluated for consideration in the comparisons and then consolidated as part of Step 4. If a fatigue evaluation is not available for a given plant/location, a fatigue evaluation from a similar location or plant could be used, consistent with the discussion in Section 4.1.3, to determine the required information. Locations are also excluded during this step based on the following criteria:

    a. Not in contact with primary coolant.

    b. Locations excluded from fatigue usage factor calculation based on fatigue waivers from ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NB.

    c. Locations with a U of 0.000.

2. **Transient Section Definition:** For this step, the Transient Sections for all applicable piping systems and equipment included in the screening evaluation must be determined. Components within common Transient Sections will be evaluated initially as a group before they are compared against other components within the same system or equipment.

    As previously discussed in Section 4.1.1, the Transient Sections are developed based on knowledge of the system function in relation to plant transients, system layouts and flow paths, and/or equipment configurations. This is typically determined from the fatigue analysis of record, since common transient local effects required for the analysis are defined for various groups of components. For systems without an explicit fatigue analysis (e.g., B31.1 piping), the Transient Sections can typically be defined using system descriptions or specifications if available, or more directly by comparison to similar systems where a fatigue analysis was required and performed. Further discussion on this comparison is presented in Section 4.1.3.

3. **Screening Environmental Fatigue Multiplier Calculation:** In this step, the fatigue information collected in Step 1 is combined with the Transient Section definitions established in Step 2 to determine a screening U_{en} value for each location susceptible to the effects of the light water reactor environment. The result of this step is an initial list of leading locations that will be further examined in the subsequent steps to determine the plant specific list of Sentinel Locations.

    a. Organize the locations susceptible to EAF identified in Step 1 into the Transient Sections defined in Step 2.

b. Adjust the U values by any applicable factors to correct for differences between the fatigue curves used in the source fatigue evaluation (e.g., Section III Appendix I of the ASME Code) and the fatigue curves applicable to the industry document used to determine the screening F_{en}, as required. This factor is represented by F_{adj} and the result of this calculation is U_{adj}. Note, this step is typically only required when the F_{en} from NUREG/CR-6909 Revision 0 or Revision 1 is used for screening.

c. Apply the maximum F_{en} of all materials to all components corresponding to the F_{en} formulas from the applicable industry EAF document. If the screening U_{en} is less than unity, the location can be removed from the potential Sentinel Location list. Refer to Section 4.2 for further details on the development of the applicable F_{en} factors.

d. For the remaining potential Sentinel Locations, determine the maximum F_{en} and F_{adj} for each component based on actual material. These material specific F_{en} and F_{adj} values are used to determine a screening U_{en} for each component (designated material F_{en}, F_{adj}, and U_{en}). Perform this calculation following the F_{en} formulas and design fatigue curves outlined in the appropriate industry EAF document for the application. If the U_{en} is less than unity the location can be removed from the list of potential Sentinel Locations. Retain at least one location per Transient Section, for example a U_{en} close to 1.0 if none exceeds 1.0, for completeness at this stage. Further treatment of these locations is addressed in Step 4a.

e. As applicable, calculate reduced screening F_{en} factors for each component in each Transient Section simply based on the maximum temperature experienced in the section, in an effort to reduce the screening U_{en} from Step 3d to a value below 1.0 (designated temperature F_{en} and U_{en}). Note that more rigorous F_{en} refinements are addressed later in the process in Step 4e, and also in Section 4.5.

4. **Sentinel Location Identification:** Step 4 establishes the Stress Basis Comparison ranking for the detailed comparison between components and the corresponding down-selection of the leading locations for EAF. The result of this step is the plant specific list of Sentinel Locations.

   a. Remove components with a material or temperature (Steps 3c through 3e) screening U_{en} of less than 1.0 from the potential Sentinel Location list. The screening U_{en} values for these locations are conservative based on the approach used to derive the screening F_{en} values. Therefore, a detailed evaluation would be expected to result in a lower U_{en} value, so further evaluations would not be required for locations with a screening U_{en} less than unity.

   b. Identify the locations with the maximum screening U_{en}, for each applicable material type, in each Transient Section.

   c. Determine the Stress Basis Comparison ranking for each remaining component.

      i. Determine the level of technical rigor and qualification criteria for each component within the Transient Section.

ii. Qualitatively determine the most limiting components in each Transient Section, using a consistent stress analysis method ranking basis for comparison. This ranking is based on the amount of conservatism considered in the analysis and is described in Section 4.1.2. Note that, while evaluating the Stress Basis Comparison for a given location, it must be confirmed that the U value used corresponds to a location that is in contact with primary coolant, in accordance with Step 1a. In some instances, this information may not be known at Step 1a, but would become evident in the details needed to determine the stress analysis basis for comparison. If the location of interest corresponds to a surface not in contact with primary coolant, the corresponding or next most limiting surface in contact with primary coolant must be considered in the Stress Basis Comparisons. This removes from consideration potential high-U value locations which are not impacted by environmental effects.

iii. For each Transient Section, systematically compare each location to the maximum screening U_{en} location considering the Stress Basis Comparison ranking. Remove locations with both a lower screening U_{en} and lower analysis rank, until the minimum number of locations is established. The goal is to identify one location in the Transient Section.

d. Compare sentinel components of different Transient Sections within common systems or equipment. This may require additional Stress Basis Comparisons to determine one or two Sentinel Locations per system or equipment.

e. The list of potential Sentinel Locations can be further reduced without detailed analysis by removing conservatism from the fatigue analysis of record (including application of 80-year projected cycles) or further refinement of the F_{en} multiplication factors.

f. Compare candidate Sentinel Locations against any NUREG/CR-6260 locations within the system. Those components with a screening U_{en} less than the screening U_{en} for the NUREG/CR-6260 location are removed from the final set of Sentinel Locations. The final list of Sentinel Locations should be included in the fatigue aging management plan for the plant and addressed as part of the license renewal application.

4.4 Application of Methodology

The following examples illustrate the application of the WEC EAF screening process to a plant specific data set. The first example, Section 4.4.1, illustrates the process applied to an equipment location. In this example a pressurizer is examined relative to the guidance presented in NUREG/CR-6909 Revision 1. The second example, Section 4.4.2, applies the WEC EAF screening process to a cold leg safety injection/accumulator piping system. The F_{en} values implemented in this example are based on the information contained in NUREG/CR-5704 and NUREG/CR-6583.

For each example, the applicable Transient Sections are defined, and the components are evaluated relative to locations exposed to similar regions of the LWR environment. As part of the down-selection process, the Stress Basis Comparison ranking is applied to each applicable location and used to determine the most fatigue sensitive locations susceptible to EAF. The final list of Sentinel Locations is summarized in each subsection to illustrate the final product of this approach.

4.4.1 Example: EAF Screening of Equipment Locations

The example for applying the WEC EAF screening process to a primary equipment component is provided below based on guidance consistent with NUREG-2192 [30]. For this example, the pressurizer upper head and shell region is the Transient Section of interest. This Transient Section is identified by first reviewing the entire pressurizer. As previously discussed in Section 4.1.1, the pressurizer is separated into two separate Transient Sections to address the differences in the thermal transients. It is worth noting that for the Transient Section presented in this example, Step3e does not have a significant impact on the results due to the high operating temperature of the pressurizer. However, this step may have a more significant impact for other Transient Sections exposed to lower operating temperatures.

Table 4-2 illustrates the application of the various steps of the methodology, starting from data collection (Step 1) through Stress Basis Comparison ranking for leading locations (Step 4c). Note, the Stress Basis Comparison ranking presented in Table 4-2 represents the shorthand notation for primary equipment discussed in Section 4.1.2. As shown in Table 4-2, only four locations have a screening U_{en} greater than unity after the maximum material and temperature specific F_{en} multiplication factors were applied (Steps 3d and 3e of Section 4.3). Each of the locations with a screening U_{en} greater than unity were analyzed using finite element analyses (Stress Basis Comparison ranking of 2, per Section 4.1.2), except for Location 4, for which stresses were derived by simple closed form solutions. Furthermore, the location with the largest screening U_{en} values is comprised of stainless steel, with the next limiting location being composed of low alloy steel followed by the carbon steel locations. The screening U_{en} values for the two carbon steel locations are of similar magnitude. However, when the Stress Basis Comparison rankings are reviewed, it is apparent that Location 3 required a more detailed analysis to produce a slightly higher U. Therefore, no further evaluation is required for Location 4, because if a similar technical rigor were applied to this location, it would remain less limiting than Location 3. Table 4-3 presents the refined set of Sentinel Locations for the Transient Section of interest.

The screening U_{en} values for the Transient Section Sentinel Locations identified in Table 4-3 all have values greater than unity for different materials, thereby confirming that additional actions such as analysis refinement (including reduction of U values and F_{en} penalty factors), cycle counting, inspection, or replacement should be used to manage EAF.

Table 4-2
Example application of EAF screening methodology for a plant specific transient section

Steps 1, 2, & 3a			Step 3b	Step 3c	Step 3d				Step 3e		Steps 4a & 4b		Step 4c
Transient Section	Component / Location	Design U	U_{adj}	Max U_{en}	Material Category	Material U_{adj}	Material F_{en}	Material U_{en}	Temp F_{en}	Temp U_{en}	Final F_{en}	Final U_{en}	Stress Basis Comparison Ranking
PZR - Upper Head/Shell	Location 1	0.848	8.480	108.629	Stainless	8.480	12.810	108.629	12.810	108.629	12.810	108.629	2
	Location 2	0.416	4.160	53.290	Low-Alloy	0.416	6.276	2.611	6.276	2.611	6.276	2.611	2
	Location 3	0.170	1.700	21.777	Carbon	0.170	6.276	1.067	6.276	1.067	6.276	1.067	2
	Location 4	0.160	1.600	20.496	Carbon	0.160	6.276	1.004	6.276	1.004	6.276	1.004	1
	Location 5	0.040	0.400	5.124	Low-Alloy	0.040	6.276	0.251					
	Location 6	0.007	0.070	0.897									

Note: The Step 4c Equipment Basis Comparison Ranking values are as follows (from Section 4.1.2):

1. NB-3200 with interaction analysis or stresses derived with closed-form solutions.
2. NB-3200 with elastic FE analysis.
3. NB-3228 plastic analysis.

Table 4-3
Example sentinel locations identified for a plant specific transient section

Transient Section	Component/Location	Material Category	Design U	Material U_{adj}	Final F_{en}	Final U_{en} (1)
PZR - Upper Head/Shell	Location 1	Stainless	0.848	8.480	12.810	108.629
	Location 2	Low-Alloy	0.416	0.416	6.276	2.611
	Location 3	Carbon	0.170	0.170	6.276	1.067

Note:

1. Final U_{en} = Material U_{adj} x Final F_{en}.

4.4.2 Example: EAF Screening of Piping Locations

The example for applying the WEC EAF Screening process to a piping system is provided below and based on guidance consistent with NUREG-1801 [1]. The example system is the cold leg safety injection/accumulator piping in a Westinghouse PWR, as depicted in Figure 4-1.

Upon gathering the applicable data, the system Transient Sections were defined consistent with the design transient definitions and component fatigue evaluations. An illustration of the Transient Sections for the piping system is shown in Figure 4-1.

Table 4-4 through Table 4-8 show the components in each section and illustrate the results of applying Steps 3 through 4c from Section 4.3; Step 3b is not applicable to this example. The Stress Basis Comparison ranking values correspond to the hierarchical list as described in Section 4.1.2.

As shown in Table 4-4, all components have a screening U_{en} greater than unity after the maximum material and temperature specific F_{en} multiplication factors were applied. All components were analyzed using NB-3600 equations (Stress Basis Comparison Ranking = 1) except for the reactor coolant loop (RCL) nozzle and the valve butt weld. The Transient Section 1 RCL nozzle was qualified to NB-3600 but using FEA for thermal and mechanical stress quantities (Stress Basis Comparison Ranking = 4). The Transient Section 1 valve butt weld was qualified to NB-3600 but using FEA only for thermal stress quantities (Stress Basis Comparison Ranking = 3). The final screening U_{en} for the Transient Section 1 valve butt weld is less than that of the Transient Section 1 RCL nozzle, and the Transient Section 1 valve butt weld was qualified using a less rigorous analysis methodology than the Transient Section 1 RCL nozzle. Therefore, it is concluded that the Transient Section 1 RCL nozzle is more limiting than the Transient Section 1 valve butt weld. The RCL nozzle is chosen as the limiting location from Transient Section 1.

In Table 4-5 for Transient Section 2, with the exception of the valve, all components have a screening U_{en} less than unity after maximum material and temperature specific F_{en} multiplication factors were applied. The valve was qualified to NB-3545 (Stress Basis Comparison Ranking = 1). Therefore, because it is the only location remaining, the valve is chosen as the limiting location from Transient Section 2.

As shown in Table 4-6 through Table 4-8, all components in Transient Sections 3, 4, and 5 have a screening U_{en} less than one after the maximum material and temperature specific F_{en} multiplication factors were applied. At this stage at least one location per Transient Section is considered. Therefore, the location with the highest screening U_{en} at the final step in each Transient Section is retained. For Transient Section 5, both the socket weld and elbow locations will be considered since the screening U_{en} in Step 3e is the same.

After applying Steps 1 through 4c for the five Transient Sections in the cold leg safety injection accumulator system, six potential leading locations remain and are listed in Table 4-9.

All potential locations in Transient Sections 3, 4, and 5 have a screening U_{en} less than unity; therefore, they can be eliminated from consideration. The basis for elimination is that a detailed EAF evaluation of these components would result in a lower screening U_{en} than that obtained using the bounding maximum material penalty, and therefore would remain below 1.0. The screening U_{en} for the Transient Section 2 valve is less than that of the Transient Section 1 RCL nozzle, and the Transient Section 2 valve was qualified using a less rigorous analysis methodology (lower Stress Basis Comparison ranking) than the Transient Section 1 RCL nozzle. Therefore, it is concluded that the Transient Section 1 RCL nozzle is more limiting than the Transient Section 2 valve. The RCL nozzle is chosen as the limiting location for the system. Furthermore, Step 4e applies for this system since the RCL nozzle is a NUREG/CR-6260 location. However, this step is trivial in this example since the nozzle is the final Sentinel Location for the system.

Similar methodology would be used to perform an EAF screening evaluation for this example if the piping was not designed to ASME Code Section III or ANSI B31.7. The additional similarity comparison discussed in Section 4.1.3 is implemented in the data collection step. Because the CLB for the Safety Class 1 piping does not include an explicit fatigue evaluation, plant-specific usage factors are not available for each piping component. Instead, the plant component fatigue qualifications available in a comprehensive database is utilized. Comparisons are made between the plant-specific piping components and those that are available in the database to justify the applicability and make relative comparisons for screening purposes. Such comparisons include, but are not limited to, materials, geometry, and transients.

Table 4-4
Safety injection/accumulator EAF screening – transient Section 1

Steps 1, 2, & 3a			Step 3c	Step 3d			Step 3e		Steps 4a & 4b		Step 4c
Transient Section	**Component / Location**	**Design U**	**Max U_{en}**	**Material Category**	**Material F_{en}**	**Material U_{en}**	**Temp F_{en}**	**Temp U_{en}**	**Final F_{en}**	**Final U_{en}**	**Stress Basis Comparison Ranking**
1	RCL Nozzle	0.95	14.58	Stainless	15.35	14.58	15.35	14.58	15.35	14.58	4
	Elbow	0.09	1.38	Stainless	15.35	1.38	15.35	1.38	15.35	1.38	1
	Butt Weld	0.10	1.54	Stainless	15.35	1.54	15.35	1.54	15.35	1.54	1
	Small Branch/Plug	0.10	1.54	Stainless	15.35	1.54	15.35	1.54	15.35	1.54	1
	Valve Butt Weld	0.54	8.29	Stainless	15.35	8.29	15.35	8.29	15.35	8.29	3
	Valve	0.48	7.37	Stainless	15.35	7.37	15.35	7.37	15.35	7.37	1

Note: The Step 4c Basis Comparison Ranking values are as follows (from Section 4.1.2):

1. Standard NB-3600 analysis
2. NB-3600 with mechanical FE stress quantities substituted in stress formulas
3. NB-3600 with thermal FE stress quantities substituted in stress formulas
4. Combination of 2) and 3)
5. NB-3200 fatigue analysis:
    a. NB-3200 with interaction analysis or stresses derived with closed-form solutions
    b. NB-3200 with elastic FE analysis
    c. NB-3228 plastic analysis

Table 4-5
Safety injection/accumulator EAF screening – transient Section 2

Steps 1, 2, & 3a			Step 3c	Step 3d			Step 3e		Steps 4a & 4b		Step 4c
Transient Section	**Component / Location**	**Design U**	**Max U_{en}**	**Material Category**	**Material F_{en}**	**Material U_{en}**	**Temp F_{en}**	**Temp U_{en}**	**Final F_{en}**	**Final U_{en}**	**Stress Basis Comparison Ranking**
2	Valve Butt Weld	0.09	1.38	Stainless	15.35	1.38	2.547	0.23			
	Elbow	0.06	0.92	Stainless	15.35	0.92					
	10"x10"x6" Tee	0.09	1.38	Stainless	15.35	1.38	2.547	0.23			
	Valve	0.48	7.37	Stainless	15.35	7.37	2.547	1.22	2.547	1.22	1

Note: The Step 4c Basis Comparison Ranking values are as follows (from Section 4.1.2):

1. Standard NB-3600 analysis
2. NB-3600 with mechanical FE stress quantities substituted in stress formulas
3. NB-3600 with thermal FE stress quantities substituted in stress formulas
4. Combination of 2) and 3)
5. NB-3200 fatigue analysis:
   a. NB-3200 with interaction analysis or stresses derived with closed-form solutions
   b. NB-3200 with elastic FE analysis
   c. NB-3228 plastic analysis

Table 4-6
Safety injection/accumulator EAF screening – transient Section 3

Steps 1, 2, & 3a			Step 3c	Step 3d			Step 3e		Steps 4a & 4b		Step 4c
Transient Section	Component / Location	Design U	Max U_{en}	Material Category	Material F_{en}	Material U_{en}	Temp F_{en}	Temp U_{en}	Final F_{en}	Final U_{en}	Stress Basis Comparison Ranking
3	Elbow	0.00									
	10"x3/4" Branch	0.09	1.38	Stainless	15.35	1.38	2.547	0.23			
	Valve Butt Weld	0.00									
	Valve	0.19	2.92	Stainless	15.35	2.92	2.547	0.48			

Table 4-7
Safety injection/accumulator EAF screening – transient Section 4

Steps 1, 2, & 3a			Step 3c	Step 3d			Step 3e		Steps 4a & 4b		Step 4c
Transient Section	Component / Location	Design U	Max U_{en}	Material Category	Material F_{en}	Material U_{en}	Temp F_{en}	Temp U_{en}	Final F_{en}	Final U_{en}	Stress Basis Comparison Ranking
4	Elbow	0.01	0.15	Stainless	15.35	0.15					
	6"x2" Branch	0.24	3.68	Stainless	15.35	3.68	2.547	0.61			
	Valve Butt Weld	0.01	0.15	Stainless	15.35	0.15					
	Valve	0.29	4.45	Stainless	15.35	4.45	2.547	0.74			

Table 4-8
Safety injection/accumulator EAF screening – transient Section 5

Steps 1, 2, & 3a			Step 3c	Step 3d			Step 3e		Steps 4a & 4b		Step 4c
Transient Section	**Component / Location**	**Design U**	**Max U_{en}**	**Material Category**	**Material F_{en}**	**Material U_{en}**	**Temp F_{en}**	**Temp U_{en}**	**Final F_{en}**	**Final U_{en}**	**Stress Basis Comparison Ranking**
5	Socket Weld	0.10	1.54	Stainless	15.35	1.54	2.547	0.25			
	Pipe	0.02	0.31	Stainless	15.35	0.31					
	Elbow	0.10	1.54	Stainless	15.35	1.54	2.547	0.25			

Table 4-9
Safety injection/accumulator EAF screening – potential sentinel locations

Transient Section	Component	Design *U*	Final F_{en}	Final U_{en}	Stress Basis Comparison Ranking
1	RCL Nozzle	0.95	15.35	14.58	4
2	Valve	0.48	2.547	1.22	1
3	Valve	0.19	2.547	0.48	
4	Valve	0.29	2.547	0.74	
5	Socket Weld	0.10	2.547	0.25	
	Elbow	0.10	2.547	0.25	

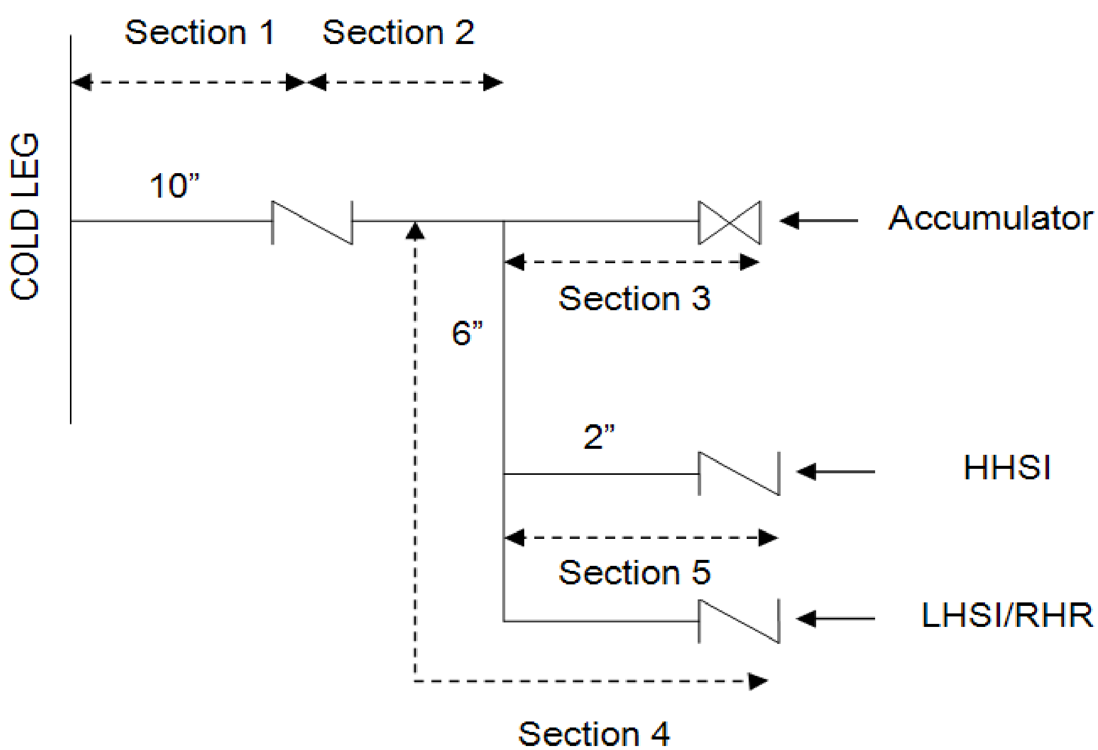


Figure 4-1
Safety injection/accumulator – transient sections

4.5 Conclusion

NUREG-1801 and NUREG-2191 require plants applying for license renewal to consider the effects of the light water reactor environment on fatigue for the sample set of components defined in NUREG/CR-6260, plus any other RCPB component(s) that may be more limiting. The primary objective of the WEC EAF screening process is to provide an approach that will develop an optimized list of Sentinel Locations, including consideration of the locations previously identified in NUREG/CR-6260, for assessing the effects of light water reactor environments. The methodology presented is applicable to all RCPB components evaluated to the guidance in ASME Code, Section III, ANSI/ASME B31.7, or ANSI/ASME B31.1 for locations that require EAF assessment. Furthermore, a plant may utilize any of the industry documents on F_{en} evaluations discussed in Section 2.4 of this report with this approach.

The WEC EAF screening approach utilizes design and analysis information coupled with analyst experience to consistently compare plant component locations without performing new detailed analyses. The method is supported by optimized definitions of regions exposed to similar transients, referred to as Transient Sections. The Transient Section definitions provide a framework to compare locations based on relative effects of the reactor water environment using F_{en} screening assignments. It further supports comparison of U and U_{en} values based on a ranking system for the complexity of the analysis relative to other locations. This Stress Basis Comparison ranking system provides a consistent characterization of the level of conservatism in the fatigue analysis, which can be used to appropriately compare the potential Sentinel Locations and minimize the final number of locations for explicit fatigue management. Furthermore, the screening U_{en} values for these locations are conservative based on the approach used to derive the screening F_{en} values. Therefore, detailed evaluations would be expected to result in lower U_{en} values. With the reduced set of potential Sentinel Locations obtained using the screening process, further refinement of the list of Sentinel Locations or ultimate evaluation of the final Sentinel Locations can be approached using methods such as the following:

- Application of projected cycles for the anticipated life of the plant.
- Ungrouping of thermal transients to provide less conservative stress profiles.
- Reduction in the number of zero stress states assumed in the analysis.
- Reduction in stress indices and stress concentration factors.
- Stress/temperature algorithm refinement such as using a finite difference solution of a one-dimensional heat transfer problem instead of closed form solutions.
- Application of improved approaches presented in later editions of the ASME code, such as the approach to address stresses beyond the yield strength in elastic analyses.
- Application of ASME code cases to refine and reduce stress or fatigue conservatisms.

The methods noted above assist in further reductions of both the list of Sentinel Locations and corresponding U_{en} values. In cases where improvements from later Code editions are employed in analysis refinements, requirements for Code reconciliation should be investigated and applied as required. The final list of Sentinel Locations determined with this approach can be used to support, or supplement, plant fatigue management and monitoring programs.

5 EDF EAF SCREENING METHOD

The purpose of this section is to describe the technical bases of the EDF EAF screening method. The EDF EAF screening method defines a process that may be used for EAF screening and ranking of components in nuclear power plant Class 1 systems.

5.1 Introduction

In the wake of numerous experimental tests carried out in air and also in a PWR environment, both abroad and in France, an update of the fatigue codification is underway.

In the U.S., the crack initiation fatigue design curve for austenitic stainless steels in the ASME Code [31] was reviewed. Then, a proposal to consider an aggravating factor that incorporated the PWR environment into EAF calculations for various types of materials was also formulated and was the subject of two ASME Code Cases [24, 32].

However, despite these efforts, international experience has been quite favorable, and no crack initiation has yet been ascribed to the effects of the PWR environment on Class 1 structures. EDF therefore concluded it was important to take a stand on the possible evolution of the fatigue design curves codified in the RCC-M Code [33] and on the assessment of environmental effects in a realistic manner.

Two RCC-M Code Cases (Rules in Probationary Phase, “RPP-2” for new design curve and RPP-3 for F_{en} factor incorporation) were developed in the frame of a French fatigue working group involving EDF, AREVA and CEA, and incorporated in the 2016 edition of the RCC-M Code. These RPPs focus exclusively on austenitic and cast duplex stainless steels as the other surfaces of ferritic steel components are either clad or the levels of *DO* in the water are too low for environmental effects to be significant.

The French Safety Authority (ASN) requested that EAF be considered in the stress report calculations for the fourth decennial inspection of the 900 MWe power plants. This has been done by implementing the new RCC-M codified rules for a certain number of Sentinel Locations that were identified using the screening methodology reported in the present section, and also reported in PVP2016-63125 [34]. That method was based on the methods documented in the draft of Revision 1 of NUREG/CR-6909 [35].

In the U.S., comparable calculations were also implemented as a part of license renewal efforts to operate power plants beyond their initial 40 years of operation. In order to reduce the scope of the calculations, preliminary screening was carried out on the various areas of the primary circuit as detailed in the EPRI Report 1024995 [4], which is updated in Section 3 of this report. The output of that screening process is a list of locations that are most prone to EAF degradation process and it is on these zones only that detailed EAF calculations are performed.

This section provides an update to the previous EDF EAF screening method based on the final publication of NUREG/CR-6909, Revision 1 [8]. The method is outlined, and the technical basis of the screening process is provided 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.

This section first provides a summary of the EDF EAF screening process. Then, the method is described that is used to screen the zones potentially impacted by the implementation of EAF in stress report calculations. This is followed by a description of the scope of the EDF screening method and its limits and exclusions are discussed. Finally, an example of the EDF EAF screening method is provided that applies the method to the primary circuit of the EDF 900 MWe reactor fleet.

5.2 Background

The previous EPRI EAF screening method [4] was used as a starting point for the EDF EAF screening methodology proposed in this section.

5.2.1 Purpose of the EDF Screening Methodology

With regard to fatigue evaluation, there are two different types of locations: locations deemed sensitive to fatigue degradation that exhibit a *CUF* that is already greater than or equal to 1.0, and locations with a fatigue degradation concern that have a *CUF* between 0.5 and 1.0.

The two main objectives of the EDF EAF screening process are:

- To identify all the locations which will have a *CUF* that exceeds 1.0 (or 0.5 – see below) after considering EAF: these locations are called "Sentinel Locations";
- To provide a ranking of the various locations within one system or sub-system (called "Thermal Zone") according to their updated *CUF* value.

During Periodic Safety Assessments (PSAs), detailed EAF calculations are carried out only for Sentinel Locations.

It should be emphasized that the locations that are already sensitive to fatigue are also included in the ranking, but they are not identified as Sentinel Locations.

In the event that a system or sub-system has no locations exceeding a *CUF* of 1.0, the limit of 0.5 is used to identify locations that could potentially become concerned when EAF is considered.

In addition to the requirements pertaining to PSAs, the screening also aims at providing an insight in the potential impacts on fatigue management: as highlighted above, the F_{en} approach may cause some transients to become significant contributors to fatigue that were not previously considered significant according to the former approach. If this is the case, this situation could bring modifications to the fatigue management requirements with a need to monitor different types of transients.

5.2.2 EDF Screening Methodology Requirements

The EDF EAF screening methodology that is detailed in this section has the following requirements:

- **Rely only on existing data:** The input is limited to the information that is contained in existing stress reports, either the original design basis reports or the reports prepared for the previous PSAs. No additional calculations should be performed for this screening;
- **Provide a relative ranking allowing various EAF methodologies and fatigue curves to be used:** The results of the EDF EAF screening methodology were developed before any ASN comments were received on the proposed modifications to the RCC-M Code. The original EDF EAF method has therefore evolved: the core methodology in this section has been updated to rely on the final published version of NUREG/CR-6909, Revision 1 [8];
- **Easy to apply through simple spreadsheet calculations:** The objective is simply to obtain a relative ranking implying no detailed calculations at this stage; and
- **Provide an accurate ranking considering the specificities of EAF calculations:** Taking EAF into account in calculations involves manipulating new quantities for fatigue evaluation. The F_{en} factor involves strain rate, which calls for a totally new post-processing approach; strain rate is a time-dependent variable in fatigue calculations that relied up to now only on peak-to-peak stress combination. Moreover, slow strain rates tend to increase the F_{en} (the F_{en} formulae are provided in Appendix A and will be discussed later) which implies that some transient combinations that were benign in the old approach could become significant. The methodology that is used needs to clearly assess these competing phenomena before starting the full-scale application to components.

5.2.3 Validity of the EDF Screening Process

When introduced in 2007, NUREG/CR-6909, Revision 0 [14] served as a basis for the ASME Code Cases that proposed modifications of the codification in three aspects:

- First, a modification of the mean fatigue curve in air;
- Second, an adjustment of the factors enabling to transition from the mean air curve to the ASME design curve. These factors are equal to 2 on the strain amplitude and 12 on the number of cycles in NUREG/CR-6909 (both revisions);
- Third, the introduction into the *CUF* calculations of a F_{en} factor that is multiplied by the partial usage factor contributions from the various transient stress combinations.

The RCC-M methodology (RPP-2 and RPP-3) is in line with this general approach. An additional $F_{en\text{-}integrated}$ value that accounts for the effects of the PWR environment already covered in the fatigue curve has been introduced. In the French methodology, the F_{en} factor described in NUREG/CR-6909 would then be divided by this $F_{en\text{-}integrated}$ value which means that the usage factors calculated with and without the $F_{en\text{-}integrated}$ are proportional. This is shown in Equation (5-4) and discussed further in Section 5.4.4.

Concerning the fatigue curves, the RPP-2 design curve is based on the Argonne National Laboratory (ANL) mean air curve to which the factors of 1.4 [36] on the strain amplitude and 10 on the number of cycles are applied [37, 38]. A mean stress correction based on the modified Goodman relationship is first applied to the mean curve, before applying the adjustment factor of 1.4 and 10 on strain amplitude and fatigue life, respectively. However, this mean stress correction has no practical effect on the design curve up to 10^6 cycles (end of its domain of application). This is due to the material parameters that were selected for this correction, representative of the RCC-M austenitic stainless steel grades.

As already mentioned earlier in this section, the work reported here was performed while discussions were still ongoing on the updated fatigue methodology (which is now incorporated in the RCC-M Code since its 2016 edition). For this reason, the following was proposed to carry out the screening:

- Using several fatigue curves all relying on the ANL mean curve but with different adjustment factors and/or input parameter values used for mean stress correction. One of the curves corresponds today to the RPP-2 curve introduced in the RCC-M Code (in the 2016 edition).
- Using the F_{en} factor defined in the draft of Revision 1 of NUREG/CR-6909 [35] in conjunction with the French proposal [39, 40].

The output is a ranking of all the zones potentially impacted by EAF, or Sentinel Locations. These locations were assessed more carefully through detailed calculations during 900 MWe PSAs.

5.2.4 Main Steps

The methodology is applied according to the following steps:

1. **Gather the necessary data:** The data necessary for the evaluation includes the partial usage factors, the transient pairing order and the local geometry which can be obtained from previous stress reports, as well as the description of the transients, the heat transfer coefficients, etc., which can be obtained from the transient description document;
2. **Divide the primary loop and auxiliary loops in austenitic stainless steel into "Thermal Zones":** A thermal zone is a zone that undergoes similar transient loadings and shares common geometrical and material characteristics. This calls for an understanding of the operation of the power plant which can be obtained through EDF training material;
3. **For each thermal zone, update the usage factors:** For the various locations of each thermal zone, the partial usage factors are updated using the new fatigue curves and then multiplied by the F_{en} factor to obtain a new cumulated usage factor, FU, incorporating EAF, also noted here FU_{en}. The F_{en} factor is determined directly from the shape of the transient (almost always cold or hot shocks) based on correlations between dimensionless numbers (such as the Fourier or the Biot numbers) and the F_{en} factor;
4. **Rank the zones according to their FU_{en}:** Within each thermal zone, a ranking is carried out in order to determine the area impacted the most by EAF calculations; and

5. **Select only representative areas or "Sentinel Locations" [39]:** The ranking then enables selection of a set of representative locations for each thermal zone, or "Sentinel Locations". Further detailed calculations should then be performed for these locations only as they cover the other locations within each thermal zone. A comparison with the thermal zones already defined as sensitive can be carried out at this stage.

5.2.5 Layout

The remainder of this section is divided into seven subsections, as follows:

- Section 5.3 summarizes the scope of the screening.
- Section 5.4 summarize the fatigue curves used and the latest F_{en} expressions from NUREG/CR-6909, Revision 1 [8], which is the version that is used for the screening. This section also presents brief descriptions of the French and the $F_{en\text{-}integrated}$ methodologies.
- Section 5.5 defines the main assumptions made for the screening approach.
- Section 5.6 provides a detailed description for implementing the screening methodology.
- Section 5.7 provides an illustration of the application of the screening methodology to a 900 MWe nuclear power plant (NPP) component, the Pressurizer Safety and Relief Line.
- Section 5.8 provides the conclusions for the screening method.
- Section 5.9 provides a summary of the work performed to update the screening approach to use the methods of NUREG/CR-6909, Revision 1 [8].

5.3 Scope of the Screening

The screening does not need to be performed on all the wetted surfaces of the 900 MWe NPP. Therefore, this section defines the limits of where the screening will be performed based on a preliminary high-level scoping.

On the primary side, EAF assessment is performed on austenitic stainless steel surfaces only. Low alloy and carbon steels are all clad with stainless steel in the RCS, so it is assumed that the environment will have no effect on these ferritic materials. Concerning nickel alloys such as Alloy 600 and Alloy 690, the maximum possible F_{en} values do not exceed 3.75: a high-level screening will therefore be conducted. With regard to cast duplex stainless steels, work carried out by AREVA in the frame of the Framatome Reactor Owners Group (FROG) Material Ageing Working Group Joint Program demonstrated that the behavior of this type of steel is similar to austenitic stainless steels.

On the secondary side, the wetted surfaces are all low alloy or carbon steels. The secondary side water chemistry does not contain the sufficient amount of dissolved oxygen for EAF to have any impact. This is only valid during operation; outside of normal operating periods, the secondary water contains more dissolved oxygen, but no loading occurs.

To summarize, the detailed screening is performed on the RCS austenitic and cast duplex stainless steels only. In addition, only the locations requiring a fatigue evaluation will be reassessed: this includes Class 1 components and piping only. This encompasses a list of components and locations which are summarized in Table 5-1.

Table 5-1
Overview of the locations to be assessed for the preliminary screening

Component	Part of the Component to Analyze	Type of Steel
Reactor Pressure Vessel (*Cuve*)	Safe Ends (*Embouts*)	Z2CND18-12 Az
	Thermocouple columns (*Colonnes de Thermocouples*)	Z2CN19-10 Az Z6CN18-10
	Control rod drive mechanism (*Mécanisme de Contrôle de Grappes*)	Z2CN18-10 Z5CN18-10
	Other reactor internals (*Autres internes de cuve*)	Z2CND17-12
Steam Generator Primary Side (*Côté Primaire du Générateur de Vapeur*)	Safe Ends (*Embouts*)	Z2CND17-12
Pressurizer (*Pressurisseur*)	Safe Ends (*Embouts*)	Z3CND17-12 Az
	Thermal Sleeves (*Manchettes Thermiques*)	Z2CND17-12
	Heater Well (*Manchette de Canne Chauffante*)	Z3CND17-12 Az
Primary Pumps (*Pompes Primaires*)	Pump Casing (*Volute de Pompe*)	Z5CN19-09M
Primary Piping (*Tuyauteries Primaires*)	Straight sections (*Sections droites*)	Z3CND17-12 Z2CND18-12 Az
	Elbows (*Coudes*)	Z3CND17-12 Z5CND19-10 Z5CND19-10 M Z3CND19-10 M
Auxiliary Piping (*Tuyauteries Auxiliaires*)	Pressurizer Expansion Line (*Ligne d'Expansion du Pressurisseur - LEP*)	Various heats of austenitic stainless steels [(1)]
	Pressurizer Safety & Relief Line (*Ligne de Décharge du Pressurisseur*)	
	Spray Line (*Ligne d'Aspersion*)	
	Safety Injection (*Injection de Sécurité - RIS*)	
	Residual Heat Removal (RHR) (*Régrigération à l'Arrêt – RRA*)	
	Chemical and Volume Control System (CVCS) (*Contrôle Volumétrique et Chimique – RCV*)	
	Blowdown Line (*Ligne de purge*)	
Valves (*Robinets*)	Valves (*Robinets*)	

Note: (1) The detailed materials were not specified for the piping components as these are all austenitic stainless steels and hence require environmental multiplier calculation.

5.4 Summary of Existing F_{en} Factor Expressions

Several EAF methodologies have been at the center of many discussions internationally. The NUREG/CR-6909, Revision 1 methodology [8] has the most widespread use and is the F_{en} method that is used in the EDF EAF screening method.

In addition, this section also introduces the $F_{en\text{-}integrated}$ approach, which is a quantity that has been proposed to the ASN as part of the French methodology and which is defined in more detail in EDF Document D02-ARV-01-062-695 [40].

5.4.1 F_{en} Calculation Thresholds

For austenitic stainless steels, various thresholds exist below which environmental effects are insignificant:

- **Positive (increasing) strain rate:** Many experimental results have shown that there are no effects in the case of a negative strain rate [8]. Only portions of the stress history with positive strain rates will be considered in the EAF calculations, while F_{en} equal to 1.0 is used otherwise;
- **Total strain amplitude above 0.10%:** For austenitic stainless steels, experimental data showed that, for very small strain amplitudes, there are no significant environmental effects.

5.4.2 F_{en} Expressions

The original EDF approach [41] was developed based on the F_{en} expressions and thresholds that were documented in the draft of Revision 1 of NUREG/CR-6909 [35]. The final publication of NUREG/CR-6909, Revision 1 [8] changed some of the F_{en} formulations for stainless steel compared to the draft report, specifically with respect to the transformed strain rate term. Therefore, the EDF EAF screening method is updated in this report to use the final F_{en} expressions for austenitic stainless steels from NUREG/CR-6909, Revision 1. Those F_{en} expressions are presented in Appendix A.2.3. Section 5.9 of this report provides the details of updating the EDF method and benchmarking to adopt the latest F_{en} expressions.

The F_{en} factor depends on the strain rate which is a quantity that is time dependent. In typical fatigue calculations, especially RCC-M B-3600 type calculations, it is not customary to deal with time dependent quantities: peak-to-peak stress combinations are used to calculate *CUF* without necessarily looking at portions of the stress histories between the extremes. A series of assumptions are made in order to calculate an accurate F_{en} factor.

5.4.3 Strain Rate Evaluation Method

In typical fatigue calculations, only the peaks in the stress time history are used. The stress time history between those extremes is not used.

For F_{en} calculations, the strain rate needs to be evaluated using the strain time history. Various methods have been proposed to integrate this quantity into calculations, the latest of which is ASME Code Case N-884 [42]. The method used in this section is an interpretation of the "Detailed Method" described in paragraph S-2220 of N-884. This interpretation corresponds to the method referred to in Section 6 of NUREG/CR-6909, Revision 1 [8] and referred to as the "Multilinear Strain Based Method".

The method consists in dividing all positive strain rate portions into small linear segments on which the F_{en} can be calculated using the minimum and maximum strain and the time step of the segment.

The F_{en} for a transient is then calculated as follows:

$$F_{en} = \frac{\sum F_{en,k}\dot{\varepsilon}_k}{\frac{\varepsilon_{max}-\varepsilon_{min}}{\Delta t_{tot}}} = \frac{\sum F_{en,k}\dot{\varepsilon}_k}{\sum \dot{\varepsilon}_k} \quad \textbf{Eq. 5-1}$$

where:

$F_{en,k}$ is the F_{en} calculated for the time step, k

$\dot{\varepsilon}_k$ is the strain rate calculated for the time step, k

ε_{max} is the maximum strain rate value of the transient considered

ε_{min} is the minimum strain rate value of the transient considered

Δt_{tot} is the total time for the entire transient event to occur

An illustration of the calculation of the F_{en} is given Figure 5-1.

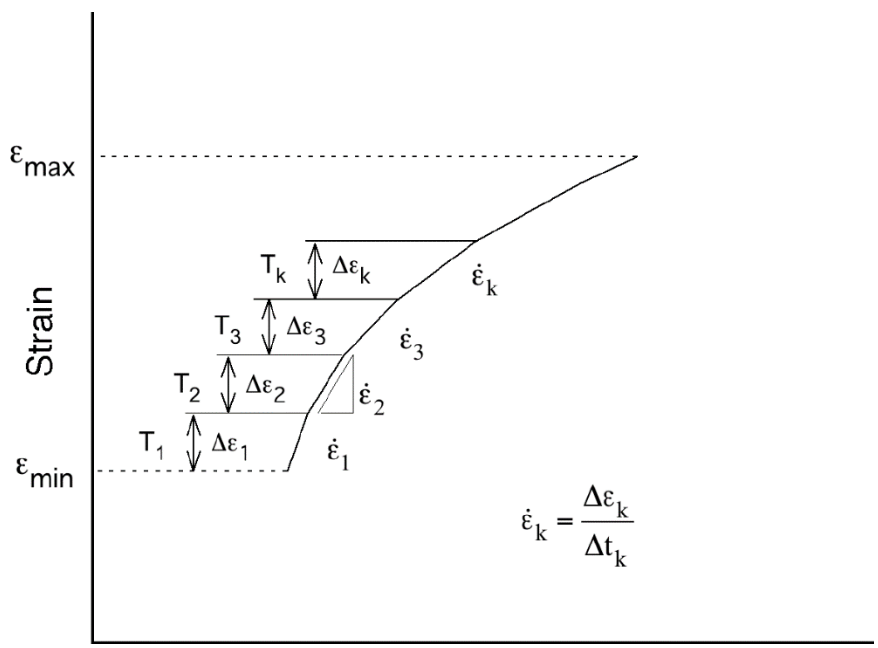


Figure 5-1
Calculation of the *F_{en}* using the multilinear strain based method [8]

5.4.4 Introduction to the $F_{en\text{-}integrated}$ Approach

As part of the proposal presented to the ASN [40], the concept of the $F_{en\text{-}integrated}$ approach was introduced. The concept was initially established by AREVA in the frame of the Olkiluoto (Unit 3) EPR reactor contract and is based on an AREVA test campaign in a PWR environment [43, 44, 45].

The objective of the testing campaign [43, 44, 45] was to simulate conditions as representative as possible to real power plant conditions, i.e. with a specimen surface finish close to the surface finish of the piping in NPPs and with strain loadings representative of cold and hot shock transients. The conclusions were that in all cases, the F_{en} from NUREG/CR-6909 was too conservative: this demonstrated that a part of the environmental effects were already partially covered in the design fatigue curves. The $F_{en\text{-}integrated}$ concept was therefore developed to quantify the part of environmental effects already integrated in the design fatigue curve.

In practice, this translates to a threshold on the F_{en}. The $F_{en\text{-}TOT}$ for a set of transient combinations for a given zone is calculated as follows:

$$F_{en-TOT} = \frac{\sum F_{en-m,n} \times FU_{m,n}}{\sum FU_{m,n}} \qquad \text{Eq. 5-2}$$

where:

$F_{en\text{-}TOT}$ is the F_{en} calculated for the selected zone for all transient combinations (m, n)

$F_{en\text{-}m,n}$ is the F_{en} calculated according to the methodology presented in Section 5.4.3 for the transient pair (m,n)

$FU_{m,n}$ is the partial usage factor resulting from the combination of the transients m and n

This quantity is then compared to the $F_{en\text{-}integrated}$. If $F_{en\text{-}TOT} < F_{en\text{-}integrated}$, this indicates that the multiplication of the usage factor by an F_{en} is not necessary. If $F_{en\text{-}TOT} > F_{en\text{-}integrated}$, then the fatigue curve does not cover all the environmental effects and the cumulative usage factor should therefore be multiplied by:

$$\frac{F_{en-TOT}}{F_{en-integrated}} \qquad \text{Eq. 5-3}$$

The $F_{en\text{-}integrated}$ is equal to:

- A minimum of 3.0; or
- 5.0 in the case of zones where hot shocks and cold shocks mainly contribute to the usage factor.

5.5 Assumptions

To perform this screening, a series of assumptions are made. These assumptions are presented in this section and serve the purpose of making the evaluation easy to implement while keeping the F_{en} calculation conservative.

- **The method is used here to perform a screening and is not sufficient to carry out a full qualification of the components**: The method only serves the purpose to screen out the locations which are the most prone to EAF degradation. Detailed calculations will be performed as a part of 900 MWe PSAs.

- **The F_{en} is calculated based on thermal transients only:** The stresses due to pressure and external loads are neglected for this screening. The F_{en} depends on the strain rate but also the temperature. The thermal transients dictate the temperature and the strain rate which justifies picking only these for the evaluation. The impact of thermal loads on the alternating stress used in usual fatigue evaluations is also the most important. Moreover, although the pressure and external loads increase the alternating stresses, they also have the effect of increasing the strain rate; this reduces the F_{en} value significantly and it is hence conservative to neglect these mechanical loads for EAF.
- **The F_{en} calculation will be based on evaluations on various pipe profiles:** The F_{en} value is based on calculations performed using an FEA software program called *code aster* on straight sections of various pipe profiles. It is assumed that nozzles or geometrical singularities such as welds do not affect the F_{en} calculation. This assumption can be justified by reviewing how the F_{en} is calculated; the F_{en} expressions are evaluated by considering the strain rate, the temperature and the level of *DO*. These three quantities are calculated on the inner surface of the component, so the overall component geometry has little influence.
- **The F_{en} correlations are established for austenitic stainless steels only:** The material properties that are used correspond to a type Z2-CND-18.10 steel from the RCC-M Code. The assumption holds valid for the EDF reactor fleet since most of the piping is made out of this grade of austenitic stainless steel or a grade with very close mechanical properties.
- **NUREG/CR-6909, Revision 1 expressions used:** As mentioned previously, the F_{en} expressions from NUREG/CR-6909 Revision 1 [8] will be used.
- **Partial usage factor combinations unchanged by incorporation of environmental effects:** In previous stress reports, the transient combinations that have led to the evaluation of the cumulative usage factor are assumed to remain unchanged when incorporating the F_{en} factor in the calculations.
- **Mean temperature is used to calculate F_{en} in the FEA *code aster* calculations:** The average metal temperature on the time step considered for the calculation is used to calculate the F_{en} in the detailed calculation. This assumption is used for the calculation of the quantity T^*.
- **Constant material properties are used for the FEA *code aster* calculations on pipes:** The material properties, i.e. Young's modulus (E), coefficient of thermal expansion (α), thermal conductivity (k), density (ρ), specific heat (c_p), were all assumed to remain constant. The Young's modulus and the coefficient of thermal expansion values were selected so as to maximize their product which maximizes the stresses and is a typical approach in stress report calculations. The thermal conductivity, the density and the specific heat were taken so as to maximize the component time constant (proportional to $\rho c_p/k$); if the time constant is maximized, the strain rates will be decreased so the F_{en} value will be increased. The values used are E = 197,000 MPa, $\alpha = 16.4\text{x}10^{-6}/°\text{C}$, k = 14.7 $\text{mW-mm}^{-1}\text{-°K}^{-1}$, and ρc_p = 4.386 $\text{mJ-mm}^{-3}\text{-°C}^{-1}$.
- **Same material used for all the screening process:** The material properties for the F_{en} correlations for the austenitic stainless steels in Group 3.1, as found in Appendix Z of the RCC-M Code, were used. This encompasses the coefficient of thermal expansion (α), the thermal conductivity (k), the density (ρ) and the specific heat (c_p).

- **K_e not considered for the F_{en} evaluation:** The elastic-plastic strain concentration factor, K_e, is not included in the F_{en} calculations since this factor amplifies the stress ranges and therefore increases strain rates. It is therefore conservative, as mentioned in Code Case N-884 [42], not to include it.

5.6 Method Description

The EDF EAF screening method was based on the original EPRI EAF screening report [4]. Any differences with the EPRI report will be highlighted here.

The method consists in four steps:

1. **Gathering input data:** The input data includes piping geometries, transients, etc. The data will be the starting point to perform the screening. The screening will be performed on the primary loop wetted surfaces in austenitic stainless steel components.
2. **Dividing the system into Thermal Zones:** The Thermal Zones [4] can be defined as those portions of the same component or system that undergo the same set of transients. For instance, the Pressurizer Safety and Relief Line and the Safe End on the Pressurizer Nozzle connected to that line can be considered as one thermal zone since they both are made from austenitic stainless steel and they both experience the same thermal transients.
3. **Updating the usage factor with environmental effects for each location within the Thermal Zone:** The usage factor that is used as a starting point can be found in previous stress reports. This calculation consists simply in updating this usage factor by evaluating the F_{en} and multiplying the partial usage factors by the F_{en}.
4. **Identifying the Sentinel Locations:** Sentinel Locations [4] are determined using the updated values of the environmental usage factors. Each location is ranked according to the value of its updated usage factor and, for each Thermal Zone, the location with the highest value is considered to be the most limiting. Detailed evaluations will be carried out on these locations only.

5.6.1 Input Data

The first step in the screening is to gather the necessary input data. This includes the following:

- Pipe layout and connectivity
- Geometry of the location to be evaluated, and more particularly the wall thickness
- Cumulative usage factors and partial usage factors from previous stress reports
- Transient pairs associated to partial usage factors
- Transient descriptions, including:
  - Temperature variation (identify in particular hot and cold shocks)
  - Heat transfer coefficient
  - Transient duration

The documents that will be used to gather the data will be previous stress reports, general layout drawings and transient description documents.

5.6.2 Thermal Zones

Based on the input information, and more particularly the pipe layout and transient description documents, the various main components of the primary loop can be divided into Thermal Zones.

A Thermal Zone is a zone within a system or component where all areas experience the same transient. This is of major importance in the screening process since Sentinel Locations will be defined for each of these Thermal Zones.

The Sentinel Locations are bounding in terms of environmental and overall fatigue damage compared to all other locations in the Thermal Zone.

For instance, based on review of the layout drawings as well as previous stress reports, the pressurizer safety and relief line including the safe end of the nozzle connected to the Pressurizer are suitable for consideration as a single Thermal Zone. The documents that were used were previous stress reports on optimization of fatigue calculations because they contain the analysis along the entire piping line. The same transients were applied to the piping and the safe end.

In addition, the piping line is made out of two types of austenitic stainless steels: Z2 CN 18-10 (the major part of the line) and Z2 CND 18-12 N2 (the blow-down nozzle). Although these two heats of material are slightly different, the RCC-M Code [33] classifies them within the same material group and sub-group (Table Z I 5.0 [33]), which means that they have identical Young's modulus, coefficient of thermal expansion and conductivity.

As a result, the choice was made to keep the entire piping line as one thermal zone despite this minor change in the materials.

5.6.3 Updating the Cumulative Usage Factor

The next step is then to update the usage factor by evaluating the impact of the PWR environment on the various Thermal Zones. Updating the usage factors is done in four steps:

1. **Within the Thermal Zones, identify whether or not the fatigue analysis was performed using the same method:** If not, additional research should be carried out to find stress reports that used the same method of fatigue analysis. For instance, the latest stress reports covering the Pressurizer Letdown line presented fatigue analyses for selected locations only. Additional research showed that initial stress reports, written at the time the first 900 MWe plants were commissioned, include a full analysis of the Letdown Line, using RCC-M B-3600.
2. **For each location, identify which transients contribute the most to the cumulative usage factor:** It can be assumed that the transient combination order remains unchanged with incorporation of the environmental effects.
3. **For each of these locations and for each transient combination, evaluate an F_{en} factor using a simplified methodology (detailed in Section 5.6.3.1):** EDF Report D305914013266 [39] presents a table with various levels of strain rates to be used in the F_{en} factor calculation. The method presented in Section 5.6.3.1 uses an analytical formula instead, depending on various quantities describing the transient applied.

4. **Multiply the partial usage factors by each of the calculated F_{en} values and calculate the cumulative FU_{en}:** This step is very similar to usual fatigue analyses, except for the product with each of the F_{en}.

5.6.3.1 F_{en} Factor Correlations

The aim of establishing these correlations is to provide a tool to easily calculate the F_{en} factors based only on the information for the applied transient shape. This allows for the use of quantities that are readily available in the stress report with minimum engineering judgment involved.

On NPP components, most transients occurring on piping that drive cumulative fatigue usage consist of hot or cold shocks. In order to establish the F_{en} correlations, these two types of transients were applied to five various piping thicknesses and diameters using FEA calculations. The dimensions that were used are summarized in Table 5-2. The F_{en} was evaluated for all these various profiles while making various controlling parameters vary, which allowed development of trends and links between variables. The F_{en} correlations were then derived on the basis of these calculations.

Two different correlations were determined for hot and cold shocks. A series of assumptions were made to obtain these correlations as discussed previously.

Table 5-2
Pipe profile dimensions

Abbreviation	Description	Thickness	Inner Radius
Asp2	Spray Line – Dimension 2	8.70 mm (0.34 in)	21.45 mm (0.84 in)
Dec1	Safety & Relief Line – Dimension 1	18.20 mm (0.72 in)	65.95 mm (2.60 in)
Dec2	Safety & Relief Line – Dimension 2	13.50 mm (0.53 in)	43.65 mm (1.72 in)
LEP	Primary Loop Piping	67.75 mm (2.67 in)	369.05 mm (14.53 in)
TP	Expansion (Surge) Line	35.20 mm (1.39 in)	106.90 mm (4.21 in)

The Fourier, Fo, and Biot, Bi, numbers are defined as:

$$Fo = \frac{k\Delta t}{\rho c_p {t_h}^2} \qquad \textbf{Eq. 5-4}$$

where:

Δt is the ramp duration of the shock (in seconds)

k is the thermal conductivity

ρ is the density

c_p is the specific heat

t_h is the pipe thickness

$$Bi = \frac{ht_h}{k} \qquad \textbf{Eq. 5-5}$$

where:

h is the heat transfer coefficient

k is the thermal conductivity

t_h is the pipe thickness

The time constant of a shock, T_p, can be calculated as follows:

$$T_p = \frac{\rho c_p {t_h}^2}{k} \qquad \textbf{Eq. 5-6}$$

where:

k is the thermal conductivity

ρ is the density

c_p is the specific heat

t_h is the pipe thickness

The Fourier number compares the conduction to the heat absorbed by the pipe wall. The Biot number compares the heat transferred by convection from the fluid to the wall and the heat transferred by conduction through the pipe wall.

The F_{en} correlation for hot thermal shocks is shown in Table 5-3.

The F_{en} correlation for cold thermal shocks is shown in Table 5-4.

Table 5-3
Stainless steel F_{en} estimation formulation for hot shocks

	F_{en} = MINIMUM(F_{en1} ; F_{en2})	
	F_{en1} = exp(-T*O* $\dot{\varepsilon}$*)	**F_{en2} = exp(-T*O*$\dot{\varepsilon}$*)**
*O**	*O** = 0.29 or 0.14, as appropriate	*O** = 0.29 or 0.14, as appropriate
*T**	T^* = 0 for T_{max}< 100°C (212°F) T^* = (T_{max} - 100) / 250 for 100°C (212°F) ≤ T ≤ 325°C (617°F) T^* = 0.9 for T_{max} > 325°C (617°F) T_{max} is the maximum temperature of the transient.	$T^* = min\left\{0.9; \frac{(T_{max} - 100)}{250} - C_1\frac{\Delta T}{250} + \frac{1}{2}\left[C_1\frac{\Delta T}{250}\right]^2\right\}$ C_1 is a coefficient defined as: $C_1 = \frac{1}{2[1 + Bi(Fo + 0.4)]}$ *Bi* is the Biot number *Fo* is the Fourier number (with all material properties taken the same as for the code aster calculation)
$\dot{\varepsilon}$	$\dot{\varepsilon} = \frac{100\alpha\Delta T A_1}{(1-\upsilon)T_p} + B_1$ α is the coefficient of thermal expansion (taken the same as for the code aster calculation) ΔT is the temperature difference during the hot shock (in °C) T_p is the time constant of the hot shock as defined above (in seconds) A_1 is a coefficient defined as: $A_1 = \frac{0.28}{Fo}(1 - e^{-2.5Fo})$ *Fo* is the Fourier number (with all material properties taken the same as for the code aster calculation) $B_1 = 2.5\times10^{-4}$	$\dot{\varepsilon}$ = 0.0004
$\dot{\varepsilon}^*$	$\dot{\varepsilon}^*$ =0 for $\dot{\varepsilon}$ > 7%/s $\dot{\varepsilon}^*$ = ln($\dot{\varepsilon}$/7) for 0.0004%/s ≤ $\dot{\varepsilon}$ ≤ 7%/s $\dot{\varepsilon}^*$ =ln(0.0004/7) for $\dot{\varepsilon}$ < 0.0004%/s	$\dot{\varepsilon}^*$= ln(0.0004/7)

Note: Differences between the draft of Revision 1 of NUREG/CR-6909 [35] and NUREG/CR-6909, Revision 1 [8] are highlighted in yellow. Refer to Section 5.9.1.

Table 5-4
Stainless steel F_{en} estimation formulation for cold shocks

	$F_{en} = \exp(-T^*O^*\dot{\varepsilon}^*)$
O^*	$O^* = 0.29$ or 0.14, as appropriate
T^*	$T^* = 0$ for $T < 100°C$ (212°F) $T^* = [(A_2T_I + (1 - A_2)(T_I - \Delta T) - 100) / 250$ for 100°C (212°F) ≤ T ≤ 325°C (617°F) $T^* = 0.9$ for $T > 325°C$ (617°F) ΔT is the temperature difference during the cold shock (in °C) T_I is the initial shock temperature (in °C) A_2 is the coefficient defined as: $$A_2 = max\left\{0.84 - 0.057ln(Bi); 1 - \frac{0.55 + 0.32Bi^{-1}}{Fo + 1.34}\right\}$$ Bi is the Biot number (with material properties taken the same as for the code aster calculation) Fo is the Fourier number
$\dot{\varepsilon}$	$$\dot{\varepsilon} = \frac{100\alpha\Delta TB_2}{(1-\upsilon)C_2T_p}$$ ΔT is the temperature difference during the cold shock (in °C) α is the coefficient of thermal expansion (taken the same as for the code aster calculation) T_p is the time constant of the cold shock as defined above (in seconds) B_2 is a coefficient defined as: $$B_2 = min\left\{1 - \frac{1.4}{\sqrt{2+Bi}}; \frac{1}{3Fo}\right\}$$ C_2 is a coefficient defined as: $$C_2 = min\left\{\frac{1.77}{Bi} + 0.74; \frac{0.07}{Bi} + Fo\right\}$$ Bi is the Biot number (with material properties taken the same as for the code aster calculation) Fo is the Fourier number
$\dot{\varepsilon}^*$	$\dot{\varepsilon}^* = 0$ for $\dot{\varepsilon} >$ 7%/s $\dot{\varepsilon}^* = \ln(\dot{\varepsilon}/7)$ for 0.0004%/s ≤ $\dot{\varepsilon}$ ≤ 7%/s $\dot{\varepsilon}^* = \ln(0.0004/7)$ for $\dot{\varepsilon} <$ 0.0004%/s

Note: Differences between the draft of Revision 1 of NUREG/CR-6909 [35] and NUREG/CR-6909, Revision 1 [8] are highlighted in yellow. Refer to Section 5.9.1.

5.6.3.2 Transient Combination

The combination of two transients is performed by weighing each of the F_{en} values obtained through the correlations derived in Section 5.6.3.1 with an estimate of the total strain variation, i.e., $\alpha\Delta T$. More specifically, the following equation is used:

$$F_{en-tot} = \frac{F_{en,1}\alpha_1\Delta T_1 + F_{en,2}\alpha_2\Delta T_2}{\alpha_1\Delta T_1 + \alpha_2\Delta T_2} \qquad \textbf{Eq. 5-7}$$

where:

$F_{en\text{-}tot}$ is the F_{en} value for the specific transient combination

$F_{en,1}$, α_1, ΔT_1 are the F_{en}, coefficient of thermal expansion and the temperature change for Transient 1, respectively

$F_{en,2}$, α_2, ΔT_2 are the F_{en}, coefficient of thermal expansion and the temperature change for Transient 2, respectively

5.6.4 Selection of Sentinel Locations

Once the cumulative usage factors are updated, a ranking within a Thermal Zone of all the locations according to these new updated values is established.

The final locations that will be retained for the detailed evaluation are the locations that rank the highest within a thermal zone (excluding locations already sensitive to fatigue damage that have $FU > 1$, per below). Depending on the Thermal Zone considered and depending on the analyses that were performed in previous stress reports, only the first location or a certain amount of top locations will be selected.

In the event that within a Thermal Zone, there is more than one location with a fatigue usage that exceeds the limit of 1.0, more than one Sentinel Location is picked, according to the following procedure:

- The location with the highest fatigue usage is identified, excluding all locations that are already sensitive to fatigue damage ($FU > 1$);
- The location with the largest change in fatigue usage value will also be in some cases identified as a Sentinel Location;
- Subsequent locations may be also included depending on engineering judgment; for instance, if two locations exhibit updated usage factors that are within 5%, it is advisable to identify those two locations as Sentinel Locations.

It should be highlighted that during future PSA calculations using the results of this screening, if the Sentinel Location selection appears not to be exhaustive enough, the ranking of all the locations within one Thermal Zone will enable the engineer to select additional Sentinel Locations accordingly.

Moreover, some analyses have already been limited to some locations only. In this case, there will be a need to use the original fatigue analyses to analyze the effect of the F_{en} on all of the locations of the thermal zone. This will have to be addressed on a system-to-system basis.

Detailed calculation efforts will be carried out at these locations only, since these locations are the bounding locations for the entire Thermal Zone.

5.6.5 Notches and Crack-Like Defect Analysis

In Annex ZD of the RCC-M Code, a method is described for analyzing notches and crack-like defects. There are no methods available that combine this methodology and the F_{en} factors. Therefore, for the purpose of this screening, the usage factors for locations close to notches and crack-like defects will be updated to account for these effects by extrapolating the effects of adjacent notches or crack-like defects.

5.7 Application of the Screening Process to 900 MWe NPP Components

The process that is detailed in the preceding sections is applied to the Pressurizer Safety and Relief Line in this section. This includes the zone from the Pressurizer Nozzle to the SEBIM® valves.

5.7.1 Gathering Input Data

The input data stems essentially from former stress reports and transient description documents.

In the case of this component, the latest fatigue analysis considers one zone only: the ¾” letdown nozzle. Therefore, the original stress reports were located to find a complete fatigue analysis of the entire piping system. This fatigue analysis was performed according to B-3600 of the RCC-M Code for all zones in the Pressurizer Safety and Relief Line.

5.7.2 Dividing the System into Thermal Zones

The Pressurizer Safety and Relief Line and the Safe End on the Pressurizer Nozzle connected to that line can be considered as one thermal zone since they are both made from austenitic stainless steel and they both undergo the same thermal transients. Therefore, there is only one thermal zone for this component. A sketch of the line geometry is shown in Figure 5-2.

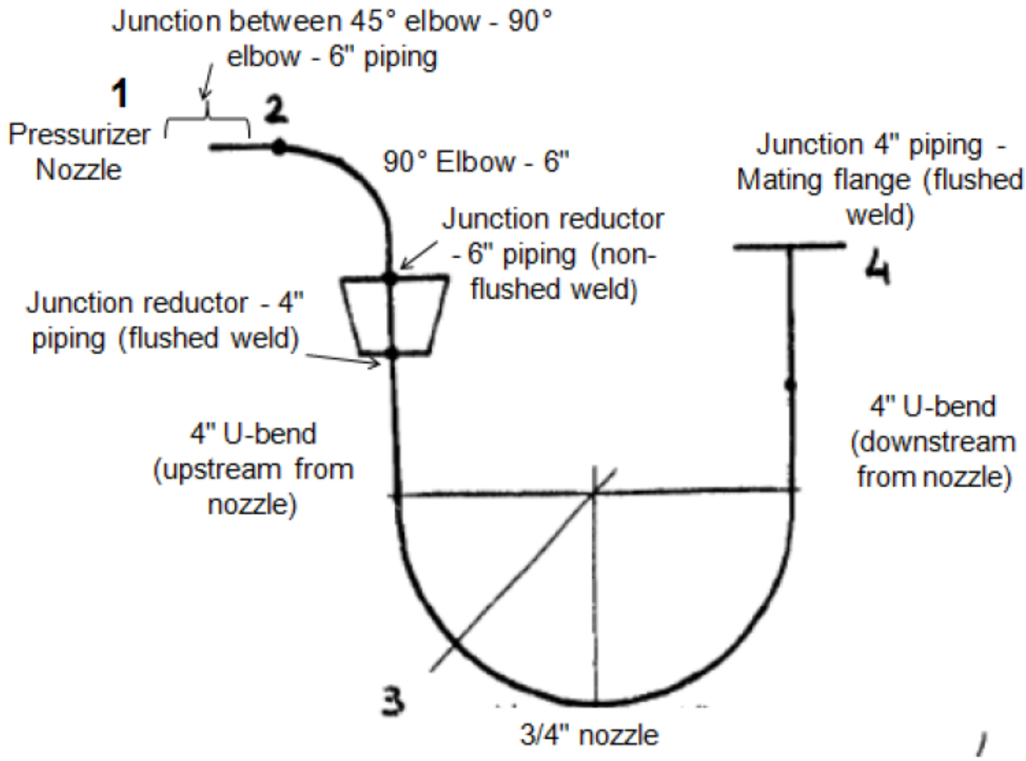


Figure 5-2
Sketch of the pressurizer safety and relief line indicating the nine locations analyzed for fatigue

5.7.3 Updating the Usage Factor with Environmental Effects for Each Location Within the Thermal Zone and Identifying Sentinel Locations

The strategy for updating the usage factor and identifying Sentinel Locations is defined on a system-to-system basis. In this case, the stress reports from the latest decennial inspection visit showed that the fatigue analysis could be optimized for the Fessenheim 1 Pressurizer Safety and Relief Line only. Moreover, the stress report for the Pressurizer Safety and Relief Line showed that the limiting location was the ¾" letdown nozzle. The fatigue analysis for the whole line can be found in the original stress report document, but already then the fatigue analysis had only been optimized for the ¾" letdown nozzle only.

For the screening, the original fatigue analysis of the whole line will be taken to see how the incorporation of the environmental multipliers might influence the usage factors and their order. Once this result is known, and if the ¾" letdown nozzle location is still the leading location, then analysis based on the latest stress report document will be led.

5.7.3.1 Update of Initial Fatigue Analysis Usage Factors

The update of the fatigue usage factors is performed using the input data gathered from the former fatigue analysis as well as the former associated transient description document. The objective here is to evaluate how the incorporation of the environmental multipliers may affect the order of the locations along the line. Table 5-5 and Figure 5-3 summarize the evaluation performed and the results.

The following comments apply based on examination of Table 5-5:

- The ¾” letdown nozzle remains the limiting fatigue location after incorporation of environmental effects. This location has an updated usage factor much higher than any of the other locations. This will be the Sentinel Location chosen for this thermal zone.
- No other Sentinel Location apart from the ¾” letdown nozzle will be selected. There are some significant increases in the usage factors, and some of the increases are even larger than the one for the ¾” letdown nozzle. Nevertheless, since the other usage factors remain below the value for the ¾” letdown nozzle, other locations will not be selected.

This completes the first step of the update of the usage factors. The Sentinel Location is identified, and the study confirmed the choice of the ¾” letdown nozzle as a representative limiting location. The next step will be to update the fatigue usages for the ¾” letdown nozzle analysis to assess whether the updated usage factors may exceed unity.

Table 5-5
Summary of usage factor updates for the pressurizer safety and relief line (1)

Location	Case No.	*FU* (2)	*FU* Rank	*FU* x F_{en}	*FU* x F_{en} Rank	Change	Equivalent F_{en}	Used as Sentinel Location?	RCC-M RPP-2 Design Curve
Pressurizer Nozzle (*Piquage pressuriseur*)	1	1.51	7	11.04	7	631%	7.31	N	1.17
Junction between 45° elbow- 90° elbow - 6" piping (*Jonction coude 45° - Coude 90° - Tuyau 6"*)	2	1.13	9	8.18	9	624%	7.24	N	0.93
90° Elbow - 6" (*Coude 90° - 6"*)	3	1.17	8	8.69	8	643%	7.43	N	0.90
Junction reducer – 6" piping (non-flushed weld) (*Jonction réduction tuyau 6" (soudure non arasée)*)	4	5.15	2	15.89	2	209%	3.09	N	3.29
Junction reducer – 4" piping (flushed weld) (*Jonction réduction tuyau 4" (soudure arasée)*)	5	3.55	3	13.02	4	267%	3.67	N	2.35
4" U-bend (upstream from nozzle) (*Partie courante cintre 4" (amont bossage)*)	6	2.57	5	12.04	5	369%	4.69	N	1.77
3/4" nozzle (*Bossage 3/4"*)	7	7.07	1	32.83	1	364%	4.64	Y	4.28
4" U-bend (downstream from nozzle) (*Partie courante cintre 4" (aval bossage)*)	8	2.40	6	11.18	6	366%	4.66	N	1.67
Junction 4" piping – Mating flange (flushed weld) (*Jonction tuyau 4" contre- bride (soudure arasée)*)	9	3.21	4	14.95	3	366%	4.66	N	2.15

Notes: (1) The correlations based on the draft of Revision 1 of NUREG/CR-6909 [35] were used for the assessments summarized in this table.

(2) The usage factors, *FU*, are all above 1.0 because no optimization was performed.

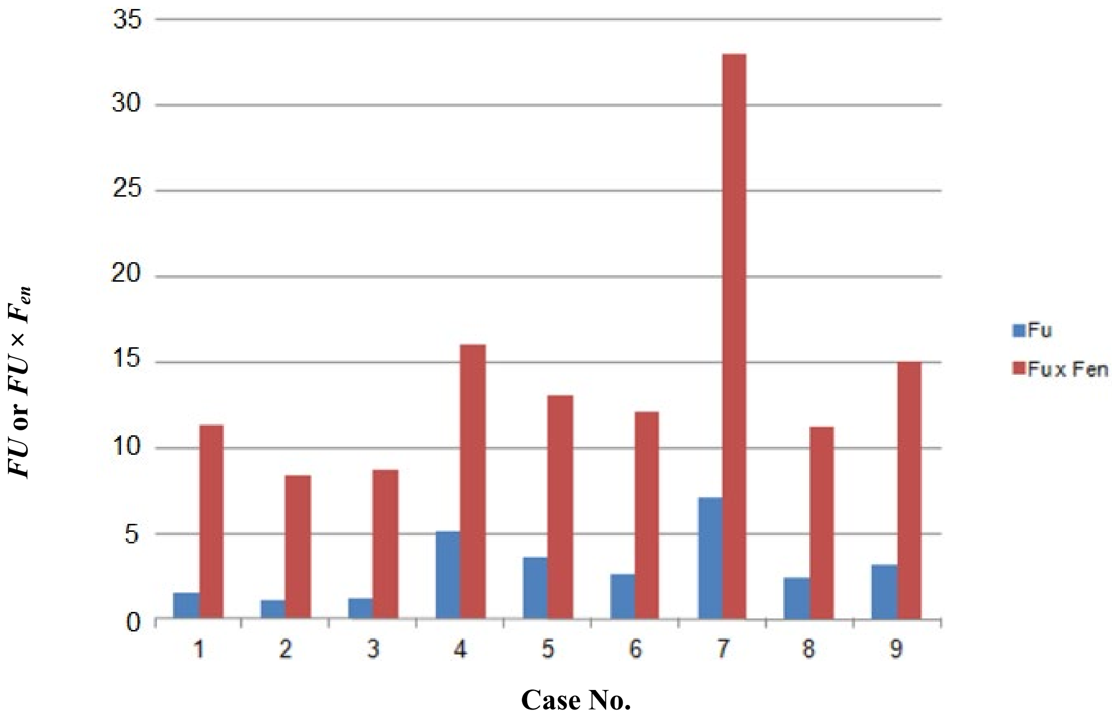


Figure 5-3
Comparison of usage factor without environmental factor (blue) and with environmental factor (red)

5.7.3.2 Update of Latest ¾" Letdown Nozzle Usage Factors

The starting point for this assessment is the updated fatigue analysis that revised the past analyses and focused on the ¾" letdown nozzle. This is assumed to be the most limiting location for all of the 900 MWe power plants in France.

The objective of this step is to evaluate the impact of the environmental multipliers and new fatigue curves on the latest fatigue calculation to conclude whether or not this location may become sensitive to fatigue damage ($FU > 1.0$).

The most limiting location in the analysis was the stress cut line (SCL) 6 located in the elbow of the nozzle region. A thickness of 15.24 mm was used based on the geometry defined in the updated fatigue analysis; this thickness corresponds to the maximum thickness between the main piping and the auxiliary piping. In the case of nozzles, the same approach was used. Table 5-6 gives an overview of the results.

Table 5-6
Summary of updated usage factors for the most limiting locations in the ¾" letdown nozzle (1)

Transient A	Transient B	No. of Cycles	*FU*	*RCC-M RPP-2 Design Curve*	FU_{en}	FU_{en} with $F_{en\text{-}integrated}$
8 (A1-A2)	16-SE (D1-D2)	10	0.120	0.080	0.480	0.116
8 (A1-A2)	16 (D1-D2)	50	0.280	0.209	1.232	0.303
8 (A1-A2)	18 (E1-E2)	20	0.100	0.076	0.448	0.110
8 (A1-A2)	20 (E3-E4)	25	0.110	0.086	0.508	0.124
17 (D1-D2)	22 (F1-F2)	60	0.120	0.111	0.536	0.161
19 (E1-E2)	22 (F1-F2)	20	0.030	0.029	0.141	0.042
21 (E3-E4)	22 (F1-F2)	25	0.030	0.031	0.147	0.044
15 (C1-C2)	22 (F1-F2)	65	0.060	0.065	0.194	0.093
15 (C1-C2)	23 (F1-F2)	170	0.030	0.045	0.134	0.064
TOTALS:			**0.870**	**0.731**	**3.820**	**1.058**

Note: (1) The correlations based on the draft of Revision 1 of NUREG/CR-6909 [35] were used for the assessments summarized in this table.

The following comments are made based on the results shown in Table 5-6:

- The impact of the F_{en} factor is significant. The former total usage factor of 0.870 is increased by an overall F_{en} of 3.820/0.731 = 5.22 to reach a value of 3.820 when the F_{en} factors are included in the analysis. When performing the calculation using $F_{en\text{-}integrated}$, the usage factor drops to 1.058. This location could then become sensitive to fatigue after incorporation of the environmental multipliers in a detailed EAF calculation.

The thermal conditions for the different transients listed in the first two columns of Table 5-6 are summarized in Table 5-7.

It should be noted that the transient description does not include the return to the initial temperature. This return is nevertheless very slow and probably generates no usage factor on the geometry which is why it was not included as part of the original transient description.

Table 5-7
Thermal conditions for the transients contributing to fatigue in the pressurizer safety and relief line ¾" letdown nozzle

Transient	$T_{initial}$, °C (°F)	T_{final}, °C (°F)
A1-A2	10 (50)	345 (653)
C1-C2	40 (104)	40 (104) (1)
D1-D2	40 (104)	350 (662)
E1-E2	40 (104)	340 (644)
E3-E3	40 (104)	340 (644)
F1-F2	30 (86)	230 (446)

Note: (1) No temperature change occurs at this location for this transient so the associated F_{en} factor is 1.0.

5.7.4 Summary of the Screening Process

The screening of the Pressurizer Safety and Relief Line has shown that:

1. One Sentinel Location can be selected (the ¾" letdown nozzle). The updated usage factor is much higher than any other location on the Pressurizer Safety and Relief Line. It is assumed that if the detailed fatigue analysis on the location can be optimized to a value under 1.0, then all other locations will also have usage factors below 1.0.
2. The impact of the F_{en} factor is significant. Previous usage factors are multiplied by values ranging from 3 to 7, as indicated by the Equivalent F_{en} values in Table 5-5.

5.8 Conclusion

In conclusion, this section provides the description of the EDF EAF screening methodology used to identify the primary loop locations most impacted by EAF (Sentinel Locations). Detailed FEM calculations for PSAs have only been carried out on these locations.

In the present report, the application of the methodology is illustrated on the case of the Pressurizer Safety and Relief Line, and one Sentinel Location is identified: the ¾" letdown nozzle.

It should be emphasized that although the F_{en} correlation expressions seem at first complicated to manipulate, they are overall easy to implement in practice. Moreover, once they are implemented in a spreadsheet, there is only a limited amount of data that needs to be updated to calculate the F_{en} factors and the updated usage factors, which are:

- The thickness of the location analyzed;
- The transient characteristics (heat transfer coefficient, rise time, and temperature difference); and
- The transient combinations and previous usage factors.

This information is accessible in previous stress reports and transient description documents.

Finally, it is also noted that EAF detailed calculations may encounter some shortcomings when it comes to providing a conclusion relative to some locations that are not Sentinel Locations. In the situation where the usage factors are updated and only the Sentinel Locations exceed the allowed value of 1.0, another valuable piece of information would be to identify how many of the other locations also have usage factors exceeding 1.0. In practice, this would be difficult to evaluate because this would require detailed EAF calculations for all of the locations.

It is for this reason why other locations may have to be selected as Sentinel Locations. This will be left to the judgment of the engineer who performs the screening. As an example, in the case of the Pressurizer Safety and Relief Line, it was finally decided to select only one Sentinel Location because the usage factors for all the other locations are much smaller.

5.9 Update of the Screening Method Based on NUREG/CR-6909, Revision 1

At the time the original EDF EAF screening report [41] was written, NUREG/CR-6909, Revision 1 was in draft form [35]. After NUREG/CR-6909, Revision 1 was finalized and published [8], some of the F_{en} formulations for stainless steel changed compared to the draft report, specifically with respect to the transformed strain rate term. Therefore, the EDF EAF screening method was updated for this report to reflect the use of NUREG/CR-6909, Revision 1 [8].

This section accomplishes the following:

1. Updates the previous EDF correlations to the new stainless steel formulations in NUREG/CR-6909, Revision 1 [8];
2. Performs a benchmarking analysis to demonstrate the suitability of the new correlations;
3. Develops a spreadsheet tool to utilize the new formulations with the following enhancements:
    a. Support for both SI and U.S. customary units.
    b. Support for both PWR and BWR environments.
    c. Support for inclusion of the K_e factor in F_{en} calculations.
4. Provides an example calculation to demonstrate the process and the spreadsheet tool.

5.9.1 Updated F_{en} Correlations

These F_{en} expressions for stainless steel were revised in NUREG/CR-6909, Revision 1 [8] and are shown in Appendix A.2.3. Those F_{en} expressions were used for the EDF EAF screening in this section. Based on a comparison of those F_{en} expressions to the expressions contained in the draft of Revision 1 of the NUREG [35], the only differences relate to the transformed strain rate and DO terms. The revised F_{en} formulations are presented in Table 5-3 for hot shocks and Table 5-4 in the formulations compared to those used in the original EDF formulations are highlighted in yellow in Table 5-3 and Table 5-4.

5.9.2 Benchmarking Analyses

To benchmark the new F_{en} formulations, a calculation matrix for each case was developed, as shown in Tables 6 and 9 in the original EDF EAF screening methodology report [41]. These include 595 cases of hot shock and 560 cases of cold shock for a range of geometry, heat transfer coefficients, initial and final temperature, and temperature ramp rate. Refer to Table 5-2 for a description and pipe profile dimensions for each component. SIA performed independent analyses for all cases in the calculation matrix using the ANSYS finite element software [46] to perform FEA of an axisymmetric cylinder. ANSYS macros were developed to enable parametric analysis of all analysis cases. This allowed for variation of the pertinent parameters for runs with a significant number of analysis cases.

The same material properties documented in the EDF EAF screening methodology report were used, except that they were input using U.S. customary units for consistency with existing practices and compatibility with existing analysis tools. In addition, a Poisson's ratio of 0.3 was assumed. Thermal analysis was performed using PLANE55 elements, and linear elastic stress analysis was performed using PLANE42 elements with extra shapes (conventional analysis). During the stress analysis, one end of the cylinder was fixed in the longitudinal (*Y*) direction, and the other end of the cylinder was coupled in the *Y*-direction to ensure that the plane section remained plane, while allowing for free axial and radial expansion of the pipe.

All stress components were first used to evaluate strain rate, in accordance with methodology outlined in EPRI Report 1025823 [47] and EPRI Report 1022876 [48]. This process consisted of computing the stress differences for each stress component at each time step relative to the previous time step, and then computing the principal stresses from the differences. The sign of the principal stress with the largest magnitude was used to determine whether the stress was increasingly tensile or increasingly compressive. Only the increasingly tensile portions were used to compute F_{en}. The strain rate was then computed as the stress intensity range divided by the elastic modulus and the time step. It was determined that by using the bi-axial stress difference between axial and radial stresses, *SY-SX*, the strain rate could be determined accurately, relative to using the stress intensity range. This is because for a linear elastic analysis, the three normal stresses are essentially principal stresses for a cylindrical piping component. The axial (*SY*) stress is slightly larger than the circumferential (*SZ*) stress, and the radial stress (*SX*) is nearly zero at the inside surface. Stress intensity can therefore be determined as the difference between axial and radial stresses.

Macros were programmed into the ANSYS input files to compute F_{en} automatically for each load case and save the results to an output file.

5.9.2.1 Hot Shock Correlation

Results of the hot shock analyses are shown in Figure 5-4. As indicated in the figure, there is excellent agreement between the F_{en} values calculated using FEA and the estimation formulas. A histogram of the errors is shown in Figure 5-5. The maximum difference between the estimation value and the value computed using FEA with the Modified Rate formulation (i.e., the "error") in the conservative direction is 0.45, and the maximum difference in the non-conservative direction is -0.74. The average error is -0.02.

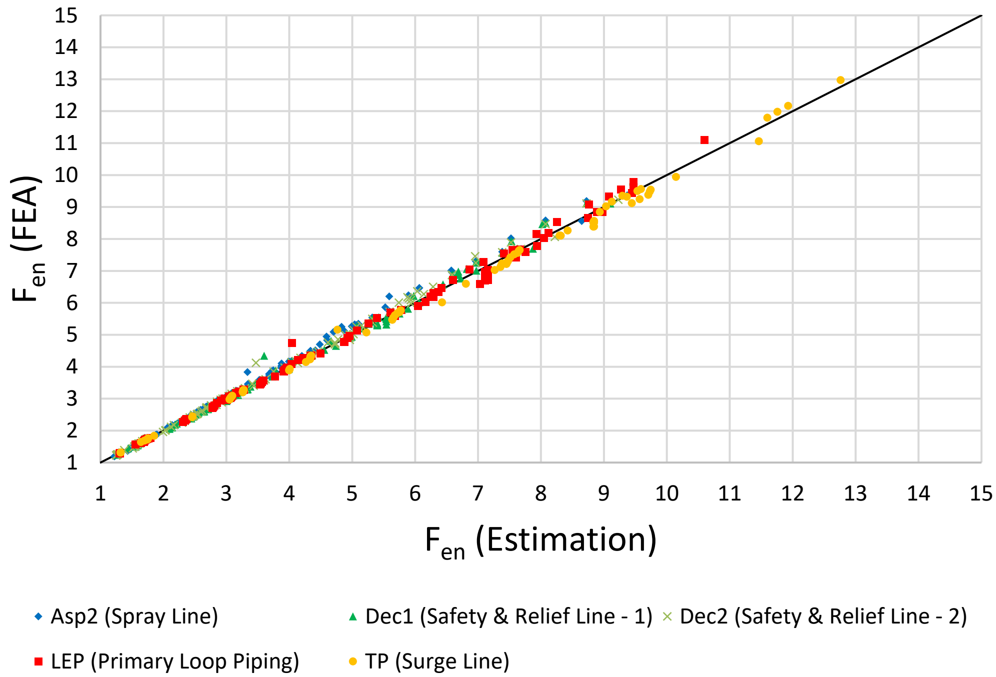


Figure 5-4
Benchmark calculations for hot shock F_{en} estimation

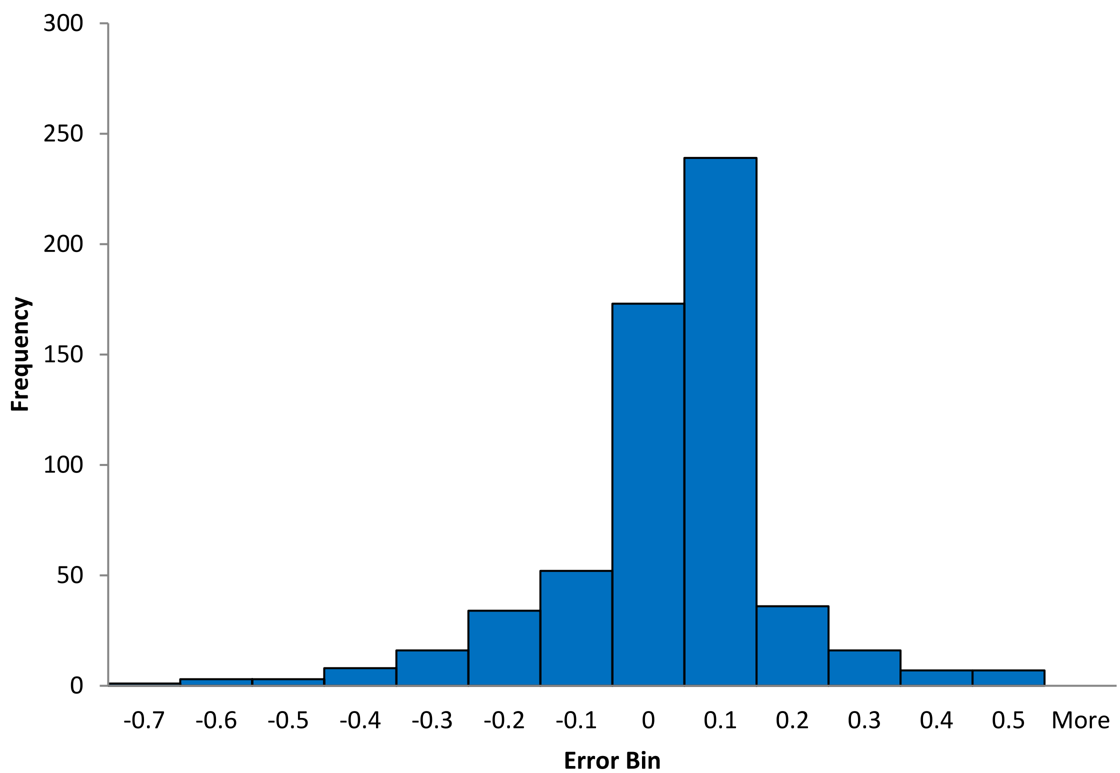


Figure 5-5
Histogram of F_{en} hot shock error

5.9.2.2 Cold Shock Correlation

Cold shock results are shown in Figure 5-6. While there is good agreement, the cold shock F_{en} estimations are not quite as accurate as those for the hot shock. This was recognized in the EDF EAF screening methodology report [41]. A histogram of the errors is shown in Figure 5-7. The maximum difference between the estimation value and the value computed using FEA with the Modified Rate formulation (i.e., the "error") in the conservative direction is 0.49, and the maximum difference in the non-conservative direction is -1.12. The average error is -0.22.

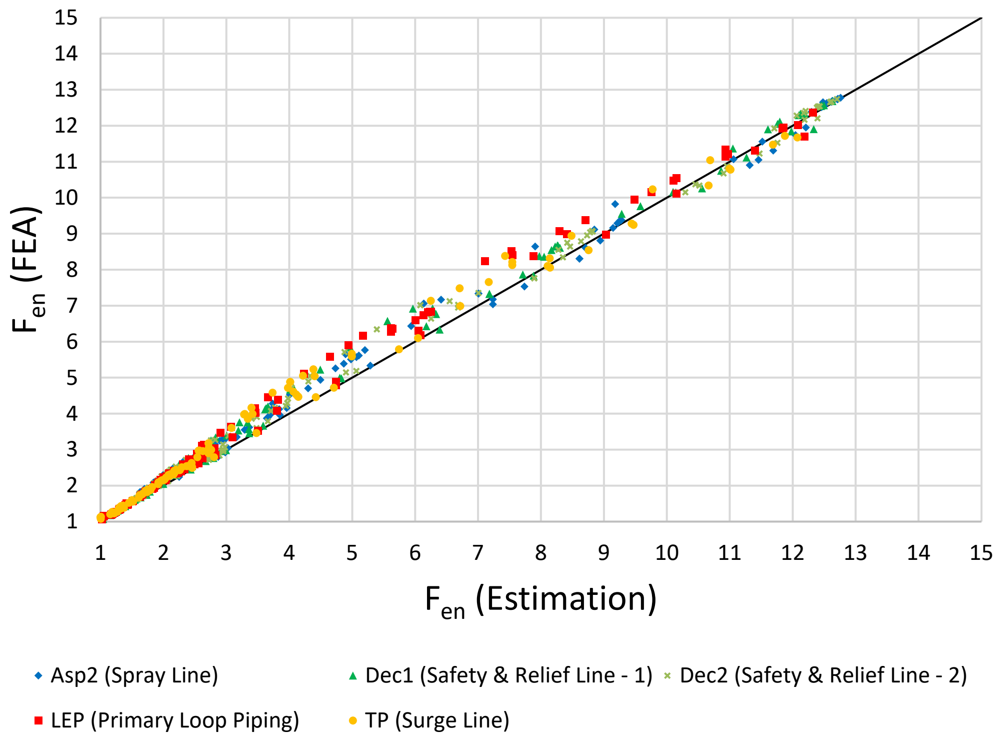


Figure 5-6
Benchmark calculations for cold shock F_{en} estimation

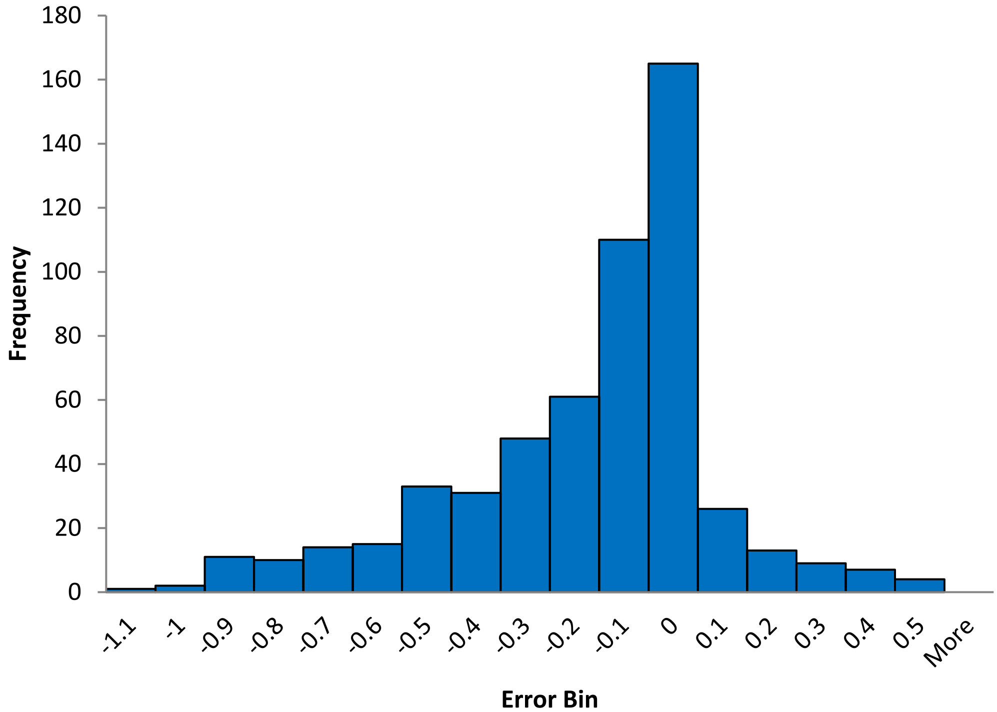


Figure 5-7
Histogram of F_{en} cold shock error

6
SPREADSHEET TOOL FOR ESTIMATING STAINLESS STEEL F_{EN} VALUES

Custom functions for use in an MS Excel® spreadsheet (*Fen_est.xlsm*) were developed for ease of use in screening applications. Both SI and U.S. customary units are supported in the spreadsheet and produce the same results given the same properly converted inputs. The spreadsheet tool was not developed under a quality assurance program and is intended for informational purposes only. The spreadsheet macro is attached as a file to this report.

The following function returns an F_{en} estimation value for austenitic stainless steel according to NUREG/CR-6909, Revision 1 [8].

U.S. Customary Units

MS Excel® function call: $Fen\_est(T_{init}, T_{final}, dtime, h, t, K_e, BWR)$

where the input parameters are:

T_{init}	=	initial temperature, °F
T_{final}	=	final temperature, °F
$dtime$	=	ramp time, sec.
h	=	heat transfer coefficient, BTU/hr-ft²-°F
t	=	thickness, in.
K_e	=	strain concentration factor (between 1.0 and 3.333)
BWR	=	0 when $O^* = 0.29$ / or 1 (or not 0) when $O^* = 0.14$

SI Units

MS Excel® function call: $Fen\_est(T_{init}, T_{final}, dtime, h, t, K_e, BWR)$

where the input parameters are:

T_{init}	=	initial temperature, °C
T_{final}	=	final temperature, °C
$dtime$	=	ramp time, sec.
h	=	heat transfer coefficient, mW/mm²-°C
t	=	thickness, mm
K_e	=	strain concentration factor (between 1.0 and 3.3333)
BWR	=	0 when $O^* = 0.29$ / or 1 (or not 0) when $O^* = 0.14$

6.1 Example Calculation

A prototypical example analysis was performed, using the EPRI EAF Expert Panel sample problem [47]. The sample problem contains geometric variations, material discontinuities, and a fatigue strength reduction factor of 2.0 applied at the location of interest. The loading contains temperature, pressure and bending moment variations, such that the resulting principal stresses at the surface rotate to some extent.

The geometry, which is similar to a PWR pressurizer surge nozzle (the nozzle at the bottom of the pressurizer), is shown in Figure 6-1. The finite element model is shown in Figure 6-2. Stresses at the inside surface of the stainless steel weld centerline ("SS Weld CL" in Figure 6-1 and "SCL1" in Figure 6-2) were used to compute F_{en} for a defined transient that includes both a cold and hot shock.

The transient definition is shown in Table 6-1 in U.S. customary units.

F_{en} values for both the hot and cold shock portions of the transient are compared. In addition, the effect of including the strain concentration factor, K_e (which is 3.333 for this example), in the strain rate was both included and neglected in separate calculations for comparison purposes.

Table 6-2 shows the results when with and without consideration of the K_e factor using U.S. customary units. Considering K_e, the overall F_{en} had a conservative error of 0.4, and without K_e the overall F_{en} had a conservative error of 1.9.

Note that other differences contribute to the error besides simplification, including the impacts on the strain history from the stress concentration factor, the pressure variation, and the moment variation. Even so, the comparison of results demonstrates that the estimations are reasonably accurate, while still being conservative overall. They are also a significant improvement to previous F_{en} estimation schemes used for the purpose of screening, which relied on analyst judgements to estimate the effective temperature and strain rate of the transients; such factors which are not always intuitive.

Table 6-1
Transient definition for prototypical sample problem in U.S. Customary Units

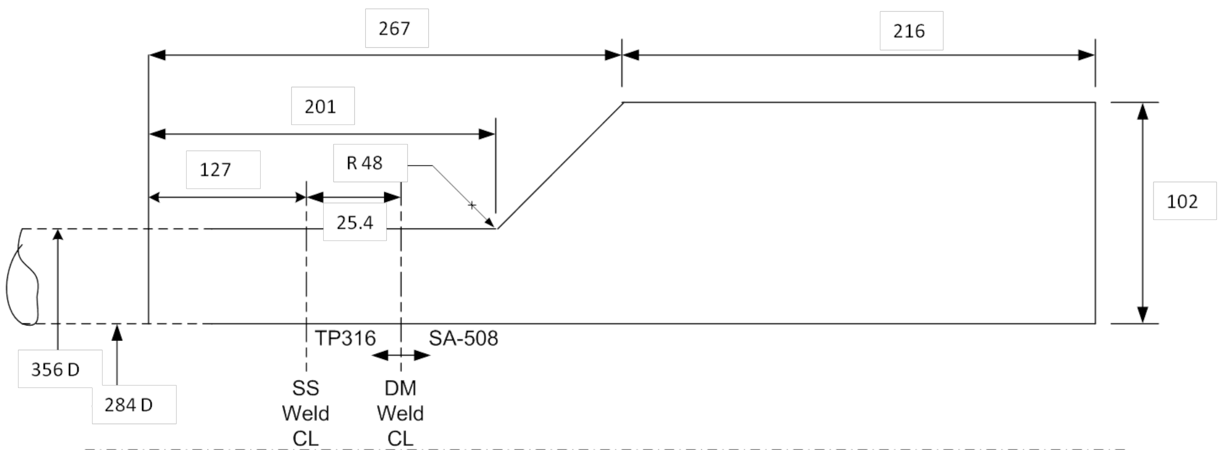
Transient 1 (Fast transient > 3,600 °F/hr) – 20 cycles				
Time (sec)	**Temperature (°F)**	**Heat Transfer Coefficient, h (BTU/s-in^2-°F)**	**Pressure (psi)**	**Resultant Moment[2] (in-kips)**
0[1]	600	0.0030	2,250	-3,000
5	600	0.0030	2,250	-3,000
205	100	0.0030	1,000	+1,000
1000	100	0.0030	1,000	+1,000
1200	600	0.0030	2,250	-3,000
3000	600	0.0030	2,250	-3,000

Notes:
1. A steady state solution for the conditions at the initial time step (t = 0 seconds) was solved.
2. A positive (+) sign indicates the direction of the applied resultant moment places the SCLs in tension, and a negative (-) sign indicates the direction of the applied resultant moment places the SCLs in compression.

Table 6-2
Sample problem results in U.S. Customary Units

Transient Portion	T_i (°F)	T_f (°F)	H (BTU/hr-ft²-°F)	ΔT (°F)	Detailed F_{en}	with K_e		without K_e	
						Fen_est()	Error	*Fen_est()*	Error
Cold Shock	600	100	1,555.20	500	2.6	2.9	0.3	3.5	0.9
Hot Shock	100	600	1,555.20	500	7.2	6.9	-0.3	9.4	2.1
Overall Values					**4.5**	**4.9**	**0.4**	**6.4**	**1.9**

Note: 1. The calculations for the results shown in this table are performed in spreadsheet *Fen_est.xlsm*, on the sheet "Sample Problem".



Note: Dimensions are in mm. To obtain inches, multiply by 25.4.

Figure 6-1
Geometry of prototypical sample problem

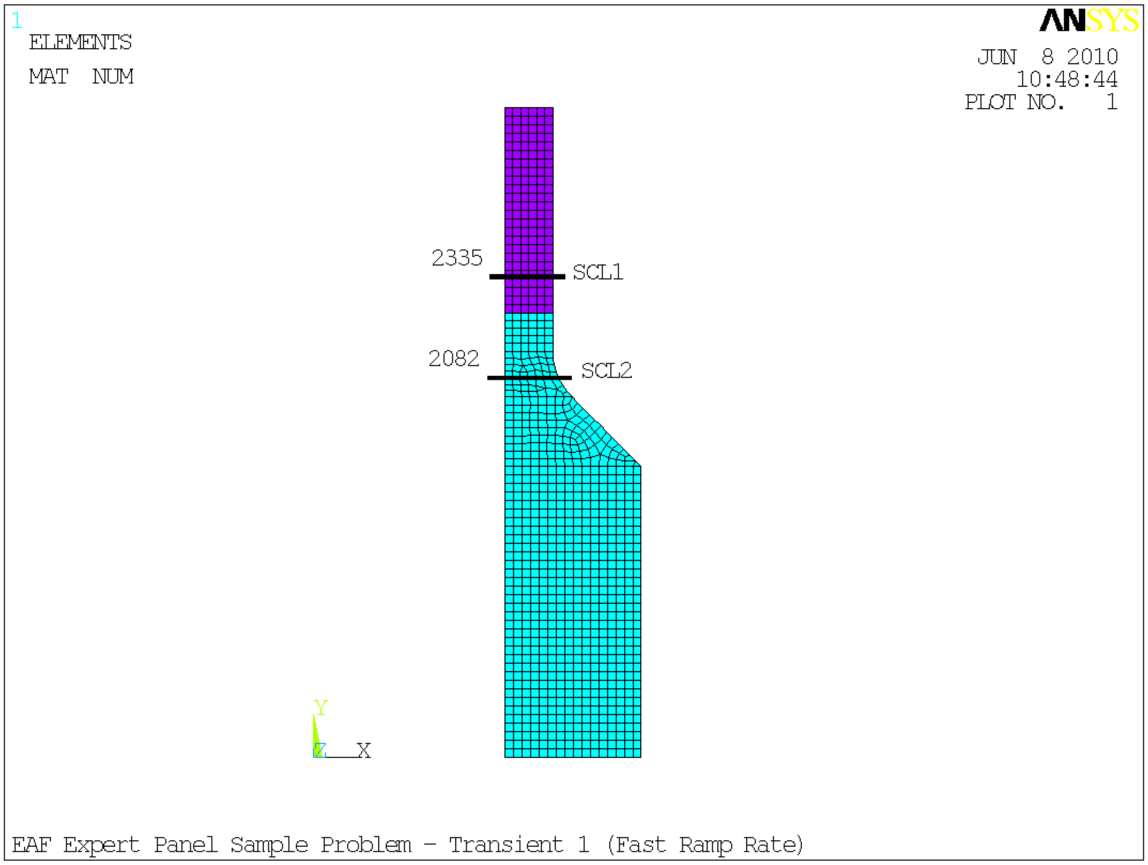


Figure 6-2
Finite element model for prototypical sample problem

7
SUMMARY

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 management.

7.1 The SIA EAF Screening Method

Section 3 provides the SIA EAF screening method and its technical basis. This method is intended for use to evaluate a plant to determine EAF limiting locations for fatigue management.

A primary reason for developing this process is to equip license renewal and subsequent 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. Optimize the level of effort involved.
3. For Class 1 piping without an explicit fatigue analysis, provide an analytical method using readily available design input from P&IDs, piping isometric drawings and piping stress reports.
4. Only basic stress or fatigue analysis is 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.

The process developed in this report provides guidance for the evaluation and relative ranking of estimated U_{en} values for locations in Class 1 components and systems without an explicit fatigue analysis (e.g., B31.1) that cannot be dispositioned as described in Section 2.8 and require EAF assessment to minimize the possibility of the need for a formal fatigue evaluation. The estimated values for potential 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 recommend-dation is to manage up to two locations with highest estimated U_{en} values exceeding 1.0.

Further analysis beyond the basic screening steps may be applied to reduce the number of locations. This may involve one or more of the following:

- Recalculating *CUF* using projected cycles for the extended operating period instead of design cycles.
- Recalculating F_{en} values based on the use of average temperature and/or actual plant transient data.
- Use of a later Code edition analysis (for example, later Code Editions of the Subsection NB Code reclassified the linear thermal gradient stress intensity from a secondary stress to a peak stress, thereby reducing K_e)

  Note*: The standard review plan for SLR (SRP-SLR) [30] requires a Code Reconciliation if a code edition other than the Construction code is used.*
- Application of ASME Code Case N-779 [11] alternative rules for calculation of K_e such that K_e is reduced.
- Calculation of *TF* and *GF* per ASME Code Case N-902 [12].
- Application of ASME Code Case N-904 [13] alternative rules for calculation of K_e such that K_e is reduced.

7.2 The WEC EAF Screening Method

Section 4 provides the WEC EAF screening method and its technical basis. NUREG-1801 and NUREG-2191 require plants applying for license renewal to consider the effects of the light water reactor environment on fatigue for the sample set of components defined in NUREG/CR-6260, plus any other RCPB component(s) that may be more limiting. The primary objective of the WEC EAF screening process is to provide an approach that will develop an optimized list of Sentinel Locations, including consideration of the locations previously identified in NUREG/CR-6260, for assessing the effects of light water reactor environments. The methodology presented is applicable to all RCPB components evaluated to the guidance in ASME Code, Section III, ANSI/ASME B31.7, or ANSI/ASME B31.1 for locations that require EAF assessment. Furthermore, a plant may utilize any of the industry documents on F_{en} evaluations discussed in Section 2.4 of this report with this approach.

The WEC EAF screening approach utilizes extensive design and analysis information and experience to consistently compare plant component locations without performing new detailed analyses. The method is supported by optimized definitions of regions exposed to similar transients, referred to as Transient Sections. The Transient Section definitions provide a framework to compare locations based on relative effects of the reactor water environment using F_{en} screening assignments. It further supports comparison of U and U_{en} values based on a ranking system for the complexity of the analysis relative to other locations. This Stress Basis Comparison ranking system provides a consistent characterization of the level of conservatism in the fatigue analysis, which can be used to appropriately compare the potential Sentinel Locations and minimize the final number of locations for explicit fatigue management. Furthermore, the screening U_{en} values for these locations are conservative based on the approach used to derive the screening F_{en} values. Therefore, detailed evaluations would be expected to result in lower U_{en} values. With the reduced set of potential Sentinel Locations obtained using the screening process, further refinement of the list of Sentinel Locations or ultimate evaluation of the final Sentinel Locations can be approached using methods such as the following:

- Application of projected cycles for the anticipated life of the plant.
- Ungrouping of thermal transients to provide less conservative stress profiles.
- Reduction in the number of zero stress states assumed in the analysis.
- Reduction in stress indices and stress concentration factors.
- Stress/temperature algorithm refinement such as using a finite difference solution of a one-dimensional heat transfer problem instead of closed form solutions.
- Application of improved approaches presented in later editions of the ASME Code, such as the approach to address stresses beyond the yield strength in elastic analyses.
- Application of ASME Code Cases to refine and reduce stress or fatigue conservatisms. Examples of such can be found in [11] through [13].

The methods noted above assist in further reductions of both the list of Sentinel Locations and corresponding U_{en} values. However, it should be noted that the SRP-SLR [30], requires a Code Reconciliation if a different ASME Code year is used for the analysis. The final list of Sentinel Locations determined with this approach can be used to support, or supplement, plant fatigue management and monitoring programs.

7.3 The EDF EAF Screening Method

Section 5 provides the EDF EAF screening method and its technical basis. The EDF method is focused exclusively on austenitic stainless steels as the other surfaces of ferritic steel components are either clad or the levels of *DO* in the water are too low for environmental effects to be significant. This methodology can be used to identify the primary loop locations most impacted by EAF (Sentinel Locations). Future detailed FEM calculations for PSAs will only be carried out on these locations.

The methodology was applied successfully to the Pressurizer Safety and Relief Line and identified one Sentinel Location: the ¾" letdown nozzle.

There is only a limited amount of data that needs to be updated to calculate the F_{en} factors and the updated usage factors, which are:

- The thickness of the location analyzed;
- The transient characteristics (heat transfer coefficient, rise time, and temperature difference); and
- The transient combinations and previous usage factors.

This information is accessible in previous stress reports and transient description documents.

Finally, it is also noted that EAF detailed calculations may encounter some shortcomings when it comes to providing a conclusion relative to some locations that are not Sentinel Locations. In the situation where the usage factors are updated and only the Sentinel Locations exceed the allowed value of 1.0, another valuable piece of information would be to identify how many of the other locations also have usage factors exceeding 1.0. In practice, this would be difficult to evaluate because this would require detailed EAF calculations for all of the locations.

It is for this reason why other locations may have to be selected as Sentinel Locations. This will be left to the judgment of the engineer who performs the screening. As an example, in the case of the Pressurizer Safety and Relief Line, it was finally decided to select only one Sentinel Location because the usage factors for all the other locations are much smaller. This is also consistent with the practice used by Framatome/AREVA in the previous PSAs.

The EDF EAF screening method was updated in this report to reflect the use of NUREG/CR-6909, Revision 1.

8
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A
SUMMARY OF F_{EN} FORMULATIONS

A.1 F_{en} Formulations for Ferritic Materials

A.1.1 NUREG/CR-6583

The following are the appropriate F_{en} relationships from NUREG/CR-6583 [A-1] for carbon and low alloy steels. These expressions are:

For carbon steel [A-1, p. 69]:

$$F_{en} = \exp\,(0.585 - 0.00124T' - 0.101S^{*}T^{*}O^{*}\dot{\varepsilon}^{*}) \qquad \textbf{Eq. A-1}$$

$F_{en} = \exp\,(0.554 - 0.101S^{*}T^{*}O^{*}\dot{\varepsilon}^{*})$ referenced to air at 25°C (77°F)

For low alloy steel [A-1, p. 69]:

$$F_{en} = \exp\,(0.929 - 0.00124T' - 0.101S^{*}T^{*}O^{*}\dot{\varepsilon}^{*}) \qquad \textbf{Eq. A-2}$$

$F_{en} = \exp\,(0.898 - 0.101S^{*}T^{*}O^{*}\dot{\varepsilon}^{*})$ referenced to air at 25°C (77°F)

Note that the above expressions are correct as summarized in EPRI/BWRVIP Memo. No. 2005-271 [A-2].

where:

F_{en}	=	fatigue life multiplication factor
T'	=	25°C (77°F) from NUREG/CR-6583, Section 6, F_{en} relative to room temperature air
S^{*}	=	transformed sulfur content = S for $0 < S \leq 0.015$ wt. %
	=	0.015 for $S > 0.015$ wt. %
S	=	weight percent sulfur of steel
T^{*}	=	transformed temperature = 0 for $T < 150$°C (302°F)
	=	(T - 150) for 150°C (302°F) $\leq T \leq$ 350°C (662°F)
T	=	metal temperature, °C
O^{*}	=	transformed DO content = 0 for $DO < 0.05$ parts per million (ppm)
	=	$\ln(DO/0.04)$ for 0.05 ppm $\leq DO \leq$ 0.5 ppm
	=	$\ln(12.5)$ for $DO > 0.5$ ppm
DO	=	DO content, ppm
$\dot{\varepsilon}^{*}$	=	0 for $\dot{\varepsilon} > 1$%/sec

	=	$\ln(\dot{\varepsilon})$ for $0.001 \le \dot{\varepsilon} \le 1$%/sec
	=	$\ln(0.001)$ for $\dot{\varepsilon} < 0.001$%/sec
$\dot{\varepsilon}$	=	strain rate, %/sec

A.1.2 NUREG/CR-6909, Revision 0

For carbon steel [A-3, p. A.1]:

$$F_{en} = \exp\,(0.632 - 0.101 S^* T^* O^* \dot{\varepsilon}^*) \qquad \textbf{Eq. A-3}$$

For low alloy steel [A-3, p. A.1]:

$$F_{en} = \exp\,(0.702 - 0.101 S^* T^* O^* \dot{\varepsilon}^*) \qquad \textbf{Eq. A-4}$$

Where S^*, T^*, O^*, and $\dot{\varepsilon}^*$ are the transformed sulfur content, transformed metal temperature, transformed DO, and transformed strain rate, respectively, which are defined as follows [A-3, pp. A.1 and A.2]:

F_{en}	=	environmental fatigue multiplier
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 (302°F)
	=	(T - 150) for 150°C (302°F) $\le T \le$ 350°C (662°F)
T	=	metal temperature, °C
O^*	=	0 for DO, $DO \le 0.04$ 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 $0.001 \le \dot{\varepsilon} \le 1$%/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:

$F_{en} = 1.0$ for strain amplitude, $\varepsilon_a \le 0.07$% or $S_{alt} \le (E_c)(0.07\%)/(100\%) = 21$ ksi (145 MPa)

A.1.3 NUREG/CR-6909, Revision 1

For carbon steel and low alloy steel [A-4, p. A-1]:

$$F_{en} = \exp\left((0.003 - 0.031\dot{\varepsilon}^*)S^*T^*O^*\right) \quad \textbf{Eq. A-5}$$

Where S^*, T^*, O^*, and $\dot{\varepsilon}^*$ are the transformed sulfur content, transformed metal temperature, transformed DO, and transformed strain rate, respectively, which are defined as follows [A-4, p. A-1]:

F_{en}	=	environmental fatigue multiplier
S^*	=	2.0 + 98S for $S \leq 0.015$ wt.%
	=	3.47 for S > 0.015 wt. %
T^*	=	0.395 for $T < 150$°C (302°F)
	=	(T - 75)/190 for 150°C (302°F) ≤ T ≤ 325°C (617°F)
T	=	metal temperature, °C
O^*	=	1.49 for $DO < 0.04$ ppm
	=	ln(DO/0.009) for 0.04 ppm ≤ DO ≤ 0.5 ppm
	=	4.02 for $DO > 0.5$ ppm
$\dot{\varepsilon}^*$	=	0 for strain rate, $\dot{\varepsilon} > 2..2$%/sec
	=	ln($\dot{\varepsilon}$/2.2) for 0.0004 ≤ $\dot{\varepsilon}$ ≤ 2.2%/sec
	=	ln(0.004/2.2) for $\dot{\varepsilon} < 0.0004$%/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:

F_{en} = 1.0 for strain amplitude, $\varepsilon_a \leq 0.07\%$ or $S_{alt} \leq (E_c)(0.07\%)/(100\%)$ = 21 ksi (145 MPa)

A.2 F_{en} Formulations for Austenitic Stainless Steel Materials

A.2.1 NUREG/CR-5704

For Types 304 and 316 stainless steels [A-5]:

$$F_{en} = \exp\left(0.935 - T^*\dot{\varepsilon}^*O^*\right) \quad \textbf{Eq. A-6}$$

where:

F_{en}	=	environmental fatigue multiplier
T	=	metal temperature, °C
T^*	=	0 for $T < 200$°C (392°F)
	=	1 for $T \geq 200$°C (392°F)

$\dot{\varepsilon}^*$ = 0 for strain rate, $\dot{\varepsilon} > 0.4\%/\text{sec}$

= $\ln(\dot{\varepsilon}/0.4)$ for $0.0004 \le \dot{\varepsilon} \le 0.4\%/\text{sec}$

= $\ln(0.0004/0.4)$ for $\dot{\varepsilon} < 0.0004\%/\text{sec}$

O^* = 0.260 for DO, $DO < 0.05$ ppm

= 0.172 for $DO \ge 0.05$ ppm

A.2.2 NUREG/CR-6909, Revision 0

For wrought and cast austenitic stainless steels [A-3, p. A.2][2]:

$$F_{en} = \exp\,(0.734 - T^*\dot{\varepsilon}^*\mathrm{O}^*) \qquad \textbf{Eq. A-7}$$

where:

F_{en} = environmental fatigue multiplier

T^* = 0 for $T < 150°\text{C}$ (302°F)

= $(T - 150)/175$ for 150°C (302°F) $\le T <$ 325°C 617°F)

= 1 for $T \ge 325°\text{C}$ (617°F)

T = metal temperature, °C

$\dot{\varepsilon}^*$ = 0 for strain rate, $\dot{\varepsilon} > 0.4\%/\text{sec}$

= $\ln(\dot{\varepsilon}/0.4)$ for $0.0004 \le \dot{\varepsilon} \le 0.4\%/\text{sec}$

= $\ln(0.0004/0.4)$ for $\dot{\varepsilon} < 0.0004\%/\text{sec}$

O^* = 0.281 for all DO 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:

F_{en} = 1.0 for strain amplitude, $\varepsilon_a \le 0.10\%$ or $S_{alt} \le (E_c)(0.10\%)/(100\%)$ = 28.3 ksi (195 MPa)

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[2] For symbol consistency throughout this document, the T′, O′, and $\dot{\varepsilon}'$ symbols used in NUREG/CR-6909 were changed to T*, O*, and $\dot{\varepsilon}^*$.

A.2.3 NUREG/CR-6909, Revision 1

For wrought and cast austenitic stainless steels [A-4, p. A-2]:

$$F_{en} = \exp\left(-T^* \dot{\varepsilon}^* O^*\right) \qquad \textbf{Eq. A-8}$$

where:

F_{en} = environmental fatigue multiplier

T^* = 0 for $T < 100°C$ (212°F)

= $(T - 100)/250$ for 100°C (212°F) $\leq T <$ 325°C (617°F)

T = metal temperature, °C

$\dot{\varepsilon}^*$ = 0 for strain rate, $\dot{\varepsilon} > 7\%$/sec

= $\ln(\dot{\varepsilon}/7)$ for $0.0004 \leq \dot{\varepsilon} \leq 7\%$/sec

= $\ln(0.0004/7)$ for $\dot{\varepsilon} < 0.0004\%$/sec

O^* = 0.29 for $DO < 0.1$ ppm

= 0.29 for $DO > 0.1$ ppm (sensitized high-carbon stainless steels)

= 0.14 for $DO > 0.1$ ppm (all other stainless steels)

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:

F_{en} = 1.0 for strain amplitude, $\varepsilon_a \leq 0.10\%$ or $S_{alt} \leq (E_c)(0.10\%)/(100\%)$ = 28.3 ksi (195 MPa)

A.3 F_{en} Formulations for Nickel Alloy Materials

A.3.1 NUREG/CR-6909, Revision 0

Application of NUREG/CR-6909, Revision 0 [A-3] is required for nickel alloy materials per the GALL report [A-6] and may also be used per Regulatory Guide 1.207, Revision 1 [A-7] and the GALL-SLR [A-8].

For nickel alloys [A-3, p. A.2][3]:

$$F_{en} = \exp\left(-T^* \dot{\varepsilon}^* O^*\right) \qquad \textbf{Eq. A-9}$$

where:

F_{en} = environmental fatigue multiplier

T^* = $T/325$ for $T <$ 325°C (617°F)

= 1 for $T \geq$ 325°C (617°F)

[3] For symbol consistency throughout this document, the T′, O′, and $\dot{\varepsilon}'$ symbols used in NUREG/CR-6909 were changed to T*, O*, and $\dot{\varepsilon}^*$.

T = metal 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 BWR NWC water

= 0.16 for PWR or BWR HWC water

For nickel 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:

F_{en} = 1.0 for strain amplitude, $\varepsilon_a \leq 0.10$% or $S_{alt} \leq (E_c)(0.10\%)/(100\%) = 28.3$ ksi (195 MPa)

A.3.2 NUREG/CR-6909, Revision 1

Application of NUREG/CR-6909, Revision 1 [A-4] is applicable to nickel alloy materials per the GALL report [A-6] and also per Regulatory Guide 1.207, Revision 1 [A-7] and the GALL-SLR [A-8].

For nickel alloys [A-4, pp. A-2 and A-3]:

$$F_{en} = \exp\left(-T^* \dot{\varepsilon}^* O^*\right) \quad \textbf{Eq. A-10}$$

where:

F_{en} = environmental fatigue multiplier

T^* = 0 for $T < 50$°C (122°F)

= $(T\text{-}50)/275$ for 50°C (122°F) $\leq T \leq$ 325°C (617°F)

T = metal 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^* = 0.06 for NWC BWR water

= 0.14 for PWR or HWC BWR water

For nickel 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:

F_{en} = 1.0 for strain amplitude, $\varepsilon_a \leq 0.10$% or $S_{alt} \leq (E_c)(0.10\%)/(100\%) = 28.3$ ksi (195 MPa)

A.4 Dissolved Oxygen (DO)

The *DO* value is determined by the evaluator. For PWRs, the value has typically been established as below 40 parts per billion (ppb) for plant conditions in which EAF is active (temperature above 150°C or 302°F). For BWRs, this value is determined based on several factors, such as use of HWC or not, and others. This value is typically obtained from individuals responsible for maintaining and monitoring the plant Water Chemistry Program.

A.5 Strain Rate ($\dot{\varepsilon}$)

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

A.6 References

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A-6. NUREG-1801, Revision 2, *Generic Aging Lessons Learned (GALL) Report*, U. S. Nuclear Regulatory Commission, Washington, DC, December 2010.

A-7. Regulatory Guide 1.207, Revision 1, *Guidelines for Evaluating the Effects of Light-Water Reactor Water Environments in Fatigue Analyses of Metal Components*, U.S. Nuclear Regulatory Commission, Washington, DC, June 2018.

A-8. NUREG-2191, *Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report*, U.S. Nuclear Regulatory Commission, Washington, DC, July 2017.

B

RULES FOR EVALUATION OF CLASS 1 PIPING

This section is extracted in part from Section III of the ASME Code [B-1]. The ASME Code rules for evaluation of Class 1 piping components are contained in Subarticle NB-3600 of Section III, and they have many similarities to the “design by analysis” rules of Subarticle NB-3200. Subsubarticle 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 Subarticle 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 Subarticle NB-3200 and Subsubarticle NB-3650 is that the latter takes a “design by formula” approach.

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

$$S_n = C_1 \frac{P_o D_o}{2t} + C_2 \frac{D_o}{2I} M_i + C_3 E_{ab} |\alpha_a T_a - \alpha_b T_b| \leq 3s_m \qquad \textbf{Eq. B-1}$$

(Section III, NB-3650, Eq. (10) [B-1][4])

where:

C_1, C_2, C_3	=	secondary stress indices for the specific component under investigation
D_o	=	pipe outside diameter, 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)

[4] Equation (B-1) represents the form of Equation (10) currently in the ASME Code. In earlier editions of the ASME Code, Equation (10) had the following form:

$$S_n = C_1 \frac{P_o D_o}{2t} + C_2 \frac{D_o}{2I} M_i + C_3 E_{ab} |\alpha_a T_a - \alpha_b T_b| + \frac{E\alpha \Delta T_1}{2(1-\upsilon)} \leq 3s_m$$

The highlighted term in the above equation was removed from Equation (10), thereby reclassifying the linear thermal gradient stress intensity range from a secondary stress to a peak stress.

α_a, α_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):

$$S_p = K_1 C_1 \frac{P_o D_o}{2t} + K_2 C_2 \frac{D_o}{2I} M_i + K_3 C_3 E_{ab} |\alpha_a T_a - \alpha_b T_b|$$

$$+ \frac{1}{2(1-\upsilon)} K_3 E\alpha |\Delta T_1| + \frac{1}{1-\upsilon} E\alpha |\Delta T_2| \qquad \textbf{Eq. B-2}$$

(Section III, NB-3650, Eq. (11) [B-1])

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 (α), 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_o, 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 ΔT_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.

i. Equation (12) shall be met:

$$S_e = C_2 \frac{D_o}{2I} M_i^* \leq 3s_m \qquad \textbf{Eq. B-3}$$

(Section III, NB-3650, Eq. (12) [B-1])

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)

ii. 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):

$$S_n = C_1 \frac{P_o D_o}{2t} + C_2 \frac{D_o}{2I} M_i + C_3{}' E_{ab} |\alpha_a T_a - \alpha_b T_b| \leq 3s_m \qquad \textbf{Eq. B-4}$$

(Section III, NB-3650, Eq. (13) [B-1])

where:

$C_3{}'$ = stress index

iii. If these conditions are met, the value of S_{alt} shall be calculated by Equation (14):

$$S_{alt} = K_e \, {}^{S_p}/_{2} \qquad \textbf{Eq. B-5}$$

(Section III, NB-3650, Eq. (14) [B-1])

where:

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

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

K_e = 1.0 for $S_n \leq 3S_m$

= $1.0 + [\ (1 - n) / \{n\ (m - 1)\}\ ]\ (S_n/3S_m - 1)$
(for $3S_m < S_n < 3mS_m$)

= $1/n$ for $S_n \geq 3mS_m$

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.

B.1 References

B-1. ASME Boiler and Pressure Vessel Code, Section III, *Rules for Construction of Nuclear Power Plant Components*, ASME, New York, NY, 2017 Edition.

C
TRANSLATED TABLE OF CONTENTS

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環境輔助疲勞篩選方法（修訂版 1）

3002018262

最終報告，2020 年 11 月

EPRI 專案經理
G. Stevens

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可交付產品編號：3002018262

產品類型：技術報告

產品標題：Environmentally Assisted Fatigue Screening Methods (Revision 1)

主要受眾：從事元件疲勞評估的國際公用事業人員

次要受眾：從事長期運營應用的國際公用事業人員

重要研究問題

有哪些環境輔助疲勞(EAF)篩選過程可用於滿足監管指南，確定反應堆冷卻劑壓力邊界內特定核電站限制部件的位置？

研究概述

大多數監管機構要求核電站所有者證明長期核電站運營可接受的環境輔助疲勞累積使用係數(CUF_{en})值，限制反應堆冷卻劑壓力邊界部件。因此，所有申請 60 年或 80 年延長運營許可證的美國核電站必須按照美國核管理委員會(NRC)通用老化經驗回饋報告(GALL)或通用老化經驗回饋報告後續許可證延期(GALL-SLR)指導方針處理環境輔助疲勞。很多非美國核電站也必須解決延長核電站運營的環境輔助疲勞問題，這些核電站遵循通用老化經驗回饋報告（GALL）或類似的指導方針。

2012 年 8 月，EPRI 在報告 1024995 *環境輔助疲勞篩選：確認環境輔助疲勞極限位置的過程和技術依據*中公佈了環境輔助疲勞篩選過程。EPRI 報告中的環境輔助疲勞篩選過程旨在描述符合通用老化經驗回饋報告指導方針的環境輔助疲勞篩選方法技術依據，確認「如果核電站特定部件位置可能比 NUREG/CR–6260 中更具限制性時，反應堆冷卻劑壓力邊界中的其他核電站特定部件位置。」本報告更新並取代了 1024995，並提供了三種環境輔助疲勞篩選方法，可用於篩選環境輔助疲勞的關鍵核電站部件，確定滿足通用老化經驗回饋報告和其他監管指南的限制核電站位置。

重要發現

- 本報告採用了從 EPRI 2012 年環境輔助疲勞篩選報告的先前應用中吸取的經驗回饋，在本報告中確定為 SI 公司(SIA)環境輔助疲勞篩選法。包括在篩選 1 級管道部件位置時減少不必要的保守性，還包括考慮美國核管理委員會先前對附加資訊的要求。
- 本報告更新了 SIA 相關環境輔助疲勞篩選方法，提出最近的後續許可證延期相關指導方針導。
- 本報告記錄了世界各地核電站開發和使用的另外兩種可用的環境輔助疲勞篩選方法。方法包括西屋電氣公司(WEC)環境輔助疲勞篩選方法和法國電力公司(EDF)環境輔助疲勞篩選方法。

為何重要

本報告中涵蓋的環境輔助疲勞篩選方法將幫助核電站所有者展示其核電站中位置的知識，用作需要環境輔助疲勞管理，滿足長期操作環境輔助疲勞指導方針的位置。環境輔助疲勞篩選方法提供了選擇這些位置的基本原理。核電站所有者可以在滿足法規要求的同時，避免正式疲勞分析的必要性，減少需要環境輔助疲勞評估的元件位置的總數，實現成本最小化，確定限制其核電站的環境輔助疲勞位置。每種環境輔助疲勞篩選方法均提供一種統一的方法，用於確定在整個核電站延長運營期內需要進行環境輔助疲勞評估和管理的限制位置。

如何應用結果

在本報告中描述的三種環境輔助疲勞篩選方法中，任何一種均可以用於確定核電站實現長期運營需要進行環境輔助疲勞評估的 1 類位置。

學習和參與機會

- 美國機械工程師協會等國際規範和標準委員會可能會發現，該報告對疲勞計算方法的可能改進有幫助。
- 材料可靠性計畫(MRP)會議。
- 沸水反應堆容器和堆內構件專案（BWRVIP）會議。

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計畫：核電，P41；沸水反應堆容器和堆內構件（BWRVIP），P41.01.03；壓水反應堆材料可靠性（MRP），P41.01.04

執行類別：參考文獻

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目錄

环境辅助疲劳筛选方法（修订版 1）

3002018262

最终报告，2020 年 11 月

EPRI 项目经理
G. Stevens

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主要受众：从事组件疲劳评估的国际公用事业人员

次要受众：从事长期运营应用的国际公用事业人员

主要研究问题

有哪些环境辅助疲劳(EAF)筛选过程可用于满足监管指南，确定反应堆冷却剂压力边界内特定核电站限制部件的位置？

研究概述

大多数监管机构要求核电站所有者证明长期核电站运营可接受的环境辅助疲劳累积使用系数(CUF_{en})值，限制反应堆冷却剂压力边界部件。因此，所有申请 60 年或 80 年延长运营许可证的美国核电站必须按照美国核管理委员会(NRC)通用老化经验反馈报告(GALL)或通用老化经验反馈报告后续许可证延期(GALL-SLR)指导方针处理环境辅助疲劳。很多非美国核电站也必须解决延长核电站运营的环境辅助疲劳问题，这些核电站遵循通用老化经验反馈报告（GALL）或类似的指导方针。

2012 年 8 月，EPRI 在报告 1024995 *环境辅助疲劳筛选：确认环境辅助疲劳极限位置的过程和技术依据*中公布了环境辅助疲劳筛选过程。EPRI 报告中的环境辅助疲劳筛选过程旨在描述符合通用老化经验反馈报告指导方针的环境辅助疲劳筛选方法技术依据，确认"如果核电站特定部件位置可能比 NUREG/CR–6260 中更具限制性时，反应堆冷却剂压力边界中的其他核电站特定部件位置。"本报告更新并取代了 1024995，并提供了三种环境辅助疲劳筛选方法，可用于筛选环境辅助疲劳的关键核电站部件，确定满足通用老化经验反馈报告和其他监管指南的限制核电站位置。

主要发现

- 本报告采用了从 EPRI 2012 年环境辅助疲劳筛选报告的先前应用中吸取的经验反馈，在本报告中确定为 SI 公司（SIA）环境辅助疲劳筛选法。包括在筛选 1 级管道部件位置时减少不必要的保守性，还包括考虑美国核管理委员会先前对附加信息的要求。
- 本报告更新了 SIA 环境辅助疲劳筛选方法，提出最近的后续许可证延期相关指导方针导。
- 本报告记录了世界各地核电站开发和使用的另外两种可用的环境辅助疲劳筛选方法。方法包括西屋电气公司(WEC)环境辅助疲劳筛选方法和法国电力公司(EDF)环境辅助疲劳筛选方法。

为什么重要?

本报告中涵盖的环境辅助疲劳筛选方法将帮助核电站所有者展示其核电站中位置的知识，用作需要环境辅助疲劳管理，满足长期操作环境辅助疲劳指导方针的位置。环境辅助疲劳筛选方法提供了选择这些位置的基本原理。核电站所有者可以在满足法规要求的同时，避免正式疲劳分析的必要性，减少需要环境辅助疲劳评估的组件位置的总数，实现成本最小化，确定限制其核电站的环境辅助疲劳位置。每种环境辅助疲劳筛选方法均提供一种统一的方法，用于确定在整个核电站延长运营期内需要进行环境辅助疲劳评估和管理的限制位置。

如何应用这些成果?

在本报告中描述的三种环境辅助疲劳筛选方法中，任何一种均可以用于确定核电站实现长期运营需要进行环境辅助疲劳评估的 1 类位置。

学习和参与机会

- 美国机械工程师协会等国际规范和标准委员会可能会发现，该报告对疲劳计算方法的可能改进有帮助。
- 材料可靠性计划(MRP)会议。
- 沸水反应堆容器和堆内构件项目（BWRVIP）会议。

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计划：核电，P41；沸水反应堆容器和堆内构件（BWRVIP），P41.01.03；压水反应堆材料可靠性（MRP），P41.01.04

执行类别：参考文献

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目录

Méthodes de recensement des zones impactées par la prise en compte de la fatigue environnementale (révision 1)

3002018262

Rapport final, novembre 2020

Chef de projet EPRI
G. Stevens

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ELECTRIC POWER RESEARCH INSTITUTE

SYNTHÈSE

Numéro de livrable : 3002018262

Type de produit : Rapport technique

Titre du produit : Environmentally Assisted Fatigue Screening Methods (Revision 1)

PUBLIC VISE : Personnel des sociétés de distribution d'énergie nucléaire réalisant des évaluations de la fatigue des composants

PUBLIC SECONDAIRE : Personnel des sociétés de distribution d'énergie nucléaire travaillant à des applications d'exploitation à long terme

OBJET ESSENTIEL DE LA RECHERCHE

Quels sont quelques-uns des processus de recensement des zones impactées par la prise en compte de la fatigue environnementale (EAF) qui peuvent être utilisés pour répondre aux exigences réglementaires en matière d'identification des emplacements de composants de centrale susceptibles de limiter la fatigue environnementale dans l'enveloppe de pression de liquide de refroidissement d'un réacteur ?

PRÉSENTATION DE LA RECHERCHE

La plupart des organismes de réglementation exigent des propriétaires de centrale nucléaire qu'ils donnent des valeurs acceptables du facteur d'utilisation cumulative (cumulative usage factor, CUF_{en}) de fatigue environnementale dans le cadre de toute exploitation à long terme en rapport avec les composants d'enveloppe de pression de liquide de refroidissement d'un réacteur qui limitent la fatigue environnementale. Par conséquent, aux États-Unis, toutes les centrales nucléaires faisant une demande de prolongation d'exploitation sur 60 ou 80 ans doivent prendre en charge la fatigue environnementale conformément aux recommandations des documents Generic Aging Lessons Learned (GALL) ou Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) de la Commission de réglementation nucléaire américaine (NRC). De nombreuses centrales non américaines devant elles aussi prendre en charge la fatigue environnementale en vue de prolonger leur durée de vie utilisent GALL ou des lignes directrices semblables.

En août 2012, l'EPRI a publié un processus de recensement pour la fatigue environnementale au sein de son rapport 1024995 intitulé *Recensement en fonction de la fatigue environnementale (EAF) : Processus et base technique pour l'identification des emplacements limitant l'EAF.* Le processus de recensement des zones impactées par la prise en compte de la fatigue environnementale défini dans le rapport de l'EPRI a été élaboré pour décrire la base technique d'une approche de recensement au regard de la fatigue environnementale répondant aux recommandations GALL en matière d'identification d'« emplacements supplémentaires de composants de centrale dans l'enveloppe de pression de liquide de refroidissement d'un réacteur s'ils limitent davantage la fatigue environnementale que ceux évoqués dans NUREG/CR-6260 ». Ce rapport, qui met à jour et remplace 1024995, présente des méthodes de recensement des zones impactées par la prise en compte de la fatigue environnementale pouvant servir à recenser des composants critiques de centrale au regard de la fatigue environnementale et d'identifier les emplacements susceptibles de limiter la fatigue environnementale, conformément aux lignes directrices GALL et à d'autres recommandations réglementaires.

PRINCIPALES CONCLUSIONS

- Le présent rapport adopte les enseignements tirés d'applications antérieures du rapport de recensement des zones impactées par la prise en compte de la fatigue environnementale de 2012 de l'EPRI, qualifié dans les présentes de méthode de recensement des zones impactées par la prise en compte de la fatigue environnementale de Structural Integrity Associates, Inc. (SIA). Il permet de réduire les valeurs modérées non nécessaires lors du recensement des zones impactées de tuyauterie de classe 1 et tient compte des demandes de renseignements supplémentaires de la NRC.
- Le présent rapport, qui tient compte des dernières recommandations pour le renouvellement des permis d'exploitation, met à jour la méthode de recensement des zones impactées par la prise en compte de la fatigue environnementale de SIA au besoin.
- Le présent rapport documente deux autres méthodes disponibles de recensement par la prise en compte de la fatigue environnementale qui ont été développées et utilisées par des centrales nucléaires du monde entier. Il s'agit de la méthodologie de recensement des zones impactées par la prise en compte de la fatigue environnementale de Westinghouse Electric Company LLC (WEC) et de la méthodologie de recensement des zones impactées par la prise en compte de la fatigue environnementale d'Électricité de France (EDF).

POURQUOI EST-CE IMPORTANT ?

Les méthodes de recensement des zones impactées par la prise en compte de la fatigue environnementale figurant dans le présent rapport aideront les propriétaires de centrale à prouver qu'ils ont identifié, dans leurs centrales, les emplacements nécessitant des actions de gestion de la fatigue environnementale afin de répondre aux recommandations liées à la fatigue environnementale pour une exploitation à long terme. Les méthodes de recensement des zones impactées par la prise en compte de la fatigue environnementale fournissent une base permettant de sélectionner ces emplacements. Les propriétaires de centrales peuvent faire baisser les coûts encourus en évitant les analyses de fatigue formelles et en réduisant le nombre total d'emplacements de composants nécessitant une évaluation de la fatigue environnementale, tout en répondant aux exigences réglementaires visant à déterminer les emplacements susceptibles de limiter la fatigue environnementale dans leurs centrales. Les méthodes de recensement des zones impactées par la prise en compte de la fatigue environnementale offrent une approche uniforme permettant de déterminer les emplacements susceptibles de limiter la fatigue environnementale, ces emplacements nécessitant une évaluation et une gestion de la fatigue environnementale lorsque la durée d'exploitation d'une centrale se prolonge.

COMMENT APPLIQUER LES RÉSULTATS

L'une ou l'autre des trois méthodologies de recensement des zones impactées par la prise en compte de la fatigue environnementale décrites dans ce rapport peuvent servir à déterminer les emplacements de classe 1 qui, dans les centrales nucléaires, nécessitent une évaluation de la fatigue environnementale à des fins d'exploitation à long terme.

POSSIBILITÉS D'APPRENTISSAGE ET D'ENGAGEMENT

- Des organismes de codes internationaux et des comités de normalisation, tels que l'American Society of Mechanical Engineers, trouveront sans doute le rapport utile pour continuer à affiner les méthodes de calcul de la fatigue.
- Réunions du Programme de fiabilité des matériaux (MRP).
- Réunions du Projet relatif aux équipements internes et aux cuves des réacteurs à eau bouillante (BWRVIP).

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PROGRAMMES : Nuclear Power, P41 ; Boiling Water Reactor Vessels and Internals (BWRVIP), P41.01.03 ; Pressurized Water Reactor Materials Reliability (MRP), P41.01.04

CATEGORIE DE MISE EN ŒUVRE : Référence

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TABLE DES MATIERES

Screening-Methoden für umgebungsbedingte Materialermüdung (Revision 1)

3002018262

Abschlussbericht, November 2020

EPRI-Projektmanager
G. Stevens

Für dieses Produkt gelten alle oder ein Teil der Anforderungen des EPRI Nuclear Quality Assurance-Programms.

YES NO

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KURZFASSUNG

Ergebnisnummer: 3002018262

Produkttyp: Technischer Bericht

Produkttitel: Environmentally Assisted Fatigue Screening Methods (Revision 1)

PRIMÄRE ZIELGRUPPE: Mitarbeiter von Kernkraftwerken, die an Bewertungen der Ermüdung von Komponenten arbeiten

SEKUNDÄRE ZIELGRUPPE: Mitarbeiter von Kernkraftwerken, die an Anwendungen für den Langzeitbetrieb arbeiten

WICHTIGE FORSCHUNGSFRAGE

Welche Screening-Verfahren für umgebungsbedingte Materialermüdung (environmentally assisted fatigue, EAF) können verwendet werden, um die behördlichen Vorgaben zu erfüllen und einschränkende anlagenspezifische Komponentenorte in der druckführenden Umschließung des Reaktorkühlmittels zu ermitteln?

FORSCHUNGSÜBERSICHT

Die meisten Aufsichtsbehörden verlangen von den Eigentümern von Kernkraftwerken den Nachweis akzeptabler Ermüdungswerte des kumulativen Nutzungsfaktors (CUF_{en}) der EAF für den langfristigen Anlagenbetrieb für einschränkende Komponenten der druckführenden Umschließung des Reaktorkühlmittels. Daher müssen alle US-Kernkraftwerke, die eine 60- oder 80-jährige verlängerte Betriebsgenehmigung beantragen, die EAF in Übereinstimmung mit den Richtlinien der U.S. Nuclear Regulatory Commission (NRC) Generic Aging Lessons Learned (GALL) oder Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) behandeln. Viele Nicht-US-Kernkraftwerke müssen sich für einen erweiterten Anlagenbetrieb ebenfalls mit der EAF befassen, und diese Anlagen verwenden entweder GALL oder ähnliche Richtlinien.

Im August 2012 veröffentlichte das EPRI im Bericht 1024995, *Environmentally Assisted Fatigue Screening: Process and Technical Basis for Identifying EAF Limiting Locations,* ein EAF-Screening-Verfahren. Das im EPRI-Bericht erwähnte EAF-Screening-Verfahren wurde entwickelt, um die technische Grundlage für einen EAF-Screening-Ansatz zu beschreiben, der den GALL-Richtlinien entspricht, um „zusätzliche anlagenspezifische Komponentenorte in der druckführenden Umschließung des Reaktorkühlmittels zu ermitteln, wenn diese möglicherweise stärker einschränkend sind als die in NUREG/CR-6260 berücksichtigten". Dieser Bericht aktualisiert und ersetzt 1024995 und stellt drei EAF-Screening-Methoden vor, die zum Screening kritischer Anlagenkomponenten für die EAF verwendet werden können, um die einschränkenden Orte in der Anlage zu ermitteln und GALL und andere behördliche Richtlinien zu erfüllen.

WICHTIGSTE ERKENNTNISSE

- Dieser Bericht übernimmt die Erkenntnisse aus früheren Anwendungen des EPRI 2012 EAF Screening Reports, die in diesem Bericht als „Structural Integrity Associates, Inc. (SIA) EAF Screening Method" (EAF-Screening-Methode von SIA) gekennzeichnet sind. Dazu gehört die Reduzierung unnötig konservativer Einstellungen beim Screening der Orte von Rohrleitungskomponenten der Klasse 1 sowie die Berücksichtigung früherer Anfragen der US-NRC nach zusätzlichen Informationen.
- Dieser Bericht aktualisiert die EAF-Screening-Methode von SIA, um bei Bedarf neuere SLR-bezogene Richtlinien zu berücksichtigen.
- Dieser Bericht dokumentiert zwei weitere verfügbare EAF-Screening-Methoden, die von Kernkraftwerken weltweit entwickelt und eingesetzt wurden. Dazu gehören die EAF-Screening-Methoden der Westinghouse Electric Company LLC (WEC) und von Électricité de France (EDF).

WARUM DAS WICHTIG IST

Die in diesem Bericht vorgestellten EAF-Screening-Methoden ermöglichen den Anlagenbesitzern, Kenntnisse über jene Orte in ihren Anlagen nachzuweisen, für die das EAF-Management langfristig die EAF-Richtlinien erfüllen muss. Die EAF-Screening-Methoden liefern die Begründung für die Auswahl dieser Orte. Anlagenbesitzer können ihre Kosten minimieren, indem sie die Notwendigkeit formaler Ermüdungsanalysen vermeiden und die Gesamtzahl der Komponentenorte, die eine EAF-Bewertung erfordern, reduzieren, während sie gleichzeitig die behördlichen Anforderungen zur Bestimmung von EAF-Orten erfüllen, die ihre Anlagen einschränken. Die EAF-Screening-Methoden liefern jeweils einen einheitlichen Ansatz zur Bestimmung der einschränkenden Orte, die EAF-Bewertung und -Management über längere Betriebszeiten der Anlage erfordern.

DIE ANWENDUNG DER ERGEBNISSE

Alle drei in diesem Bericht beschriebenen EAF-Screening-Methoden können zur Bestimmung der Orte der Klasse 1 in Kernkraftwerken verwendet werden, die für den Langzeitbetrieb eine EAF-Bewertung erfordern.

MÖGLICHE LERN- UND GESCHÄFTSCHANCEN

- Internationale Kodex- und Normenausschüsse, wie die American Society of Mechanical Engineers, könnten den Bericht für mögliche zukünftige Verfeinerungen der Ermüdungs-Berechnungsmethoden nützlich finden.
- Meetings zum Material-Zuverlässigkeitsprogramm (MRP)
- Meetings zum Siedewasserreaktor-Behälter- und Einbauten-Projekt (BWRVIP)

EPRI-KONTAKTPERSON: Gary L. Stevens, Technischer Geschäftsführer, gstevens@epri.com

PROGRAMME: Nuclear Power, P41; Boiling Water Reactor Vessels and Internals (BWRVIP), P41.01.03; Pressurized Water Reactor Materials Reliability (MRP), P41.01.04

UMSETZUNGSKATEGORIE: Referenz

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INHALT

環境誘起疲労スクリーニング法（改訂 1 版）

3002018262

2020 年 11 月　最終報告書

EPRI プロジェクトマネージャー
G. Stevens

本成果物には、EPRI 原子力品質保証プログラム (Nuclear Quality Assurance Program) の要件の全てまたは一部が適用される。

YES NO

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納品番号：**3002018262**

製品タイプ：**技術報告書**

製品タイトル：**Environmentally Assisted Fatigue Screening Methods (Revision 1)**

主な対象読者: 原子力事業者で、構成部品の疲労評価に携わる職員

二次的な対象読者: 原子力事業者で、長期運転の適用に携わる職員

主な研究課題

原子炉冷却材圧力バウンダリ内における、発電所固有の構成部品の制限位置を特定し、規制ガイダンスを満たすために使用できる環境誘起疲労（EAF）スクリーニングプロセスにはどのようなものがあるか？

研究概要

多くの規制当局は、原子力発電所の所有者に、原子炉冷却材の圧力バウンダリ構成部品を制限するための、長期発電所運転の許容可能な疲労 EAF 累積使用係数（CUF_{en}）値を示すことを要求している。従って、60 年間または 80 年間の長期運転許認可を申請するすべての米国の原子力発電所は、米国原子力規制委員会（NRC）の「一般的な経年脆化について得られた知見（GALL）」、または「二回目の許認可更新に向けた、一般的な経年脆化について得られた知見（GALL-SLR）」ガイダンスに沿った EAF に対処しなければならない。米国以外の多くの原子力発電所も、発電所の長期運転のために EAF に対処する必要があり、それらの発電所は GALL または同様のガイダンスのいずれかを使用する。

2012 年 8 月、EPRI は報告書 1024995「*環境誘起疲労スクリーニング：EAF 制限位置特定プロセスおよび技術的根拠*」の中で、EAF スクリーニングプロセスを発表した。EPRI 報告書の EAF スクリーニングプロセスは、「NUREG/CR-6260 で検討されるものより制限的である場合、原子炉冷却材圧力バウンダリ内における、追加の発電所固有の構成部品位置」を特定するための GALL ガイダンスを満たす、EAF スクリーニングアプローチの技術的根拠を説明するために開発された。GALL およびその他の規制ガイダンスを満たすために、発電所の制限位置を特定し、EAF にとって重要な発電所構成部品のスクリーニングに使用できる 3 つの EAF スクリーニング法を提供する本報告書は、1024995 を更新したものであり、1024995 に優先する。

主な所見

- 本報告書は、EPRI 2012 年度 EAF スクリーニング報告書の以前の適用から学んだ教訓を採用しており、本報告書では Structural Integrity Associates, Inc.（SIA）EAF スクリーニング法として言及されている。これには、クラス 1 配管構成部品位置をスクリーニングする際に不要な保守性の削減や、以前の米国 NRC の要求する追加情報に関する考慮事項が含まれている。
- 本報告書は、より最近の SLR 関連ガイダンスに対処するために、必要に応じて SIA EAF スクリーニング法を更新している。
- 本報告書は、世界中の原子力発電所によって開発および使用されている、他の 2 つの利用可能な EAF スクリーニング法についても文書化している。これには、Westinghouse Electric Company LLC（WEC）の EAF スクリーニング法と、Électricité de France（EDF）の EAF スクリーニング法が含まれる。

本文書の重要性

本報告書の EAF スクリーニング法により、発電所の所有者は、長期運転 EAF ガイダンスを満たすために EAF 管理が必要な位置として、発電所内の位置に関する知識を明らかにすることができるようになる。EAF スクリーニング法は、これらの位置を特定するための論理的根拠を提供する。発電所の所有者は、発電所で制限的な EAF の位置を決定するための規制要件を満たしながら、正式な疲労分析の必要性を回避し、EAF 評価を必要とする構成部品位置の全体的な数を減らすことで、コストを最小限に抑えることができる。EAF スクリーニング法はそれぞれ、発電所の長期運転期間を通じて、EAF の評価と管理を必要とする制限位置を決定するための一貫したアプローチを提供する。

研究結果の使い方

本報告書に記載されている 3 つの EAF スクリーニング法のいずれかを使用して、長期運転のために EAF 評価を必要とする原子力発電所のクラス 1 の位置を特定できる。

学習および参加の機会

- 米国機械学会などの国際的な規格および基準委員会は、本報告書が疲労計算法の将来的な改良に有用と考えるかもしれない。
- 材料信頼性プログラム（MRP）会議。
- 沸騰水型原子炉容器および内部構造物プロジェクト（BWRVIP）会議。

EPRI 連絡先： Gary L. Stevens (テクニカルエグゼクティブ)、gstevens@epri.com

プログラム： 原子力、P41、沸騰水型原子炉容器および内部構造物（BWRVIP）、P41.01.03、加圧水型原子炉材料信頼性（MRP）、P41.01.04

実施カテゴリー:参考文献

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目次

환경 피로 스크리닝 방법
(제 1 개정판)

3002018262

최종 보고서, 2020 년 11 월

EPRI 프로젝트 매니저
G. Stevens

본 보고서에는 EPRI 원자력 품질 보증 프로그램의
요건이 전부 또는 일부 적용됩니다.

YES NO

미국 전력연구소(EPRI: ELECTRIC POWER RESEARCH INSTITUTE)
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결과물 번호: 3002018262

결과물 유형: 기술 보고서

결과물 제목: Environmentally Assisted Fatigue Screening Methods (Revision 1)

1 차 대상: 구성품 피로 평가에 종사하는 원자력 발전사업자 직원

2 차 대상: 장기 운전 적용에 종사하는 원자력 발전사업자 직원

주요 연구 관련 질문

원자로 냉각제 압력 경계에서 발전소 특정 구성품 위치 제한을 식별하는 규제 지침을 충족하기 위해 사용할 수 있는 몇 가지 환경 피로(EAF) 스크리닝 공정은 무엇입니까?

연구 개요

대부분의 규제기관은 원자로 냉각제 압력 경계 구성품을 제한하기 위해 장기 발전소 운전을 위해 수용 가능한 피로 EAF 누적 사용 인자(CUF_{en}) 값을 입증하도록 원자력 발전소 소유주에게 요구합니다. 따라서 60 년 또는 80 년의 연장된 운전 인허가를 신청하는 모든 미국 원자력 발전소는 미국 원자력규제위원회(NRC) 일반 에이징 학습(GALL) 또는 후속 인허가 갱신을 위한 일반 에이징 학습(GALL-SLR) 지침과 일치하는 EAF 를 반드시 다루어야 합니다. 많은 미국 외 원자력 발전소 또한 연장된 발전소 운전을 위해 EAF 를 반드시 다루어야 하며, 이런 발전소는 GALL 또는 비슷한 지침을 사용합니다.

2012 년 8 월 EPRI 는 보고서 1024995, *환경 피로 스크리닝: EAF 제한 위치 파악을 위한 공정 및 기술 기반*에서 EAF 스크리닝 공정을 발표했습니다. EPRI 보고서의 EAF 스크리닝 공정은, "구성요소 위치가 NUREG/CR–6260 에서 고려한 것보다 더 제한적일 수 있는 경우 원자로 냉각제 압력 경계에서 추가적인 발전소 특정 구성품 위치"를 식별하는 GALL 지침을 충족한 EAF 스크리닝 접근법에 대한 기술 기반을 기술하기 위해 개발되었습니다. 이 보고서는 1024995 를 갱신하고 대체하며 GALL 및 기타 규제 지침을 충족하는 제한적 발전소 위치를 식별하기 위해 EAF 에 절대적인 발전소 구성품 스크리닝에 사용될 수 있는 세 가지 EAF 스크리닝 방법을 제공합니다.

주요 결과

- 본 보고서는 Structural Integrity Associates, Inc. (SIA) EAF 스크리닝 방법으로 식별된 EPRI 2012 EAF 스크리닝 보고서를 이전에 적용하는 과정에서 얻은 교훈을 채택합니다. 이것은 Class 1 배관 컴포넌트 위치를 스크리닝하고 추가 정보에 대한 이전 미국 NRC 요청의 고려사항을 포함할 때 불필요한 보수적 경향을 줄이는 것을 포함합니다.
- 본 보고서는 더 최근의 SLR 관련 지침을 다루기 위해 필요에 따라 SIA EAF 스크리닝 방법을 갱신합니다.
- 본 보고서는 전 세계 원자력 발전소에서 개발하여 사용 중인 이용 가능한 EAF 스크리닝 방법 두 가지를 기록합니다. 이것은 Westinghouse Electric Company LLC (WEC) EAF 스크리닝 방법 그리고 Électricité de France (EDF) EAF 스크리닝 방법을 포함합니다.

이것이 중요한 이유

이 보고서의 EAF 스크리닝 방법으로 발전소 소유주는 장기 운전 EAF 지침을 충족하기 위해 EAF 관리가 필요한 위치 역할을 할 수 있는 발전소의 위치에 대한 지식을 입증할 수 있습니다. EAF 스크리닝 방법은 이들 위치를 선택한 근거를 제공합니다. 발전소 소유주는 발전소에서 제한적인 EAF 위치를 결정하기 위한 규제 요건을 충족하면서 공식 피로 분석의 필요성을 회피하고 EAF 평가가 필요한 구성품 위치의 전체적인 수를 줄임으로써 비용을 최소화할 수 있습니다. EAF 스크리닝 방법은 각기 연장된 발전소 운전 기간 동안 EAF 평가와 관리가 필요한 제한적 위치를 결정하는 단일 접근법을 제공합니다.

결과 적용 방법

이 보고서에 기술된 세 가지 EAF 스크리닝 방법은 어느 것이든 장기 운전을 위해 EAF 평가가 필요한 원자력 발전소의 Class 1 위치를 결정하는 데 사용할 수 있습니다.

학습 및 참여 기회

- 국제 규격 및 표준위원회(예, 미국 기계학회)는 이 보고서가 피로 계산 방법을 추후 개선하는 데 유용함을 알 수 있습니다.
- 재료 신뢰성 프로그램(MRP) 회의.
- 비등수형 원자로 용기 및 내부구조물 프로젝트(BWRVIP) 회의.

EPRI 연락처: Gary L. Stevens, Technical Executive, gstevens@epri.com

프로그램: 원자력, P41; 비등수형 원자로 용기 및 내부구조물(BWRVIP), P41.01.03; 가압경수로 재료 신뢰성(MRP), P41.01.04

시행 범주: 참조문헌

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목차

Métodos de triagem de fadiga assistida pelo meio ambiente (Revisão 1)

3002018262

Relatório final, novembro de 2020

Gerente de projeto EPRI
G. Stevens

Os requisitos do Programa de Garantia de Qualidade Nuclear do EPRI se aplicam a este produto total ou parcialmente.

YES NO

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SUMÁRIO EXECUTIVO

Número do entregável: 3002018262

Tipo de produto: Relatório técnico

Título do produto: Environmentally Assisted Fatigue Screening Methods (Revision 1)

PÚBLICO PRINCIPAL: Equipes do setor de utilidades nucleares que trabalham em avaliações de fadiga de componentes

PÚBLICO SECUNDÁRIO: Equipes do setor de utilidades nucleares que trabalham em aplicações de longo prazo

QUESTÃO PRINCIPAL DE PESQUISA

Quais são alguns processos de triagem de fadiga assistida pelo meio ambiente (EAF) que podem ser usados para atender às orientações regulatórias para identificar localizações limitantes de componentes específicos da usina no limite da pressão de resfriamento do reator?

VISÃO GERAL DA PESQUISA

A maioria dos reguladores exige que proprietários de usinas nucleares demonstrem valores aceitáveis de fator (CUF_{en}) de uso cumulativo de EAF para operações de usinas de longo prazo para componentes do limite da pressão de resfriamento do reator. Portanto, todas as usinas nucleares dos EUA que solicitaram licenças de operação prolongadas de 60 ou 80 anos devem abordar EAF de forma consistente com as orientações o Comitê de Regulamentação Nuclear dos EUA, o relatório de Lições aprendidas genéricas sobre envelhecimento (GALL) ou o relatório de Lições aprendidas genéricas sobre envelhecimento para renovações de licenças subsequentes (GALL-SLR). Muitas usinas nucleares fora dos EUA também precisam abordar a EAF para operação prolongada da usina, e essas usinas usam o relatório GALL ou orientações similares.

Em agosto de 2012, o EPRI publicou o processo de triagem de EAF no relatório 1024995, *Triagem de fadiga assistida pelo meio ambiente: processo e base técnica para identificar locais com limitações de EAF.* O processo de triagem de EAF no relatório do EPRI foi elaborado para descrever a base técnica para uma abordagem de triagem de EAF que atendesse à orientação do GALL de que se deve identificar "locais de componentes específicos da usina adicionais no limite de pressão do resfriamento do reator de estes puderem ser mais limitantes do que aqueles considerados no NUREG/CR-6260." Este relatório atualiza e substitui o documento 1024995 e apresenta três métodos de triagem que podem ser usados para fazer a triagem de componentes críticos da usina em relação à EAF para identificar as localizações limitadoras da usina para atender aos requisitos do GALL e de outras diretrizes regulatórias.

RESULTADOS PRINCIPAIS

- O relatório adota lições aprendidas de aplicações anteriores do Relatório de triagem de EAF de 2012 do EPRI, identificado neste relatório como o Método de triagem de EAF da Structural Integrity Associates, Inc. (SIA). Isso inclui a redução de conservadorismos desnecessários na triagem de localizações de componentes de tubulação de Classe 1 e na inclusão da consideração de solicitações de NRC dos EUA anteriores para informações adicionais.
- Este relatório atualiza o Método de triagem de EAF da SIA conforme necessário para abordar orientações relacionadas a SLR mais recentes.
- Este relatório documenta dois outros métodos de triagem de EAF disponíveis que foram elaborados e utilizados por usinas nucleares em todo o mundo. Eles incluem o Método de triagem de EAF da Westinghouse Electric Company LLC (WEC) e o Método de triagem de EAF da Électricité de France (EDF).

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PORQUE ISSO É IMPORTANTE

Os métodos de triagem de EAF neste relatório permitirão que os proprietários demonstrem conhecimento dos locais de suas usinas que podem servir como localizações que exigem gestão de EAF para atender às orientações de EAF de operações de longo prazo. Os métodos de triagem de EAF apresentam os raciocínios usados para selecionar esses locais. Os proprietários de usinas podem minimizar custos evitando a necessidade de análises formais de fadiga e reduzindo o número geral de localizações de componentes que exigem avaliação de EAF e atender a requisitos regulatórios para determinar os locais de EAF que são limitantes em suas usinas. Cada um dos métodos de triagem de EAF apresenta uma abordagem uniforme para se determinar os locais limitantes que exigem avaliações e gestão de EAF durante os períodos de operação prolongada da usina.

COMO APLICAR OS RESULTADOS

Qualquer um dos três métodos de triagem de EAF descritos neste relatório pode ser utilizado para determinar os locais de Classe 1 em usinas nucleares que exigem avaliação de EAF para operações de longo prazo.

OPORTUNIDADES DE APRENDIZADO E ENVOLVIMENTO

- Comitês de códigos e normas internacionais, tais como a American Society of Mechanical Engineers (Sociedade Norte-americana de Engenheiros Mecânicos) podem considerar que o relatório seja útil para possíveis ajustes futuros dos métodos de cálculo de fadiga.
- Reuniões do Materials Reliability Program (Programa de confiabilidade dos materiais)
- Reuniões do projeto de vaso e parte interna de reatores de água fervente (BWRVIP).

CONTATO DO EPRI: Gary L. Stevens, Executivo técnico, gstevens@epri.com

PROGRAMAS: Energia nuclear, P41; Vasos e partes internas de reatores de água fervente (BWRVIP), P41.01.03; Confiabilidade de materiais de reatores de água pressurizada (MRP), P41.01.04

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ÍNDICE

Métodos de análisis de la fatiga asistida por el ambiente (revisión 1)

3002018262

Informe final, noviembre de 2020

Jefe de Proyecto de EPRI
G. Stevens

Son de aplicación a este producto la totalidad o parte de los requisitos del Programa de aseguramiento de la calidad nuclear de EPRI.

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RESUMEN EJECUTIVO

Número del documento para entregar: 3002018262

Tipo de producto: Informe técnico

Título del producto: Environmentally Assisted Fatigue Screening Methods (Revision 1)

PÚBLICO PRINCIPAL: personal de las empresas de servicios del sector nuclear que trabajan en las evaluaciones de la fatiga en componentes

PÚBLICO SECUNDARIO: personal de las empresas de servicios del sector nuclear que trabajan en aplicaciones con funcionamiento a largo plazo

FUNDAMENTO DE LA INVESTIGACIÓN

¿Cuáles son los procesos de análisis de la fatiga asistida por el ambiente (EAF) que pueden usarse para cumplir con las directrices normativas para identificar ubicaciones limitantes de componentes específicos de la central en la barrera de presión del refrigerante del reactor?

RESUMEN DE LA INVESTIGACIÓN

La mayoría de entidades reguladoras requieren que los propietarios de las centrales nucleares demuestren unos valores aceptables del factor de uso acumulado de la fatiga asistida por el ambiente (CUF_{en}) para el funcionamiento a largo plazo en la central de los componentes que estén en la barrera de presión del refrigerante del reactor. Por tanto, todas las centrales nucleares de EE. UU. que soliciten una prórroga de sus licencias de funcionamiento por un periodo de 60 u 80 años deben abordar la fatiga asistida por el ambiente de acuerdo con las Enseñanzas Genéricas Extraídas sobre Envejecimiento (GALL) de la Comisión Reguladora de la Energía Nuclear (NRC) de EE. UU. o las directrices de las Enseñanzas Genéricas Extraídas sobre Envejecimiento para la posterior renovación de licencias (GALL-SLR). Muchas centrales nucleares no estadounidenses deben también abordar la fatiga asistida por el ambiente para prorrogar su funcionamiento, y dichas centrales utilizan las GALL o una guía similar.

En agosto de 2012, EPRI publicó un proceso de análisis de la fatiga asistida por el ambiente en el informe 1024995: *Análisis de la fatiga asistida por el ambiente: proceso y fundamentos técnicos para la identificación de las ubicaciones limitantes de la fatiga asistida por el ambiente*. El proceso de análisis de la fatiga asistida por el ambiente del informe de EPRI se desarrolló para describir los fundamentos técnicos de un método de análisis de la EAF que siguiera las orientaciones de las GALL para identificar «ubicaciones adicionales de componentes específicos de la central en la barrera de presión del refrigerante del reactor si estas pueden ser más limitantes que las consideradas en NUREG/CR-6260». Este informe actualiza y substituye al 1024995 y proporciona tres métodos de análisis de la fatiga asistida por el ambiente que pueden usarse para analizar los componentes de la central que son cruciales para la fatiga asistida por el ambiente y así identificar las ubicaciones limitantes de la central que satisfagan las GALL y otras directrices normativas.

CONCLUSIONES PRINCIPALES

- Este informe adopta las enseñanzas extraídas de anteriores aplicaciones del Informe sobre el análisis de la fatiga asistida por el ambiente de 2012 de EPRI, que se identifican como el Método de análisis de la fatiga asistida por el ambiente de Structural Integrity Associates, Inc. (SIA). Este incluye reducir conservadurismos innecesarios al analizar las ubicaciones de componentes de tuberías de clase 1 y considerar la obtención de información adicional de solicitudes anteriores de la NRC estadounidense.
- Este informe actualiza el método de análisis de la fatiga asistida por el ambiente de SIA y lo considera necesario para abordar una guía más reciente relacionada con la renovación de posteriores licencias.
- También recoge otros dos métodos de análisis de la fatiga asistida por el ambiente disponibles que han sido desarrollados y usados por centrales nucleares de todo el mundo. Son el método de análisis de la fatiga asistida por el ambiente de Westinghouse Electric Company LLC (WEC) y el método de análisis de la fatiga asistida por el ambiente de Électricité de France (EDF).

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RESUMEN EJECUTIVO

POR QUÉ ESTO ES IMPORTANTE

Los métodos de análisis de la fatiga asistida por el ambiente de este informe permitirán a los propietarios de las centrales demostrar un conocimiento de las ubicaciones de sus centrales que pueden servir de ubicaciones que requieran una gestión de la fatiga asistida por el ambiente para cumplir las directrices relacionadas con esta durante el funcionamiento a largo plazo. Los métodos de análisis de la fatiga asistida por el ambiente proporcionan la justificación necesaria para seleccionar estas ubicaciones. Los propietarios de las centrales pueden minimizar costes evitando la necesidad de realizar análisis formales de la fatiga y reduciendo el número total de ubicaciones de componentes que requieran una evaluación de la fatiga asistida por el ambiente mientras cumplen los requisitos normativos para determinar las ubicaciones de la fatiga asistida por el ambiente que son limitantes en sus centrales. Cada uno de los métodos de análisis de la fatiga asistida por el ambiente proporciona un método uniforme para determinar las ubicaciones limitantes que requieren una evaluación de la EAF a lo largo de los periodos de funcionamiento prorrogados de la central.

CÓMO USAR LOS RESULTADOS

Cualquiera de los tres métodos de análisis de la fatiga asistida por el ambiente que se describen en este informe puede usarse para determinar las ubicaciones de clase 1 en centrales nucleares que requieran una evaluación de la fatiga asistida por el ambiente para un funcionamiento a largo plazo.

OPORTUNIDADES DE APRENDIZAJE Y PARTICIPACIÓN

- Los comités de códigos y normas internacionales, como la Sociedad Estadounidense de Ingenieros Mecánicos (ASME), pueden encontrar útil el informe para posibles mejoras futuras de los métodos de cálculo de la fatiga.
- Reuniones del Programa de fiabilidad de los materiales (MRP).
- Reuniones del Proyecto de vasija e internos del reactor de agua en ebullición (BWRVIP).

CONTACTO DE EPRI: Gary L. Stevens, ejecutivo técnico, gstevens@epri.com

PROGRAMAS: ENERGÍA NUCLEAR, P41; VASIJA E INTERNOS DEL REACTOR DE AGUA EN EBULLICIÓN (BWRVIP), P41.01.03; FIABILIDAD DE LOS MATERIALES DEL REACTOR DE AGUA A PRESIÓN (MRP), P41.01.04

CATEGORÍA DE APLICACIÓN: REFERENCIA

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CONTENIDO

Granskningsmetoder för miljöassisterad utmattning (revision 1)

3002018262

Slutrapport, november 2020

EPRI-projektansvarig
G. Stevens

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SAMMANFATTNING

Leveransnummer: 3002018262

Produkttyp: Teknisk rapport

Produkttitel: Environmentally Assisted Fatigue Screening Methods (Revision 1)

PRIMÄR MÅLGRUPP: personell vid kärnkraftsanläggningar som arbetar med komponentutmattningsutvärderingar

SEKUNDÄR MÅLGRUPP: personell vid kärnkraftsanläggningar som arbetar med tillämpningar för långtidsdrift

HUVUDSAKLIG FORSKNINGSFRÅGA

Vad finns det för granskningsprocesser för miljöassisterad utmattning (EAF, environmentally assisted fatigue) som kan användas för att uppfylla regleringsriktlinjer för att identifiera anläggningsspecifika komponentplatser i reaktorkylvätskans tryckgränsyta?

ÖVERSIKT ÖVER FORSKNINGEN

De flesta tillsynsmyndigheter kräver att ägare av kärnkraftsanläggningar uppvisar acceptabla värden för kumulativa användningsfaktorer för EAF (CUF_{en}) vid anläggningsdrift under lång tid, för att begränsa komponenter i reaktorkylvätskans tryckgränsyta. Alla kärnkraftsanläggningar i USA som ansöker om en 60 eller 80 års utökad driftlicens, måste därför behandla EAF enligt riktlinjerna U.S. Nuclear Regulatory Commission (NRC) Generic Aging Lessons Learned (GALL) eller Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR). Många kärnkraftsanläggningar utanför USA måste också behandla EAF för utökad anläggningsdrift och dessa anläggningar använder antingen GALL eller liknande riktlinjer.

I augusti 2012 publicerade EPRI en granskningsprocess för EAF i rapport 1024995, *Environmentally Assisted Fatigue Screening: Process and Technical Basis for Identifying EAF Limiting Locations*. Granskningsprocessen för EAF i EPRI-rapporten utvecklades för att beskriva det tekniska underlaget för en granskningsmetod för EAF som uppfyller GALL-riktlinjer för att identifiera ”ytterligare anläggningsspecifika komponentplatser i reaktorkylvätskans tryckgränsyta om de är mer begränsande än de som beaktas i NUREG/CR–6260.” Den här rapporten uppdaterar och ersätter 1024995 och innehåller tre granskningsmetoder för EAF som kan användas för att granska kritiska anläggningskomponenter för EAF och för att identifiera de begränsande anläggningsplatserna. Det för att i sin tur uppfylla GALL och andra regleringsriktlinjer.

HUVUDSAKLIGA SLUTSATSER

- Den här rapporten antar lärdomar från tidigare tillämpningar av EPRI 2012 EAF Screening Report, vilken identifieras som granskningsmetod för EAF från Structural Integrity Associates, Inc. (SIA) i den här rapporten. Det här inkluderar att reducera onödig försiktighet vid granskning av platser för rörkomponenter av klass 1 och tar hänsyn till tidigare begäran från NRC i USA för ytterligare information.
- Den här rapporten uppdaterar vid behov granskningsmetoden för EAF från SIA för att behandla nyare SLR-relaterade riktlinjer.
- Den här rapporten dokumenterar två andra tillgängliga granskningsmetoder för EAF som har utvecklats och används av andra kärnkraftsanläggningar runt om i världen. De här inkluderar granskningsmetoderna för EAF från Westinghouse Electric Company LLC (WEC) och Électricité de France (EDF).

PÅ VILKET SÄTT HAR DETTA BETYDELSE

Granskningsmetoderna för EAF i den här rapporten gör det möjligt för ägare av anläggningar att uppvisa kunskap om platserna i deras anläggningar som passar som platser som kräver EAF-hantering, för att uppfylla EAF-riktlinjer för drift under lång tid. Granskningsmetoderna för EAF ger motiveringen för att välja de här platserna. Ägare av anläggningar kan sänka kostnader genom att undvika behovet av formella utmattningsanalyser och reducera det totala antalet komponentplatser som kräver EAF-utvärdering, samtidigt som de uppfyller regleringskraven att bestämma EAF-platser som är begränsande i deras anläggningar. Varje granskningsmetod för EAF tillhandahåller ett konsekvent tillvägagångssätt för att bestämma de begränsande platserna som kräver EAF-utvärdering och -hantering, under anläggningarnas utökade driftsperioder.

TILLÄMPA RESULTATEN

Alla tre granskningsmetoder för EAF som beskrivs i den här rapporten kan användas för att bestämma klass 1-platserna i kärnkraftsanläggningar som kräver EAF-utvärdering för drift under lång tid.

TILLFÄLLEN FÖR INLÄRNING OCH ENGAGEMANG

- Internationella kommittéer för koder och standarder, som till exempel American Society of Mechanical Engineers, kan använda rapporten för eventuella framtida finjusteringar av beräkningsmetoder för utmattning.
- Möten gällande programmet för materialtillförlitlighet (MRP).
- Mötengällande projektet för kokvattenreaktorkärl och interna komponenter (BWRVIP).

KONTAKT HOS EPRI: Gary L. Stevens, teknisk chef, gstevens@epri.com

PROGRAM: Kärnkraft, P41, Kokvattenreaktorkärl och interna komponenter (BWRVIP), P41.01.03, materialtillförlitlighet i tryckvattenreaktorer (MRP), P41.01.04

GENOMFÖRANDEKATEGORI: Hänvisningar

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