Chapter 11

Fatigue Control in Angra Nuclear Power Plants

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Fatigue Control in Angra Nuclear Power Plants

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Abstract

The fatigue control of Nuclear Power Plants (NPP) operating in Brazil is performed through different procedures. In a first analysis, the occurrence of thermal transients is observed. If an unspecified transient occurs, a detailing of the event is carried out with the aim of reducing it to design values. At the same time, an automatic transient monitoring system will show the consequences of the transient that has occurred, as well as fatigue monitoring at selected locations. The Angra NPP were designed to operate during a period of 40 years of service. It should be noted that the Angra NPP is expected to operate for another 20 years of service. The new 60-year operational limits can be obtained by following the United States Nuclear Regulatory Commission (USNRC) regulations or through German standards (KTA). Fatigue control at Angra NPP is done by controlling the number of transients and/or the cumulative fatigue usage factor (CUF), not necessarily in that order.

Keywords: Thermal Stratification; Fatigue Monitoring; EAF Environmentally Assisted Fatigue; Fen Environmental Correction Factor; TLAA Summary from NUREG and KTA

1 Introduction

Fatigue analysis is performed to ensure the safe operation of components subject to cyclic transients, such as those generated by events where temperature changes occur (heating/cooling, safety injection, etc.), and are known as design transients.

The design transients were obtained by defining a conservative occurrence for the magnitude and frequency of most events expected to occur in a Nuclear Power Plant. A catalog of allowable design limits for 40 years of operation was provided by the NSSS Vendor (Nuclear Steam Supply System).

Based on these events, the fatigue analyzes obtain the so-called cumulative fatigue usage factor (CUF), whose limit corresponds to 1.0. It is considered that values greater than 1.0 would be the beginning of a crack in the material of the pipes and components and it is intuitive to realize that if the number of observed design transients is less than the allowable limit, the CUF will not be exceeded.

At the Angra NPP, an automatic event counting system is used and employs the operating parameters generated (temperature, pressure, flow, valve position, pump condition, etc.) by the existing instrumentation in the plant, and through appropriate algorithms, and based on the actual transients that have occurred, calculate the cumulative fatigue usage factor at a specific location.

The Angra NPP were designed to operate during a period of 40 years of service (40 Years of Operation). However, the Angra NPP should operate for additional periods of 20 years of service. This new period is known as 60-Year Transient Projected Cycles.

In the 1980´s, it was discovered in Japan that the fatigue life that occurs in severe environments could be reduced.

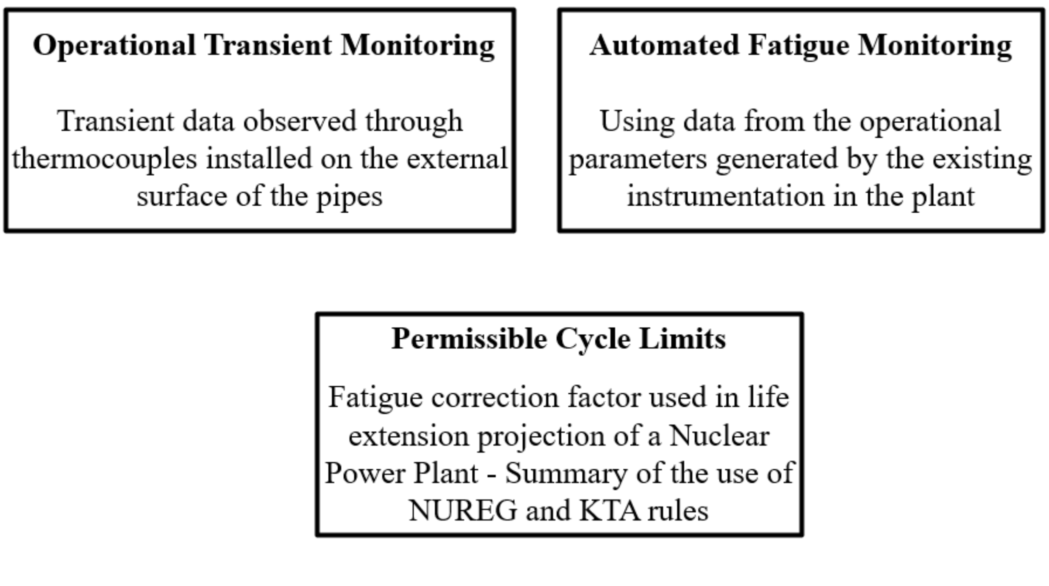
New tests carried out at the Argonne National Laboratory (ANL), O.K. Chopra and A. Sather [1990], showed that the fatigue effects in a water environment, at the nominal temperature at full power, are significantly greater than the effects of fatigue in air environment, where the fatigue curves were developed.

The Angra NPP use the fatigue curves of the ASME III code [1989] and also the German standards, Nuclear Safety Standards Commission (KTA) [1996]. The current design fatigue curves of the ASME Code were based primarily on strain–controlled fatigue tests of small polished specimens at room temperature in air.

Using the current version of the ASME code requires the application of a penalty, which was called Environmental Fatigue Correction Factor (Fen), detailed in NUREG-1801, Rev. 2 [2010]. It should be noted that this penalty reduces fatigue life as it increases the cumulative fatigue usage.

Thus, to compensate for the increase in the CUF, which is limited to 1.0, the fatigue analyzes must necessarily be revised. After reviewing the analyses, new transient limits should be considered as admissible values. An example of this threshold reduction is the Plant Heat up event, with a reduction from 200 (40 years of operation) to 150 (60-year transient projected cycles).

The procedures applied at the Angra NPP to control fatigue are presented next.



2 Operational Transient Monitoring

2.1 Observation of Operational Temperature Transients and Thermal Stratification Phenomenon

Temperature transients that occur during the operation of a Nuclear Power Plant are observed through thermocouples installed on the external surface of the pipes, in various cross sections, M. Cisternas [2009].

Two typical thermocouple configurations are used in Angra NPP:

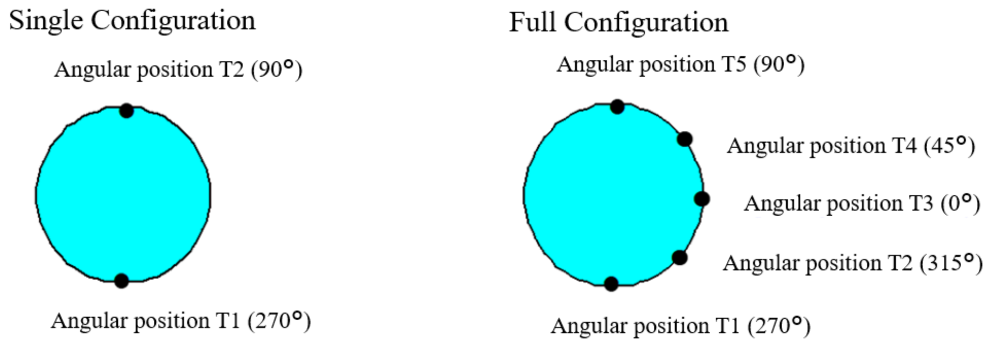


Figure 1: Typical configurations of thermal monitoring

The temperature difference between the hot water (top) and cold water (bottom) in a piping is known as thermal stratification (Figure 2).

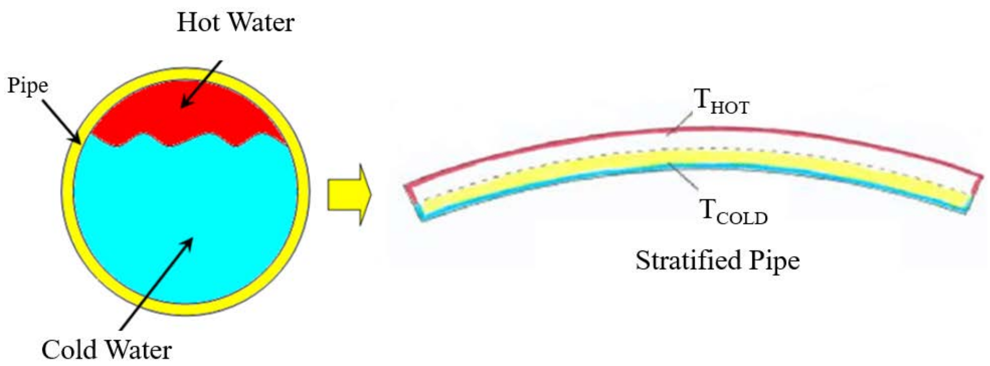


Figure 2: Thermal stratification

Thermal stratification can be divided into three types:

a) Global Stratification: Global bending effects in the piping system that can produce modified piping thermal expansion moments remote from the regions affected by the stratification. That is, can produce pipe bending and can cause the piping to exceed the thermal displacements considered in the original design.

b) Local stresses at the region of stratification related to there being a non-linear stress distribution around the circumference of a piping system. That is, local stratification causes an increase in thermal stresses on the tube wall and a consequent reduction in fatigue life.

c) Transient and / or steady through wall stress (currently defined by ΔT_1 and ΔT_2 in the code). These stresses would also not be uniform around a piping system for stratified conditions. This item will be detailed in section 3.2

2.2 Temperature Stratification in the Hot-Leg of the Reactor Coolant Line at the Angra NPP

Temperature stratification occurs in the hot-leg of the Reactor Coolant Line (RCL) during normal operation due to the outsurge flow of the hot water of the Pressurizer into the Hot Leg of the RCL.

Thus, it is important to know the behavior of temperatures inside the pipe.

At the Angra NPP, thermal stratification is observed through thermocouples installed on the external surface of the pipes.

Data acquisition is performed by a software, FieldChart Novus Version 2.0.3.1. [2013], that allows to display the data in digital and graphic format, in batches or in real time, and in real time trend and historic trend views. The software allows to zoom, superimpose and link graphs, and print or export all data.

Figure 3 shows Hot Leg location.

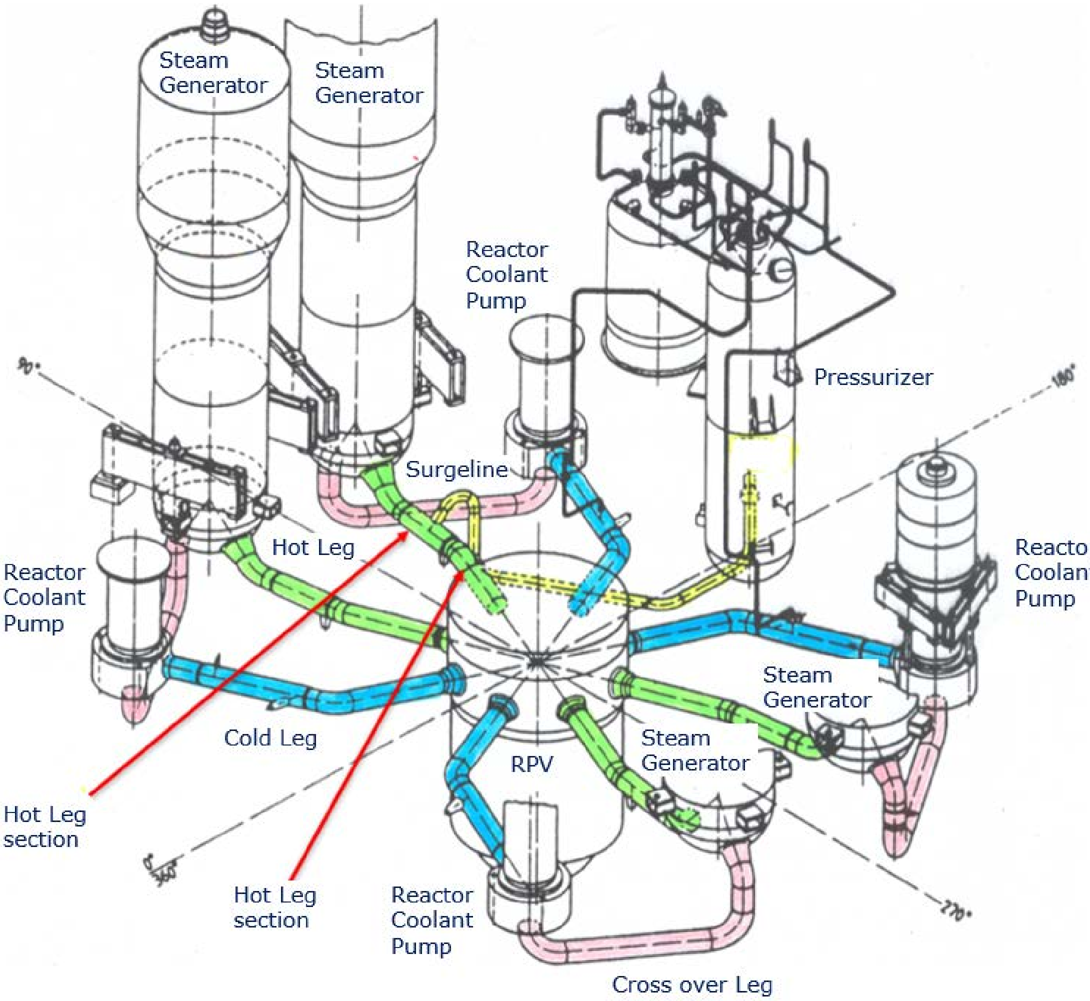


Figure 3: Hot leg location

Figure 4 shows temperature data for a section of the hot leg of the Reactor Coolant Line, located next to the Pressurizer Surgeline.

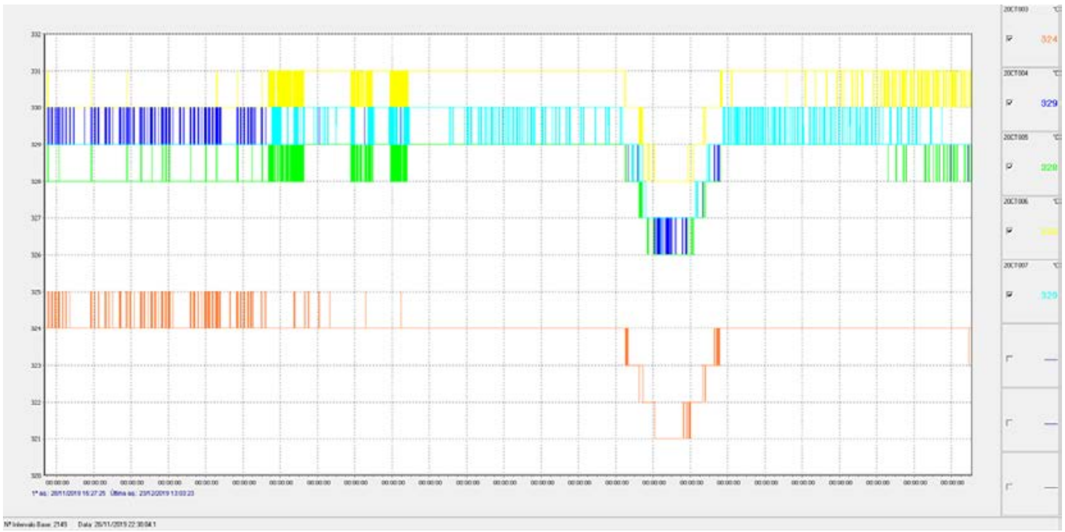


Figure 4: Data temperature at Hot Leg Location

Figure 5 shows temperature of the thermocouples installed on the external surface of the pipe.

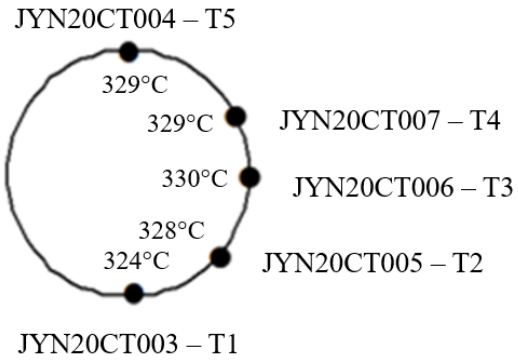


Figure 5: Temperature on the external surface of the pipe at hot leg location

Figure 6 shows the internal points where the temperature is unknown. The transfer of temperatures from the external surface to the internal points of the pipe is carried out using the Boundary Elements Method, M. Cisternas at al. [1987].

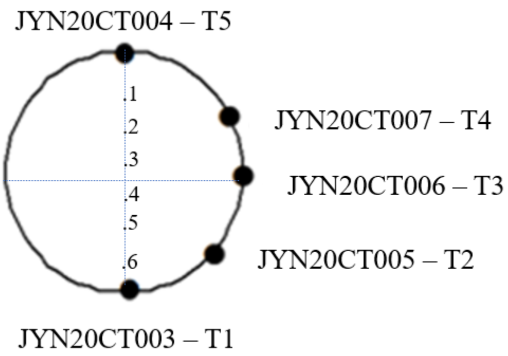


Figure 6: Internal points of the pipe where the temperature is unknown

Table 1 shows the temperature in the internal points using the Boundary Elements Method.

Table 1: Temperature in the Internal Points using the Boundary Elements Method.

INTERNAL POINTS

R	Z	POTENTIAL (U)
0.00000	343.52000	329.06112
0.00000	263.59000	329.13072
0.00000	56.52000	329.11331
0.00000	-56.52000	328.76312
0.00000	-263.59000	326.85224
0.00000	-343.52000	325.49331

Note: The expected value in normal plant operation is around 325.1 °C. This shows that the tube working stratified, presents values observed above those expected in normal plant operation.

The current Section–III design fatigue curves of the ASME Code [1989], were based primarily on strain–controlled fatigue tests of small polished specimens at room temperature in air. The Angra NPP use the fatigue curves of the project following this methodology, which correspond to the ASME III code, year 1989 and also to the German standards, Nuclear Safety Standards Commission (KTA) [1996].

The ASME Code fatigue evaluation procedures are described in NB-3600, "Piping Design," and NB-3200, "Design by Analysis". For each stress cycle or load set pair, an individual fatigue usage factor is determined by the ratio of the number of cycles anticipated during the lifetime of the component to the allowable cycles. Fatigue design curves defines the allowable number of cycles as a function of applied stress amplitude.

The CUF is the sum of the individual usage factors, and ASME Code Section III requires that at each location the CUF must not exceed 1.

2.3 Effects of Thermal Stratification

The effects of stratification are considered by determination of piping bending moments due to stratification and by calculating additional local stresses due to a non-linear temperature difference across the pipe diameter.

In a stratified section, there may be stresses developed even if the piping section is free to expand. The nonlinear top-to-bottom temperature distribution in a pipe produces a non-uniform stress distribution around the pipe circumference. The stresses have the same characteristics as a non-linear through wall stress distribution in that they will not result in gross thermal displacement of the piping system and contribute only to fatigue. If a piping program is being used to perform stress analysis, this additional stress may be transformed on equivalent ΔT_1.

The additional stresses due to through wall thermal gradients should also determine for each loading condition. Thermal transients produce large ΔT_1 and ΔT_2 stresses, and gross structural discontinuities T_a - T_b axial stress that shall be considered. However, there can be also be some through-wall temperature differences that occur with steady stratification that cannot be neglected.

Once these moments, piping parameters and / or local stresses are determined, the piping fatigue analysis can proceed using the methods commonly used for piping analysis. The code allowable for primary stress intensities should be unaffected by thermal loading. The service Level A/B stress intensity ranger shall be evaluated per NB-3650 of section III of the ASME code. Eletronuclear uses Finite Element Program KWUROHR and Finite Element Program PIPESTRESS. More detailed analysis may be conducted using the rules of NB-3200, where the stresses at multiples locations around the circumference of a component may be considered. Eletronuclear uses Finite Element Program ANSYS.

2.4 Temperature Gradients ΔT_1 And ΔT_2 and Range of Average Temperature on Gross Structural Discontinuities Ta (Tb)

The temperature distribution in the radial thickness of the pipe varies from the inner surface to the outer surface. This variation in temperature or distribution along the radial thickness of the tube is known as the thermal gradient. Figure 7 shows a temperature distribution t as a function of the radial thickness a:

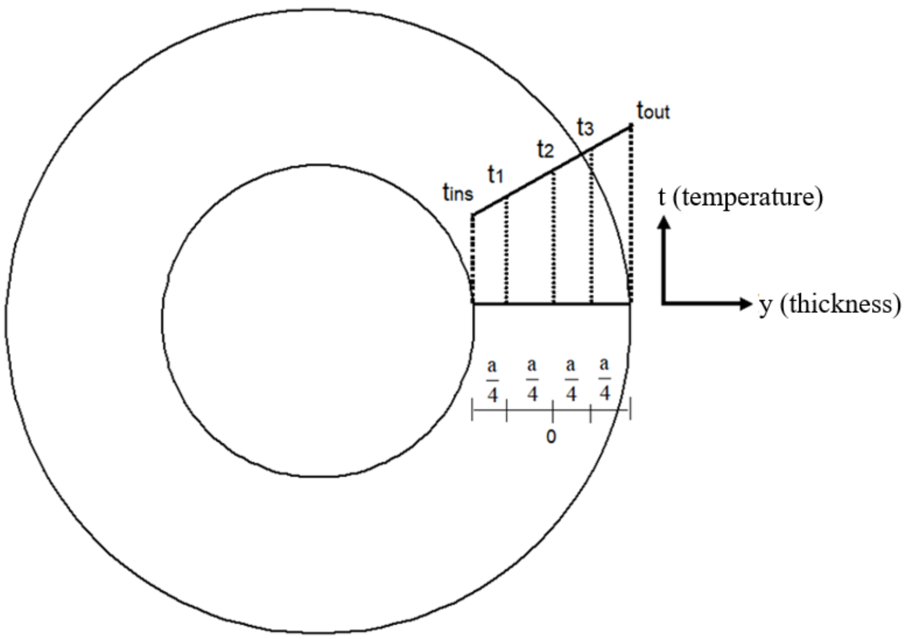


Figure 7: Temperature distribution in the thickness of the pipe

According to the ASME III code, the thermal gradient can be decomposed into three different parameters, ΔT_1, ΔT_2 and T_a-T_b, where ΔT_1 represents the temperature difference between the outer surface and internal tube, based on an equivalent linear distribution:

$$\Delta T_1 = \frac{12}{a^2} \int_{-a/2}^{a/2} y \, t(y) \, dy \tag{1}$$

$$\Delta T_1 = -\frac{5}{8} t_{ins} - \frac{3}{4} t_1 + \frac{3}{4} t_3 + \frac{5}{8} t_{out} \tag{2}$$

ΔT_2 represents the temperature difference between the surface and the equivalent linear distribution:

$$\Delta T_2 = \text{Maximum}\begin{cases} \left|t_{out} - T_{Avg}\right| - \frac{|\Delta T_1|}{2} \\ \left|t_{ins} - T_{Avg}\right| - \frac{|\Delta T_1|}{2} \\ 0 \end{cases} \tag{3}$$

T_a-T_b or Ta (Tb) represents the average temperature on the tube wall:

$$T_{Avg} = \frac{1}{a}\int_{-a/2}^{a/2} t(y)\, dy \tag{4}$$

$$T_{Avg} = \frac{t_{ins}}{8} + \frac{t_1}{4} + \frac{t_2}{4} + \frac{t_3}{4} + \frac{t_{out}}{8} \tag{5}$$

2.5 Contribution of Thermal Stratification in ASME NB-3653.2

- No terms are added in equation 10 of ASME NB-3653.2.
- Term added in Equation 11 of ASME code NB-3653.2, Structural Integrity Associates, 2008.

$$E\,\alpha\,|\Delta T_3| \tag{6}$$

Where:

$E\alpha$ = modulus of elasticity (E) times the mean coefficient of thermal expansion (α), both at room temperature

$|\Delta T_3|$ = Top Temperature - Bottom Temperature

2.6 Contribution of Thermal Stratification in ASME NB-3200

- No contribution to the term S_n = primary + secondary stress intensity; Equivalent to value calculated by equation 10 of ASME NB-3653.2.

- Contribution is added to term S_P = peak stress intensity; Equivalent to value calculated by equation 11 of ASME NB-3653.2.

2.7 Unspecified thermal stratification in the Main Feedwater line of the Angra Nuclear Power Plant

The occurrence of unspecified thermal stratification was observed in the Main Feedwater line (figure 8) of the Angra NPP.

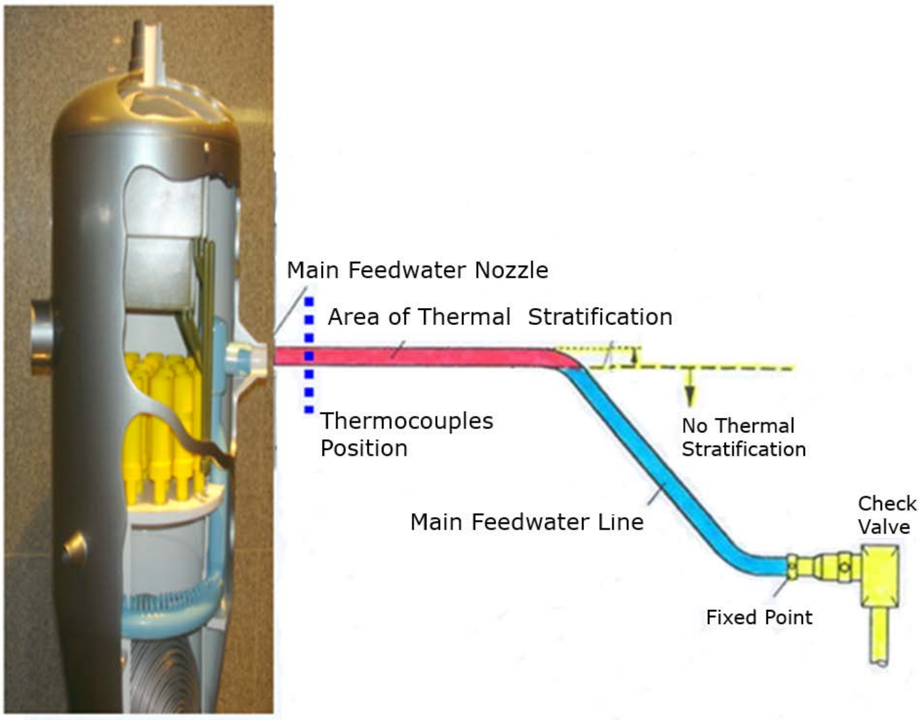


Figure 8: Main feedwater line of Angra NPP

Figure 9 shows a graph with the temperature values observed in the thermocouples installed in the Main Feedwater line of the Steam Generator, M. Cisternas [2009].

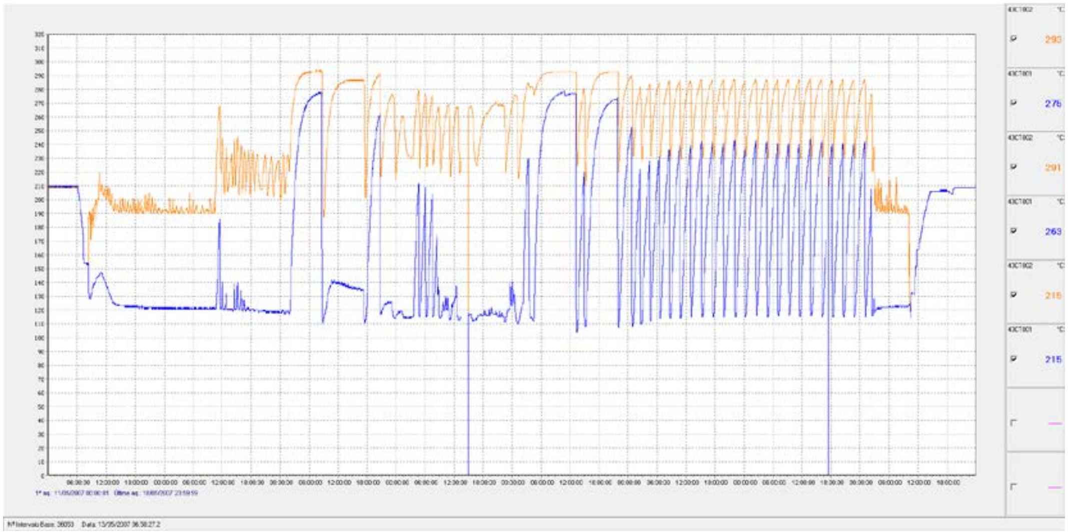


Figure 9: Temperatures observed in a cross section of the main feedwater line

Note: Observed temperature transients are not included in the NSSS vendor transient catalog with the same frequency and magnitude.

2.8 Evaluation of Plant Data

During the Start-up or Reactor Trip (no cooldown) of a Nuclear Power Plant, the supply of steam generators occurs at low feedwater mass flow rates, and significant stratification of fluid temperatures in the steam generator feedwater nozzles has been detected. This stratification leads to undesired temperature gradients which in turn lead to strain and hence stress in the nozzle material. This all contributes to accelerated fatigue and possible shortening of the life of the component. For this reason, it has become necessary to change the feeding procedure to reduce thermal loading on the steam generator feedwater nozzles.

After observing unexpected temperatures, it is necessary to perform an evaluation of the plant data. At least three distinct regions can be seen;

- Near constant temperature difference
- Near constant mean temperature difference with mild fluctuations, and
- Fluctuations and increase in temperature difference coupled with fluctuations in mean absolute temperature

The three distinct regions can be seen in figure 10:

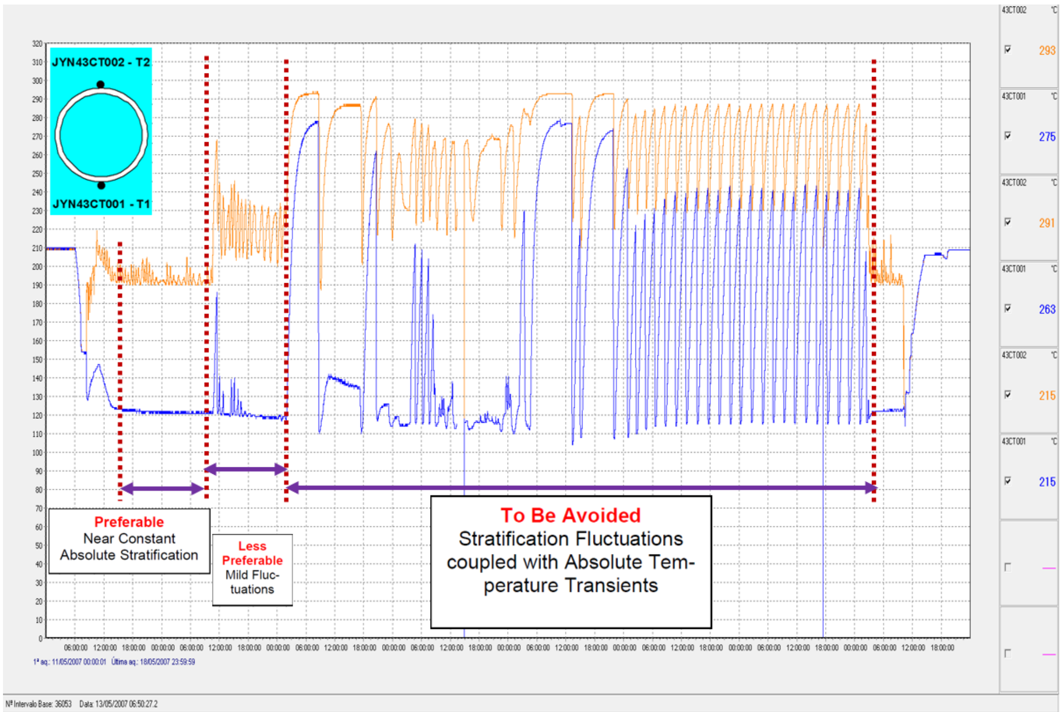


Figure 10: Evaluation of plant data

If stratification is present, it is preferable that the temperature difference remains as small and constant as possible. Fluctuations in temperature differences result in fluctuations in strains, stresses and finally fatigue.

Change the feeding procedure: Continuous feeding rather than intermittent feeding is preferable. A constant and inevitable stratification temperature difference in the main feedwater line during Start up or Reactor Trip (no cooldown) is to be expected.

Impact on pipes and components: Needs to be investigated.

2.9 Verification of the Occurrence of Thermal Stratification Phenomenon

In 1988, the USNRC (Nuclear Regulatory Commission) issued bulletin 88-08 advising about the occurrence of thermal stratification in lines connected to the Reactor Coolant Lines (RCL) of Nuclear Power Plants. In addition, bulletin 88-11 notifies that the Pressurizer Surge line should include thermal stratification loading in the fatigue analysis to verify for unexpected thermal movements that could cause the closing of the gap between the surge line and the supports provided against pipe whip restraints.

Many plants started commercial operation or were conceived before 1988, and only new reactors were designed to prevent this new load case. The other plants, had a high cost to include the thermal stratification loading, as a case of normal operation.

The following figure 11 shows the Angra NPP primary circuit, Finite Element Program, KWUROHR system [2014].

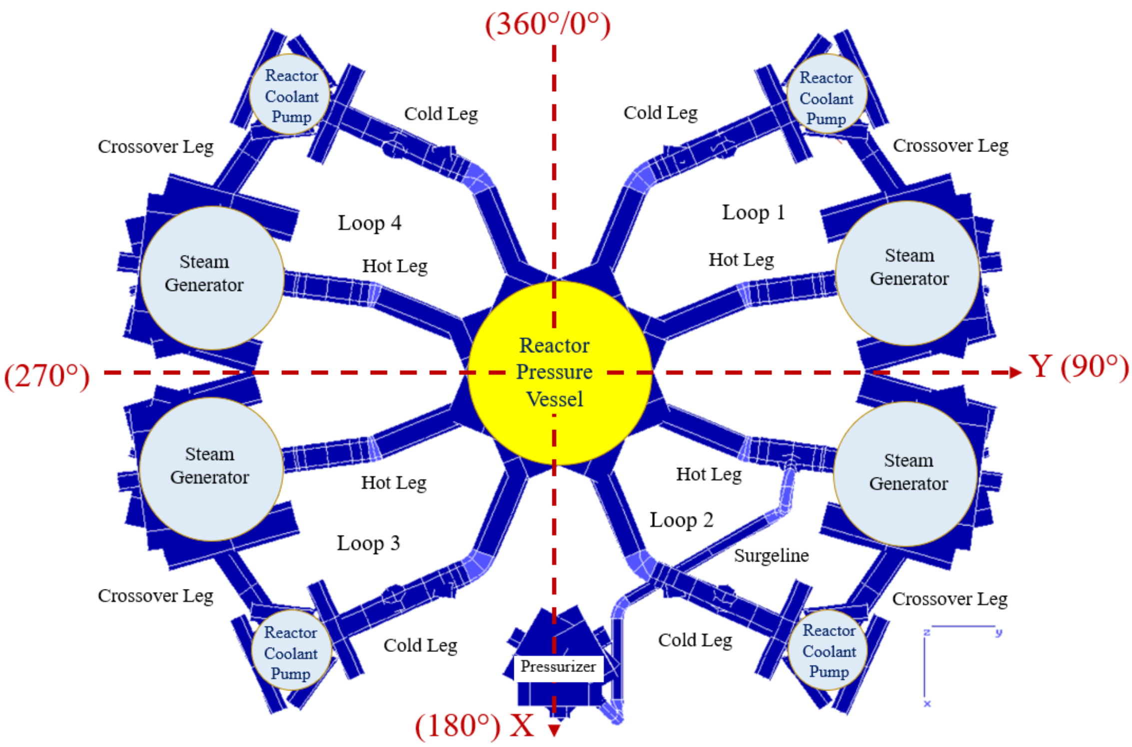


Figure 11: Angra Nuclear Power Plant primary circuit

Unexpected temperature transients were observed in loops 1 and 4 of the Main Feedwater line. In this chapter, the analysis will be concentrated on loop 4.

The displacement of the nozzle of loop 4, where it is connected to the Main Feedwater line, was obtained from the thermal analysis of the Angra NPP Reactor Coolant Line (RCL).

Figure 12 presents the finite element model of Angra NPP Reactor Coolant Line:

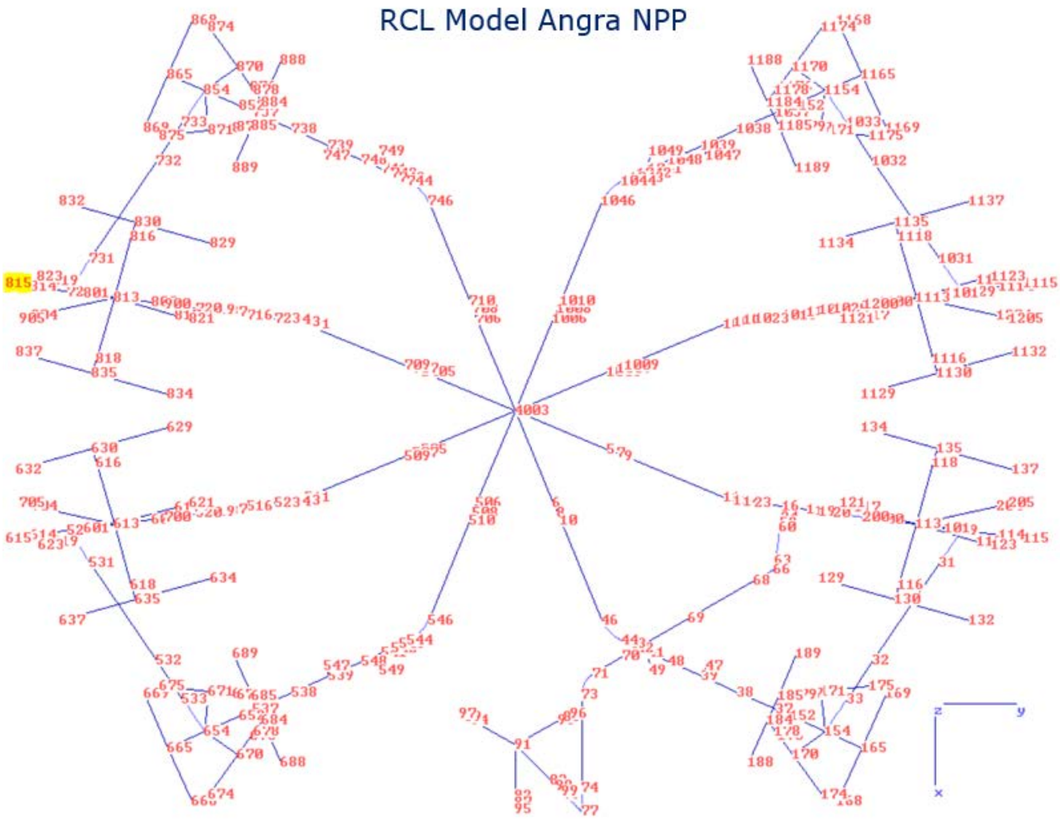


Figure 12: Finite element model of Angra NPP RCL

The results of the displacements of the connection between the nozzle of the Steam Generator loop 4 (nodal point 815) and the Main Feedwater line, obtained from the primary circuit model, are presented below in table 2:

Table 2: Displacement of the nozzle RCL/main feedwater line at loop 4 of Angra NPP.

DISPLACEMENTS and ROTATIONS at Fittings (Individual Load Cases and Superpositions)
Local Coordinate-System with Rotation-No. :

Load Case	Category	u	v	w	uu	vv	ww
		mm			mrad		
Operation		41.83	-4.22	60.52	-0.23	-1.30	0.00

Nodal Pt. : 815

The GAP adopted by the NSSS vendor for the installation of the Pipe Whip Restraints type support was 41 mm.

However, the finite element model that includes the main feedwater line shows that a higher value would be needed, and is shown in Figure 13.

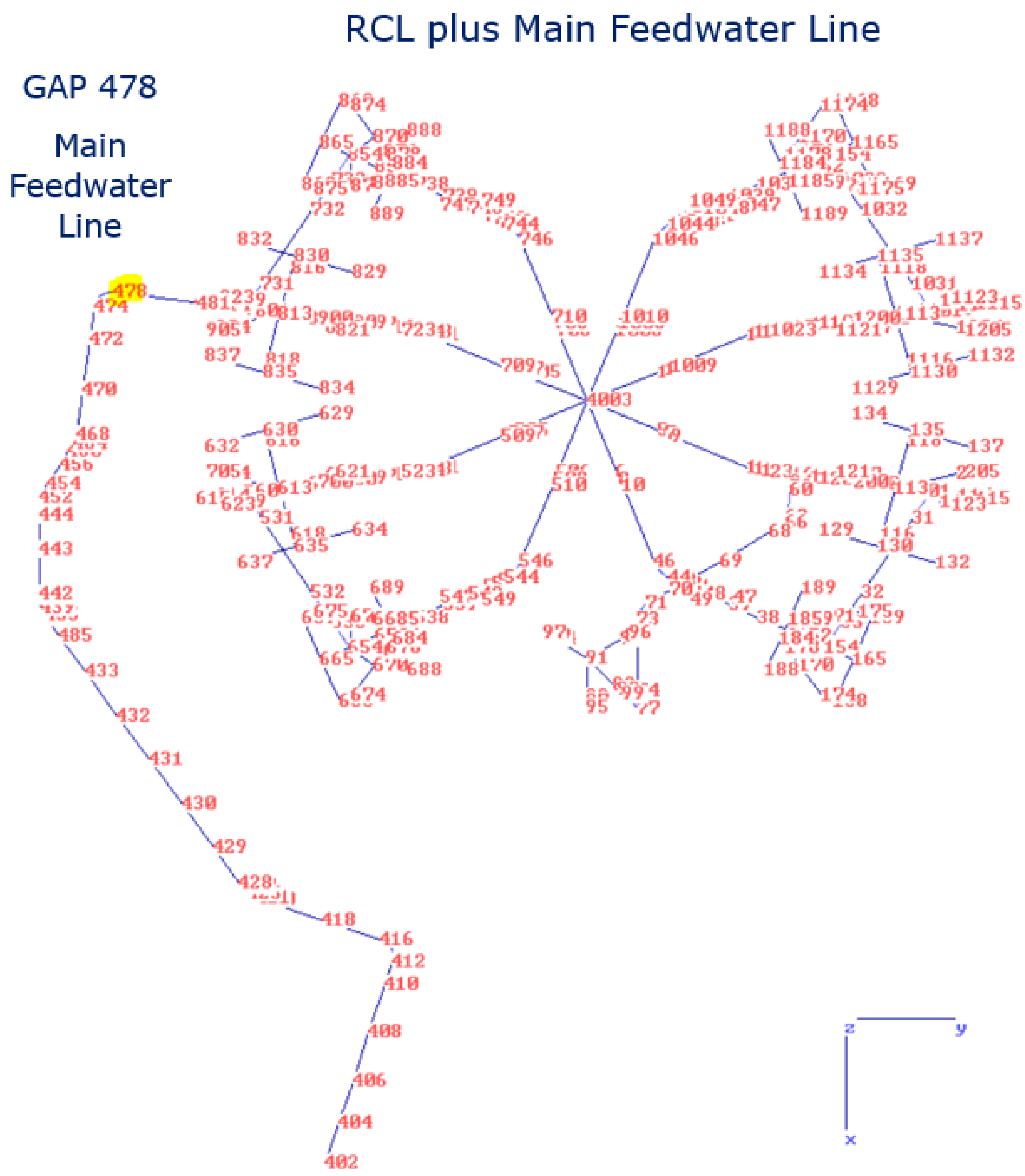


Figure 13: Finite element model of Angra NPP to RCL plus main feedwater line

The results of the displacements of the nodal point 478 of the Main Feedwater line besides Pipe Whip Restraints location are presented below in table 3:

Table 3: Displacement of the nodal point 478 of the main feedwater line.

DISPLACEMENTS and ROTATIONS at Fittings (Individual Load Cases and Superpositions)

Local Coordinate-System with Rotation-No. :

Load Case	Category	u	v	w	uu	vv	ww
		mm			mrad		
Operation		53.68	1.79	52.26	-6.24	5.05	4.24

Nodal Pt. : 478

The Angra Steam Generator including the connection to the Main Feedwater line are presented in figure 14, that shows the location of the Pipe Whip Restraints (PWR) support:

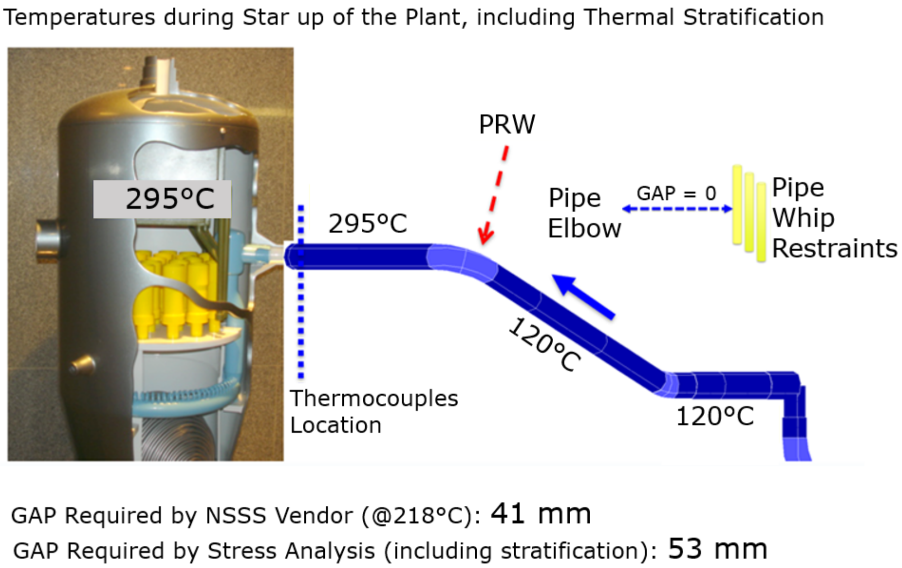


Figure 14: Pipe whip restrain location

After detecting unexpected temperature transients in the Main Feedwater line, the pipe and nozzle inspections were performed. Tests were also carried out with penetrating liquid at the welds. No damage to the pipe and nozzle welds were identified. Figure 15 shows inspection of the Pipe Whip Restraints (located on the wall) of the Main Feedwater line.

Figure 15: Inspection in main feedwater line and pipe whip restrain support

Figure 16, shows the top view of the original configuration of the Pipe Whip Restraints (PWR) support.

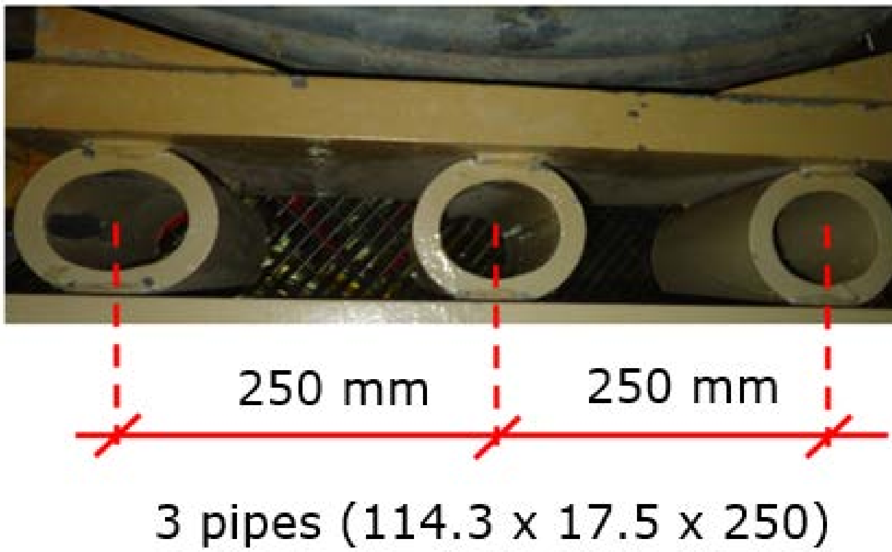


Figure 16: original configuration of the pipe whip restraints

Inspection of Pipe Whip Restraints showed that they had been affected by excessive movement of the Main Feedwater line.

In other words, the GAP between the line and the wall, where the PWR is located, has closed, causing a permanent deformation of the PWR-type support. The same effect was seen in another loop.

Figure 17 shows the top view of the Pipe Whip Restraints with permanent deformations.

Figure 17: deformed pipe whip Restraints (top view)

Finding: For this discovery, in Angra NPP, Pipe Whip Restrains supports installed beside the Main Feedwater line may be characterized as similar to bulletin 88-08.

A Comparison of deformed structure x Stress analysis with the ANSYS finite element system, [2017], considering stress distribution for 20 mm displacement is presented in figure 18, W. Menezes [2018].

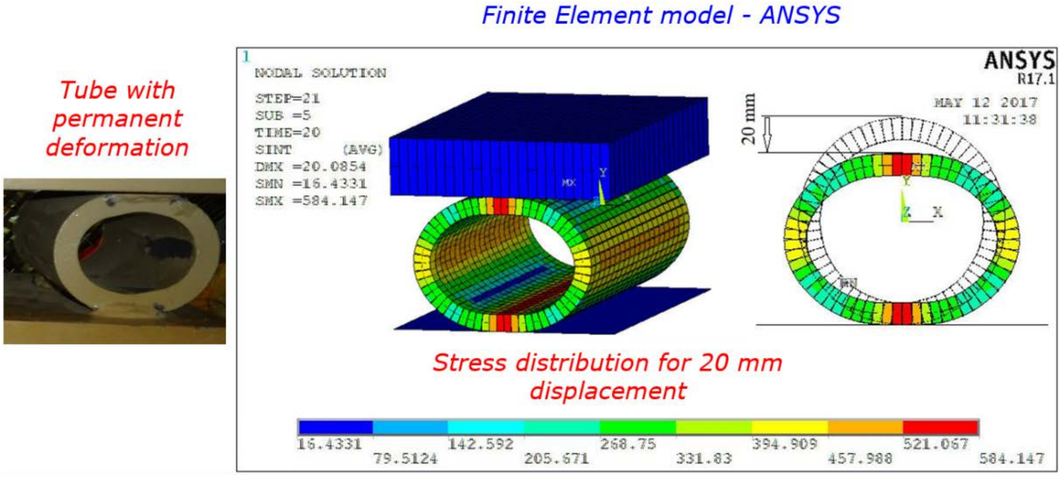


Figure 18: Deformed structure (Ovalization) x ANSYS model

In the cold shutdown mode of operation, the observed GAP is approximately 30 mm as shown in figure 19.

Figure 19: Top view of PWR in cold mode

Summary:

a) Actual GAP plus deformed model by Ansys = 50 mm.

b) Theoretical GAP required by calculation is 53 mm

c) The last inspection accomplished that the PWR was not fully deformed.

Conclusion: There was still energy in the pipeline trying to move towards the PWR.

Note: This location was not included in the check of support clearances performed during first hot and cold plant conditions.

Solution: Is required a Pipe Whip Restraints support with smaller dimensions, so that the Main Feedwater line does not have movement restrictions when it is in operation.

The new adopted Pipe Whip Restraints support is shown below in figure 20, W. Menezes [2018].

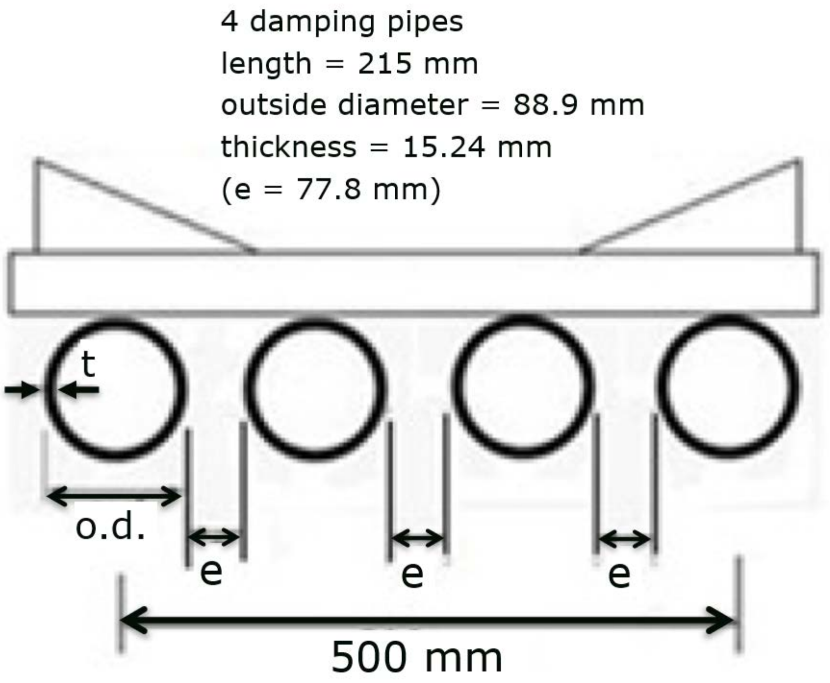


Figure 20: New pipe whip restraints

It can be noted that at Angra NPP, the phenomenon of thermal stratification was not included in the transient catalog provided by the NSSS vendor, with the magnitude that was observed. Therefore, the Main Feedwater line had to change the operating mode to reduce the number of monitored transients.

3 Automated Fatigue Monitoring

3.1 Fatigue Monitoring System in Angra NPP

An automated fatigue monitoring system is used to track the fatigue condition of Angra nuclear power plants over time.

The fatigue monitoring system performs an incremental analysis, in that all results are cumulative, reflecting all input files processed since monitoring began. As each new data file is input and processed, the results are updated to reflect the entire input history. This approach imitates the fatigue behavior of plant components, which gradually accumulate fatigue usage as they experience the various events and transients that occur during plant life.

The fatigue monitoring system allows to review the fatigue status of the plant, from the start of monitoring to the current date, in terms of plant cycles and fatigue usage factors. The staff of the plant can also sharpen focus down to a particular plant event or transient, to investigate the cause of each increment of usage. The fatigue monitoring system database for each plant unit maintains an extensive history of plant data and other computed parameters, to provide system engineers with the data they need to evaluate unusual transients or to justify alternate fatigue evaluations.

3.2 Automated Cycle Counting (ACC)

Automated Cycle Counting (ACC) analyzes the Angra NPP input data to recognize patterns, and uses those patterns to identify when recordable plant events occur. Each time an event is identified, the fatigue monitoring system records all the relevant parameters and stores it in the List of Events. It is possible also maintain the list by manually adding, editing, or deleting events.

The fatigue monitoring system is able to correctly model and analyze fatigue for all known types of Angra NPP under a wide range of operating procedures and Tech-Spec requirements, FSAR, [2020].

3.3 Plant Operating Data

The fatigue monitoring system requires data collected by plant instrumentation devices and are listed below, C. Carney et al. [2001]:

- Piping and components temperature
- Piping and components pressure
- Valve position (Open/Closed)
- Pump status (Stop/Run)
- Component flow
- Nuclear Power
- Component signal (Off/On)
- Component level (0 to 100%)
- Reactor Trip signal (normal/TRIP)
- Manual Reactor Trip signal (normal/TRIP)
- Turbine Trip signal (normal/TRIP)
- Manual safety injection signal (Off/On)

3.4 Cycle Counting Summary

The fatigue monitoring system performs the identification, recording and monitoring of traceable plant events. The cycle counting summary shows automated cycle counting events occurred during a report period (based on the events' end times).

The following reactor coolant system transients are automatically monitored in Angra NPP:

- Accumulator safety injection
- Auxiliary spray during cooldown
- Control rod drop
- High head safety injection
- Inadvertent reactor coolant system depressurization
- Inadvertent safety injection
- Inadvertent Auxiliary spray
- Large step load decrease with steam dump
- Loss of load
- Loss of power
- Operating basis earthquake
- Pressurizer cooldown
- Partial loss of reactor coolant flow
- Primary side hydrostatic test
- Primary side leakage test
- Plant heat up and cooldown
- Residual heat removal operation during cooldown
- Reactor trip from full power
    - Case A - with no inadvertent cooldown
    - Case B - with cooldown and no safety injection actuation
    - Case C - with cooldown and safety injection actuation
- Refueling
- Step load increase and decrease of 10 percent of full power
- Turbine roll test
- Unit loading and unloading between 0 and 15 percent of full power

Automated Cycle Counting of the fatigue monitoring system identify a new event whenever the event's initiation conditions are met. Once an event is identified, it will track the values for any output parameters until the termination conditions occur. At that time, it records the event in the List of Events, then returns to looking for the next initiation.

Some events are manually accounted for by a plant administrative procedure. However, it must be ensured that the event in fact occurred. Some events that did not actually occur can increase the number of events. A classic example is plant heat up, which is stopped before reaching the nominal temperature condition.

The logic of the automatic cycle counting procedure uses a threshold value for the Reactor Coolant Lines temperature values in the events plant heat up and cooldown.

These two events are not counted if the average temperature difference of the Reactor Coolant Line between the previous temperature of the event and the current temperature is $\Delta T \leq 83$ °C, that is, for the accounting of the heating event, the difference of average temperature of the Reactor Coolant Line between the previous temperature of the event and the temperature current must be above 83°C and must remain above this value for at least 5 minutes.

An example of the Angra NPP Cycle Summary is shown in Table 4.

Table 4: Example of transient cycle summary.

Cycle Summary - Angra

File Edit View Cycles Tools Window Help

Results reflect the period from:
Not Filtered: All Event Types displayed

Event type	Initial	Increment	Ending	flag	Allowable
ACC. Safety Injection	0	0	0		
Aux. Spray During CD	2	0	2		
Control Rod Drop	3	0	3		
High Head Safety Injection	21	1	22		
Inadv RCS Depress	0	0	0		
Inadv. Safety Injection	21	1	22		
Inadvertent Aux Spray	0	0	0		
Large Step Load Decrease	8	0	8		
Loss of Load w/o RX Trip	0	0	0		
Loss of Off-Site Power	8	0	8		
Operating Basis Earthquake (OBE)	0	0	0		
PZR Cooldown	75	2	77		
Partial Loss of Flow	0	0	0		
Primary Side Hydrostatic test	1	0	1		
Primary Side Leak Test	4	0	4		
RCS Cooldown	78	2	80		
RCS Heatup	78	2	80		
RHR Operation during Cooldown	78	2	80		
Reactor Trip (CD and SI)	0	0	0		
Reactor Trip (CD no SI)	0	0	0		
Reactor Trip (No Cooldown)	123	3	126		
Refueling	27	1	28		
Step Load Decrease of 10% power	71	3	74		
Step Load Increase of 10% power	87	4	91		
Turbine Roll Test	1	0	1		
Unit Loading from 0-15% Power	95	1	96		
Unit Unloading from 15-0% Power	100	1	101		

The control of cycles in accounted events is carried out through limits of alarm and related actions. In general, all locations have Cumulative Usage Factor (CUF) below the design limits. Controlling cycles in accounted events allows you to ensure that no site exceeds the limits of the cumulative fatigue usage provided for in the original design.

Output results must be reviewed and modified as necessary to correct errors in the numbers of cycles or usage factors calculated by fatigue monitoring system. These errors occur when the plant computer outputs are modified to produce some event logic in order to perform required surveillance testing, thus indicating transients that did not actually happen. An example of an erroneous event is when a Reactor Shut down (TRIP) signal is generated without a corresponding reduction in power.

This event should be excluded from data files and the fatigue monitoring system should be run again in order to update the results. If data is lost during generation of the data files so than an actual transient is not included, then this transient must be manually input into fatigue monitoring system.

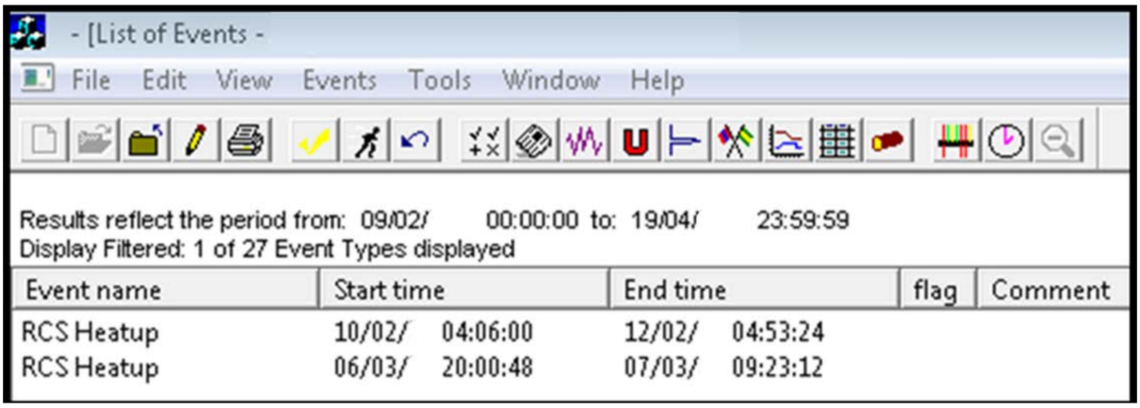
With the fatigue monitoring system, it is possible to review the fatigue status of the plant, from the start of monitoring to the current date, in terms of plant cycles and fatigue usage factors. It is possible also sharpen focus down to a particular plant event or transient, to investigate the cause of each increment of usage. The fatigue monitoring system database for the Nuclear Power Plant maintains an extensive history of plant data and other computed parameters, to provide system engineers with the data they need to evaluate unusual transients or to justify alternate fatigue evaluations (such as partial cycle counts, although this is not licensed in Angra NPP).

3.5 List of Events

The List of Events shows a list of all recorded plant events over a given period of time (the report period). The event record includes the starting date/time and the duration of the event.

Example of List of Events that occurred during a Plant heat up are shown in Table 5.

Table 5: List of events for reactor coolant line heat up during a plant cycle.


- [List of Events -

File Edit View Events Tools Window Help

Results reflect the period from: 09/02/ 00:00:00 to: 19/04/ 23:59:59
Display Filtered: 1 of 27 Event Types displayed

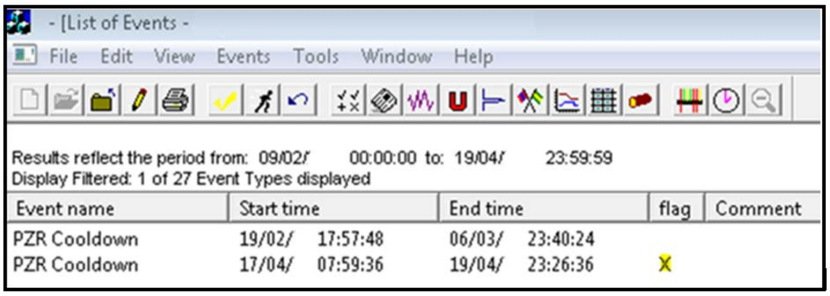
Event name	Start time	End time	flag	Comment
RCS Heatup	10/02/ 04:06:00	12/02/ 04:53:24		
RCS Heatup	06/03/ 20:00:48	07/03/ 09:23:12		

3.6 Verification of Warning Displays in a List of Events Table

If any device exceeds the plant's operating limits, a warning will be activated.

In this case the index flag will be activated with the letter X. This can be seen in table 6.

Table 6: List of events for pressurizer cool down during a plant cycle.


- [List of Events -

File Edit View Events Tools Window Help

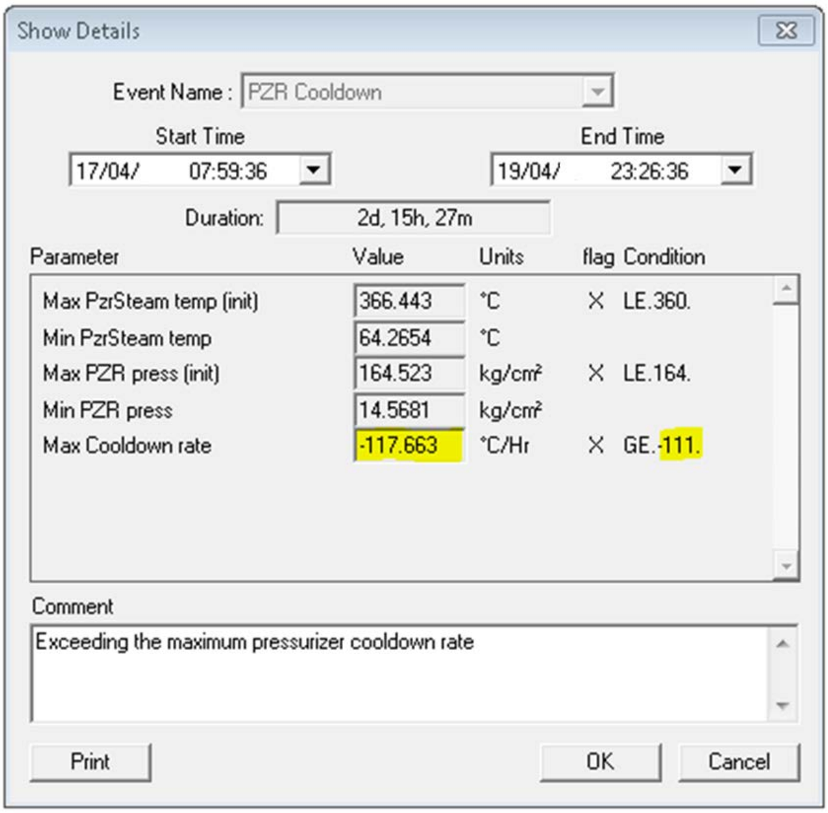
Results reflect the period from: 09/02/ 00:00:00 to: 19/04/ 23:59:59

Display Filtered: 1 of 27 Event Types displayed

Event name	Start time	End time	flag	Comment
PZR Cooldown	19/02/ 17:57:48	06/03/ 23:40:24		
PZR Cooldown	17/04/ 07:59:36	19/04/ 23:26:36	X	

Example of event details showing the X-activated flag are shown in table 7.

Table 7: Details list of events.


Show Details

Event Name : PZR Cooldown

Start Time: 17/04/ 07:59:36 End Time: 19/04/ 23:26:36

Duration: 2d, 15h, 27m

Parameter	Value	Units	flag	Condition
Max PzrSteam temp (init)	366.443	°C	X	LE.360.
Min PzrSteam temp	64.2654	°C		
Max PZR press (init)	164.523	kg/cm²	X	LE.164.
Min PZR press	14.5681	kg/cm²		
Max Cooldown rate	-117.663	°C/Hr	X	GE.-111.

Comment: Exceeding the maximum pressurizer cooldown rate

Print OK Cancel

The details presented in table 7 for a Pressurizer event, shows the fatigue monitoring system verifying the cooling rate every 30 continuous minutes (rate using a 30-minute running average), from a preset value (360°C) to the temperature considered as steady-state (below 65.6°C). Below the stationary value, it is considered that there is no metal fatigue. This class of data allows the fatigue monitoring system verify design parameters with values presented by plant instrumentation during a cycle or event.

3.7 Fatigue Monitoring in Selected Locations - Cycle-Based Fatigue

Fatigue monitoring in selected locations is performed using the Cycle-Based Fatigue (CBF) methodology that utilizes the compiled Automated Cycle Counting event list to compute fatigue usage for monitored fatigue locations based on event patterns.

In Angra there are 2 types of Cycle-Based Fatigue locations:

a) Per-Cycle (CBPC).
b) Event Pairing (CBEP)

3.7.1 Per-Cycle Usage (CBPC)

In the Per-Cycle Usage locations, a specific fatigue usage increment is associated with each type of event. This usage increment can either be a fixed amount for all events of a given type, or it can be a “partial-cycle” usage based on one or more of the event's recorded parameters. The total fatigue usage for each CBPC location is the sum of the individual usage increments of all of the recorded events in the Event List.

In Angra NPP some locations were analyzed using the Per Cycle Method (CBPC), which were chosen due to the high Cumulative Usage of Fatigue (CUF) presented in the original project. These locations perform fatigue control through the occurrence of one of the transients 1 to 27 listed in Table 3. Figures 16 and 17 shows locations CBPC.

The location shown in figure 21, monitors fatigue usage factor in the pipe weld to check valve in the safety injection line.

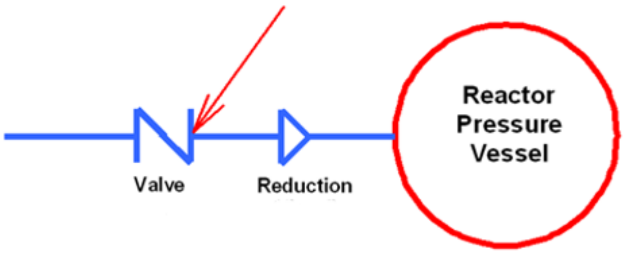


Figure 21: Location SI_VALVE - safety injection line - austenitic stainless steel

The location shown in figure 22, monitors CUF on the pressurizer spray line Tee.

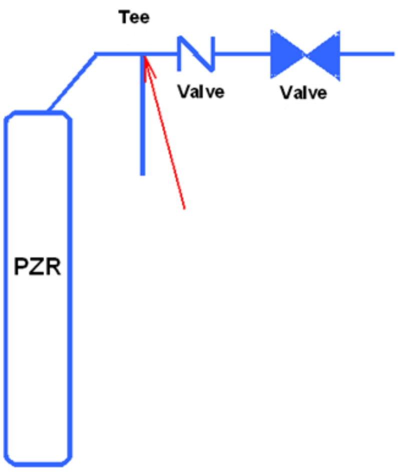


Figure 22: Location PZR_TEE - pressurizer spray line tee - austenitic stainless steel

3.7.2 Event Pairing (CBEP)

For the Event Pairing method, each type of event is associated with a set of transients (that is, stress states) at the monitored location. Then, a pair-usage table is derived, giving the fatigue usage that would result from each possible pair of transients. Note that many transient pairs produce no significant usage, so these pairs are not included in the pair-usage table. To compute the fatigue usage, all events are reduced to their component transients, which are then counted. The transients are paired, highest-usage pairs first, and the usage for that pair is counted. Then the paired transients are removed, and the process continues with the next lower non-empty pair of transients. At the end, any unmatched transients are counted using the highest-usage pair which contains that transient.

In Angra NPP other locations were analyzed using the Event Pairing Method (CBEP) methodology, which increases the factor of accumulated fatigue usage whenever a predefined pair of load cases occurs. Figure 18 shows locations CBEP.

Figure 23 shows two fatigue monitoring locations type CBEP. One is the Residual Heat Removal Tee to the Accumulator Line (nodal point 138) and the other is the Accumulator Line Cold Leg Nozzle (nodal point 212), M. Cisternas [2012].

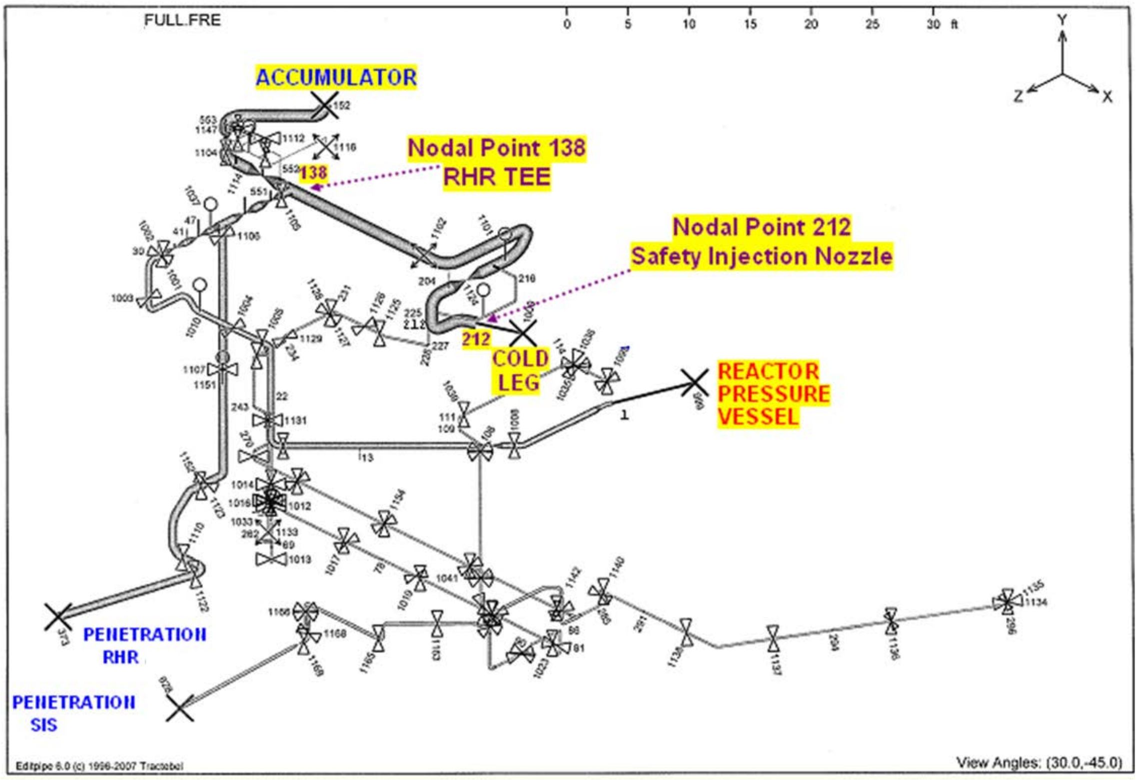


Figure 23: Locations RHRTEE and ACSI - austenitic stainless steel

The location shown in figure 24, monitors fatigue usage factor at both of the Reactor Pressure Vessel outlet nozzles.

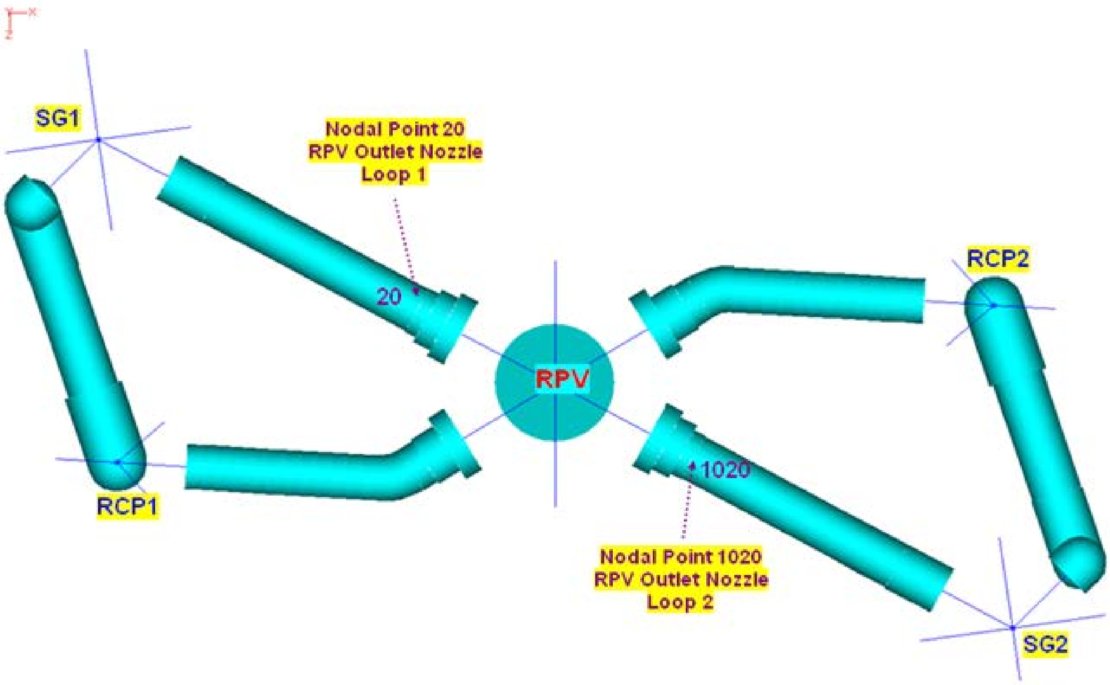


Figure 24: Reactor vessel outlet nozzles - ferritic steels

The location shown in figure 25, monitors fatigue usage factor at both of the Reactor Pressure Vessel inlet nozzles.

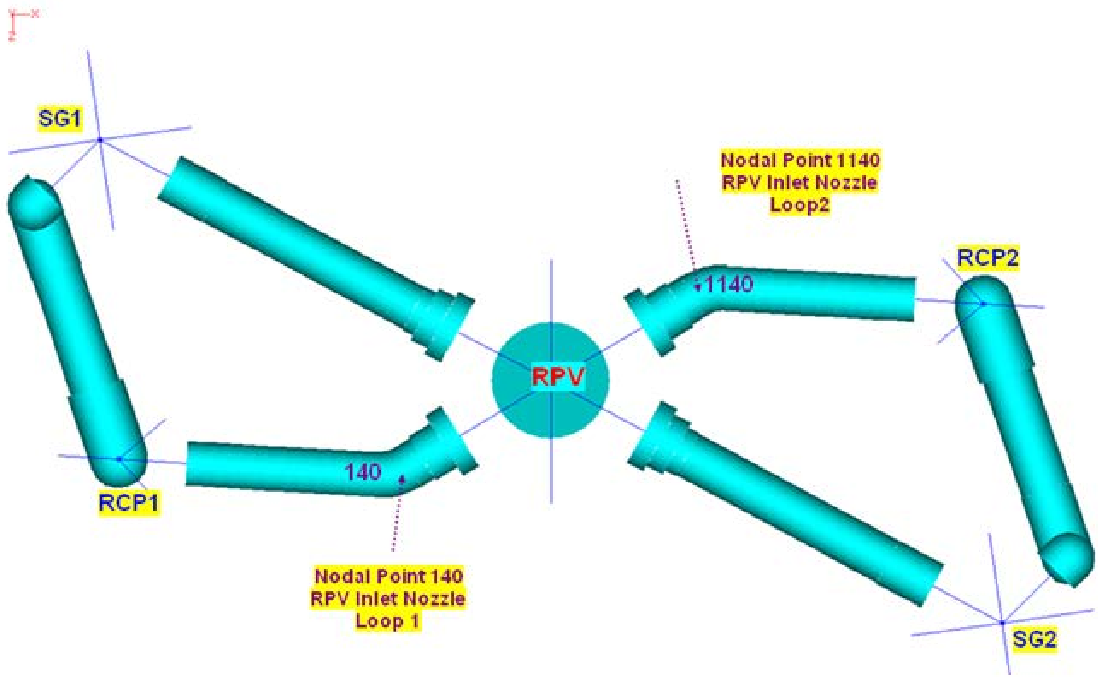


Figure 25: Reactor vessel inlet nozzles - ferritic steels

3.8 Summary of Cumulative Fatigue Usage Factors

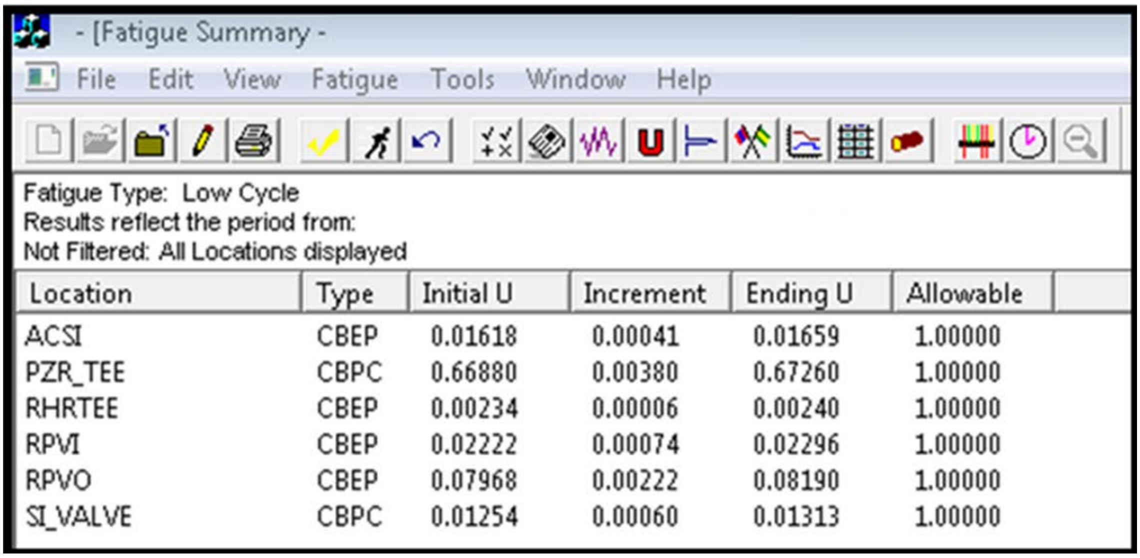
The fatigue monitoring system in nuclear power plant components based on actual plant data are performed in some selected locations. Actual plant transients generally cause stress cycling that is typically less severe than that computed in the original design analysis. The use of a fatigue monitoring system can show that accumulated fatigue usage is much less than would be expected, based on consideration of design transients. The potential economic benefits derived from accurate cumulative usage factor assessment through the use of a fatigue monitoring system can be significant with regard to component life extension. Direct benefit is also provided in the areas of plant operation and maintenance throughout plant operating life.

Cumulative Fatigue Usage (CUF) is the measure of accumulated damage underwent by a component due to cyclic stresses.

The fatigue monitoring system performs an incremental analysis, in that all results are cumulative, reflecting all input files processed since monitoring began. As each new data file is input and processed, the results are updated to reflect the entire input history. This approach imitates the fatigue behavior of plant components, which gradually accumulate fatigue usage as they experience the various events and transients that occur during plant life.

An example with cumulative fatigue usage factor in selected locations, showing the maximum value where redundant locations exist are shown in Table 8.

Table 8: Example of cumulative usage factors.



Location	Type	Initial U	Increment	Ending U	Allowable
ACSI	CBEP	0.01618	0.00041	0.01659	1.00000
PZR_TEE	CBPC	0.66880	0.00380	0.67260	1.00000
RHRTEE	CBEP	0.00234	0.00006	0.00240	1.00000
RPVI	CBEP	0.02222	0.00074	0.02296	1.00000
RPVO	CBEP	0.07968	0.00222	0.08190	1.00000
SI_VALVE	CBPC	0.01254	0.00060	0.01313	1.00000

The Stress and fatigue analysis engineer evaluates the significance of the transient cycle counts. Any transient behavior beyond the design basis may indicate the need to monitor additional locations for fatigue and to modify operations to minimize piping or component fatigue.

3.9 Projection of Cumulative Fatigue Usage (CUF) Location

The design limit on fatigue assumes that the starting point of a crack occurs when the Cumulative Fatigue Usage Factor reaches the value 1.0.

Figure 26 shows the projection of the cumulative fatigue usage factor for 40-year operating life of the plant for location Pressurizer Tee (PZR_TEE) of table 7.

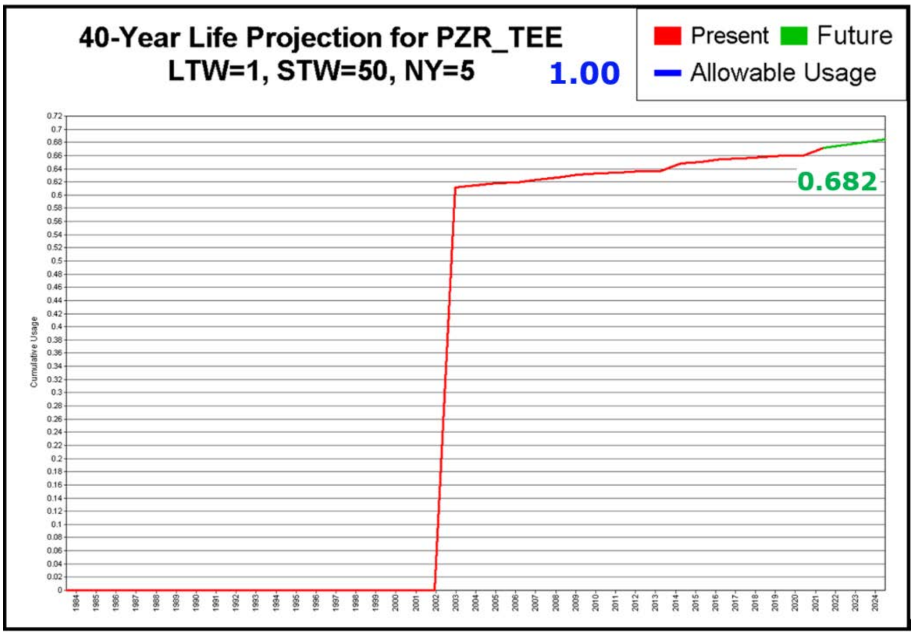


Figure 26: Location pressurizer tee (PZR_TEE) – 40-year life projection

Figure 27 shows the CUF projection for 60 years of operation.

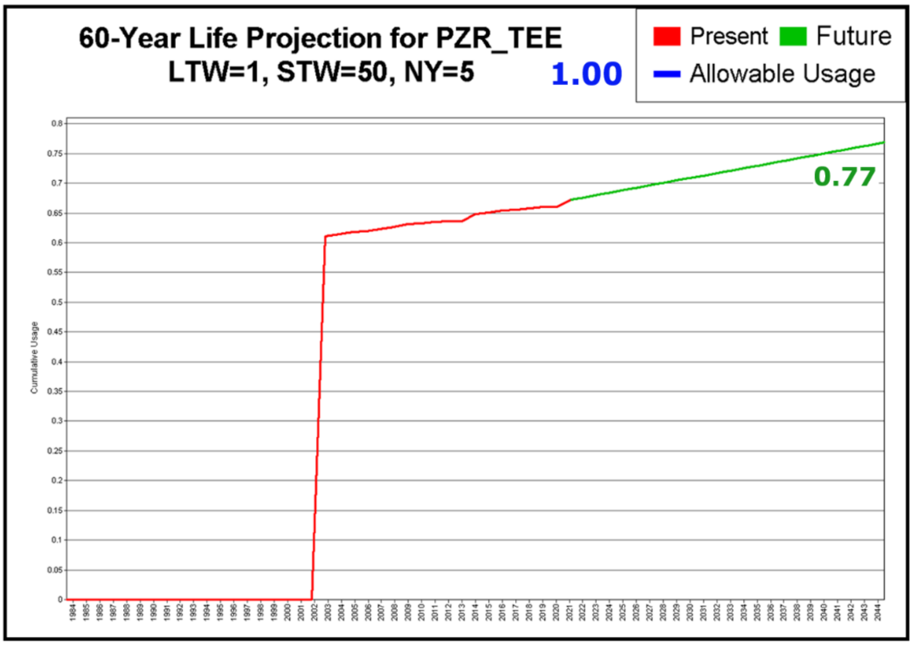


Figure 27: Location pressurizer tee (PZR_TEE) – 60-year life projection

4 Permissible Cycle Limits

4.1 Permissible Cycle Limits for the Planning of 60 Years of Operation

The Angra NPP were designed to operate during a period of 40 years of service (40 Years of Operation). However, the Angra NPP should operate for additional period of 20 years of service. This new period is known as 60-Year Transient Projected Cycles.

A catalog of allowable design limits for 40 years of operation was provided by the NSSS Vendor (Nuclear Steam Supply System). Limits for 60 years of operation can be obtained following the rules of the U.S. Nuclear Regulatory Commission (USNRC), or through the German standards, Nuclear Safety Standards Commission (KTA).

Currently, in the nuclear industry, there is no uniform procedure of rules and techniques to be used to obtain the limits for 60 years of operation.

To cover the possible environmental influence on fatigue resistance, plants that follow USNRC procedures, such as Angra 1 NPP, are subject to the so-called Environmentally Assisted Fatigue (EAF). In this case, there is an environmental penalty arising from tests carried out at the Argonne National Laboratory (ANL), which showed that the effects of fatigue in a water environment, at nominal temperature at full power, are significantly greater than the effects of fatigue in an air environment, in which the current fatigue curves of the ASME code were developed. Note that this penalty reduces fatigue life as it increases the accumulated fatigue usage factor. Basically, it consists of the assumption that the number of cycles allowed in the Reactor is subject to the considered environment. That is, it depends on the operating temperature, oxygen concentration and deformation rate of the materials used. The environmental correction factor (Fen) is defined through a mathematical formula, which has been continuously updated. Selected locations in NUREG/CR-6260 [1995], need to apply the Fen penalty.

For Austenitic materials, NUREG/CR-5704 (Effects of Light Water Reactor Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels) [1999], gives a maximum value for Fen equal to:

$$Fen = 15.3485$$

In general, the value calculated using “ANSYS” finite element systems, after great computational effort using refined models, presents an average value around:

$$Fen = 5.0.$$

For Ferritic materials, NUREG/CR-6583 (Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels) [1998], gives a maximum value for Fen equal to:

$$Fen = 2.4547.$$

The Fen penalty is applied directly to the CUF factor (Cumulative Fatigue Usage Factor), that is:

$$CUF_{60\text{-}YEAR} = F_{en} \times CUF_{40\text{-}YEAR}$$

Alternatively, both NUREG/CR-5704 and NUREG/CR-6583 may be replaced by NUREG/CR-6909 (Effect of Light Water Reactor Coolant Environments on the Fatigue Life of Reactor Materials) [2007], requiring the use of new fatigue curves.

In other words, there is still no consensus on the applied formulations. The fatigue analyzes that follow this procedure are called TLAA analyses, which are analyzes whose validity is limited by the time or cycles of use (Time-Limited Aging Analysis).

On the other hand, for plants that follow German Safety Standards, such as Angra 2, the German KTA standard also presented rules in case the environmental effects cannot be excluded, as in the event projection for 60 years of operation, and are presented Next:

a) KTA 3201.4 [2010], describes the following rules:

In projecting events for 60 years of operation, the number of allowable transients must be reduced until reaching the following limits for the cumulative fatigue usage factor (CUF), which will be considered as fixed values:

CUF = 0.2 for Austenitic stainless steel.

CUF = 0.4 for Ferritic steels (carbon steel and low alloy steels).

b) KTA 3211.2 [2013], describes the following rules:

In projecting events for 60 years of operation, the number of allowable transients must be reduced until reaching the following limits for the cumulative fatigue usage factor (CUF), which will be considered as fixed values:

CUF = 0.4 for Austenitic and Ferritic materials.

The values adopted for CUF as being 0.2 or 0.4 depending on the standard, are called attention threshold, and correspond to 60 years of plant operation.

The definition of attention thresholds reflects the current state of knowledge of environmental influences. This state of knowledge must evolve further in the future, and the decision will be made on the basis of this new basis and at the level of attention thresholds of the KTA standard or, if necessary, its omission will be decided.

Table 9 next presents a summary of the use of rules from NUREG and KTA for transient projections for 60 years of operation:

Table 9: Fatigue correction factor - summary of the use of rules from NUREG and KTA.

NUREG		KTA	
NUREG/CR-5704 (1999-04) Austenitic	NUREG/CR-6583 (1998-02) Ferritic	KTA 3201.4 (2010-11) Austenitic and Ferritic	KTA 3211.2 (2013-11) Austenitic and Ferritic
Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels	Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels	Components of the Reactor Coolant Pressure Boundary of Light Water Reactors Part 4: In-service Inspections and Operational Monitoring	Pressure and Activity Retaining Components of Systems Outside the Primary Circuit Part 2: Design and Analysis
Fatigue Life Correction Factor = Fen CUF 60-YEAR = Fen x CUF		Cumulative Usage Factor = CUF	
Austenitic Fen = 15.3485 (Fen maximum) Fen ~ 5 (Fen calculated)	Ferritic Fen = 2.4547	Austenitic Limit = CUF = 0.2 (Fen = 5)	Austenitic Limit = CUF = 0.4 (Fen = 2.5)
		Ferritic Limit = CUF = 0.4 (Fen = 2.5)	Ferritic Limit = CUF = 0.4 (Fen = 2.5)

Summary for Ferritic materials:

For the accounting of the number of transients occurring in Ferritic materials, the use of attention threshold is very similar if obtained using NUREG rules or the German KTA standard. That is:

Adopting KTA for Ferritic materials implicitly means using Fen = 2.5000

Adopting NUREG for Ferritic materials means using Fen = 2.4547

Summary for Austenitic materials:

For the accounting of the number of transients occurring in Austenitic materials, the use of attention threshold is very similar if obtained using NUREG rules after a long computational effort or the German KTA standard. That is:

Adopting KTA for Austenitic materials implicitly means using Fen = 5.0

Adopting NUREG for Austenitic materials means using Fen ~ = 5.0

(After great computational effort using refined models)

Application of the rules to obtain the CUF for 60-year transient projected cycles

Table 10 shows the cumulative fatigue usage factor (CUF) for 60 years of operation, using the results output shown in table 8.

Table 10: Cumulative fatigue usage factor for 60 years of operation.

System or Component	Location	Material	$CUF_{40\text{-Year}}$ (Table 8)	Environmental Correction Factor F_{en} (Maximum threshold)	$CUF_{60\text{-Year}}$ (Projected)	Warning threshold (60% $CUF_{60\text{-Year}}$)
Accumulator Safety Injection Line	(ACSI)	Austenitic	0.01659	15.3485	0.25463	0,60000
Pressurizer Spray Line	(PZR_TEE)	Austenitic	0.67260	N/A	0.77000[1); 2)]	0,60000
Residual Heat Removal Line	(RHRTEE)	Austenitic	0.00240	15.3485	0.03684	0,60000
Cold Leg Nozzles	(RPVI)	Ferritic	0.02296	2.4550	0.05637	0,60000
Hot Leg Nozzles	(RPVO)	Ferritic	0.08190	2.4550	0.20106	0,60000
Safety Injection Line	(SI_VALVE)	Austenitic	0.01313	N/A	0.02400[1); 2)]	0,60000

1) It is not a selected location defined in NUREG/CR-6260, so it only needs to make projections for 40 and 60 years of operation considering the limit for CUF ≤ 1.0 (Figures 26 and 27).

2) Projection for 60 years of operation performed by a fatigue monitoring system.

Conclusion: In general, the methodology used by NUREG when compared to the methodology used by the German KTA standard to address the environmental-assisted fatigue problem seems to lead to similar values, after great computational effort using refined models.

The use of fixed value penalties (NUREG and KTA for Ferritic materials and KTA for Austenitic materials) can reduce costs and avoid creating very refined models.

The employ of the maximum NUREG threshold value for Austenitic materials, should preferably be used in fatigue control of selected locations, which make use of automatic systems.

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