

# Stress and fatigue analyses of a PWR reactor core barrel components

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*Abstract: The integrity of the nuclear reactor core barrel and its components is one of the most critical issues of the reactor pressure vessel internals due to the unacceptable consequences of their failure and due to the difficulty of their replacement.*

*Especially for such components thermal fatigue is one of the significant long-term degradation mechanisms, due to the fact that thermal loadings lead to most fatigue relevant stresses and strains.*

*In present paper nonlinear structural FE- analyses with subsequent fatigue evaluations are performed in order to predict crack initiation or failure of the reactor core barrel (RCB) components in spite of a failed bolt, revealed in one of the support brackets mounted to the RCB. The elastic-plastic analyses are aimed to determine the cyclic stresses in the support brackets, originated from constrained thermal deformations as well as to compute the cumulative usage factor by performing fatigue analyses according to the ASME Pressure and Vessel Code [2] and the corresponding German Nuclear Safety Standard KTA3204 [5].*

*The elastic-plastic analyses are performed with the help of the finite element program Abaqus in two separate models. Different possible boundary conditions are analyzed and the one yielding the largest stresses is used for further fatigue analysis and computation of the cumulative usage factor.*

*The investigations show, that the absence of the bolt has an influence on the neighbouring support brackets, leading to increase in stresses in those components. However, the stress evaluations according to the German nuclear safety standard KTA3204 [5] show that those increased stress values still lie within the safety limits. The fatigue cumulative usage factor is computed to have the value less than unity, which provides the safety of the reactor core barrel in spite of the damaged bracket.*

*Keywords: Fatigue prediction, cumulative usage factor, reactor barrel, thermal stresses, nonlinearities*

## 1 Introduction

In the Class 1 components of operating nuclear power plants (NPPs) fatigue cycling may occur due to temperature variations. Temperature transients during operation can cause local or global temperature gradients, resulting in thermal cycling at the interface of material and environment. Smooth and sharp temperature transients result in slow or rapid thermal cycling, both being sources for accumulation of fatigue usage. Especially for these components, thermal fatigue is one significant long-term degradation mechanism, due to the fact that thermal loadings lead to most fatigue relevant stresses and strains.

Causes for smooth transients are generally start up and shutdown procedures or load following operation modes.

The nuclear plant design specifications, such as ASME Boiler and pressure Vessel Code [2] or the corresponding German Nuclear Safety Standard 3201.2 [4] identify that the design cycles for fatigue evaluation should be based on a 40-year life expectancy of the plant. This method ensures freedom from fatigue cracking during the mentioned period. Using the code fatigue curves, the cumulative usage factor must be computed based on estimated number of cycles for the postulated period, and has to be equal or less than unity.

In present paper fatigue assessment is made in order to predict crack initiation or failure of the mounting brackets, attached to the reactor core barrel (RCB) in the pressurized water reactor of an NPP.

Four irradiation channels are mounted to the outer wall of the mentioned RCB and each of them is clamped to the RCB with the help of 15 support brackets.

These brackets are mounted to the RCB with cheese head bolts and alignment pins.

Recently, during the inspection of the RCB with underwater camera, damage in a support bracket of one of the irradiation channels was discovered. The inspection revealed that one of the bolts of the support bracket No.9 was broken and the spacer was missing.

Several studies following the inspection proved that the failure cause for the affected support bracket could not be the flow-induced vibrations [1]. Hence, further investigations were needed to study whether the cause of the bolt failure was fatigue induced damage and whether the lack of the mentioned bolt would affect the other support brackets and safety of the core barrel during the further operation of the NPP.

In present paper detailed nonlinear structural analyses with subsequent fatigue evaluations are performed in order to predict failure of the reactor core barrel taking into account the failed bolt of the support bracket in the NPP. The evaluations are performed by determining the cyclic stresses appearing in the support brackets originated from constrained thermal deformations as well as the cumulative usage factor. The cumulative usage factor is determined by performing fatigue analyses according to the ASME Pressure and Vessel Code [2] and the corresponding German Nuclear Safety Standard KTA 3201.2, section 7.8 [4].

The study is performed with the help of finite element method (FEM) with the Software Abaqus [6].

The progressive increase in computational resources and the recent extensive FEM development has enabled the use of these codes in the mechanical calculation of class 1 components such as the "Design by Analysis" in nuclear calculation codes due to the high level of accuracy and reliability of these methods.

The effects of the thermal expansion of the RCB on the support brackets mounted to its outer wall were examined in two separate FE models by means of nonlinear elastic-plastic structural analyses. Based on results of the structural analyses the cyclic stresses were evaluated with the help of fatigue analyses according to the above-mentioned standard.

## 2 Structural analysis of the RCB subjected to thermal loadings

The first step in the evaluation of the support brackets integrity is the assessment of thermal strains and stresses of the core barrel during each transient.

The reactor core barrel, the major structural member among the reactor internals, is a cylinder including an external ring flange at the top end and an internal ring flange at the lower end.

Due to  $\gamma$  radiation the RCB is heated up to 340°C. The heat, generated in the reactor is released to the primary coolant circuit, which consists of 4 identical parallel connected circuits. Water as a coolant is pumped through the reactor to carry away the heat. The coolant enters the reactor pressure vessel (RPV) at the temperature of 291,3 °C via 4 coolant inlet nozzles, flows downward through the downcomer space between core barrel and pressure vessel wall (outer wall of the RCB), is rotated through 180° and re-routed upwards through the reactor core, meanwhile warmed up to 328,3°C. The temperature of the surrounding coolant is variable from point to point causing inhomogeneous temperature distribution along the boundary surface of the core barrel.

During the loading and unloading cycles the temperature increases from 23 °C to the actual temperatures on the inner and outer surfaces of the core barrel and cause thermal expansion. This inhomogeneous expansion leads to cyclic thermal stresses and strains arising in the mounting regions of the support brackets of the irradiation channels, attached to the RCB.

The variable temperature values along the surface of the RCB are measured experimentally and consist of totally 2304 regions [1]. The temperature values are measured on both: the inner and outer walls of the RCB.

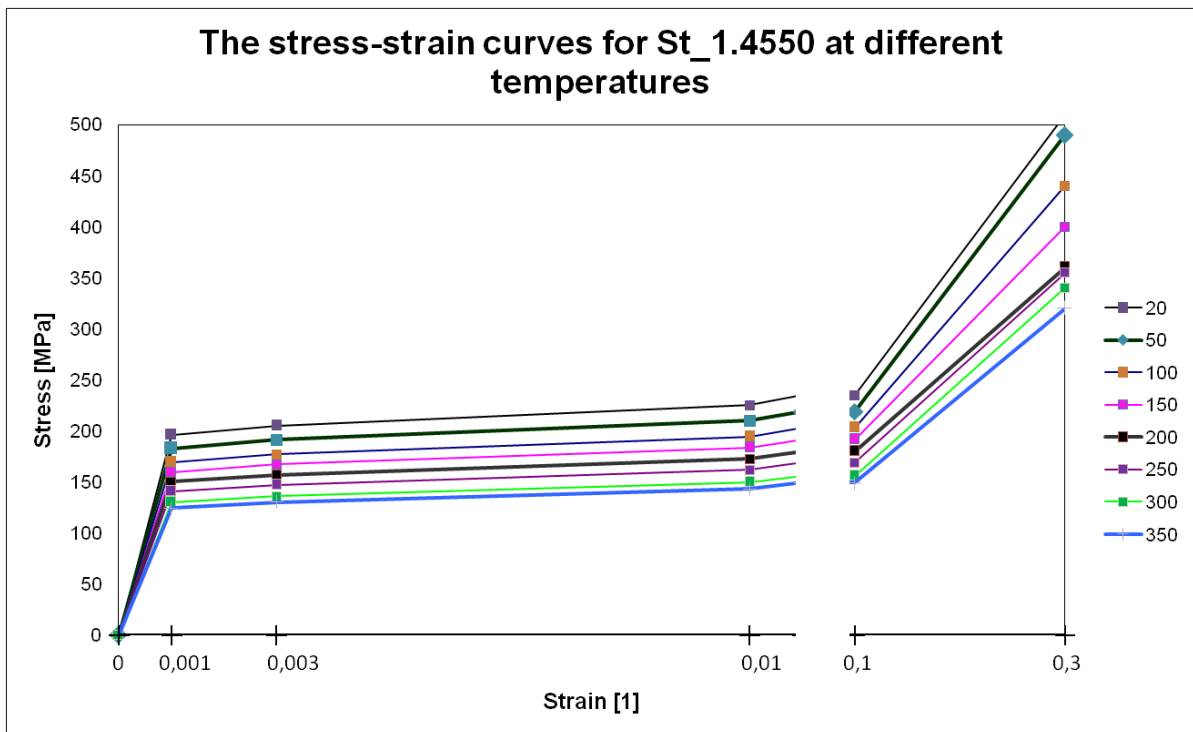
In order to obtain the thermal stresses and strains, at first the FEM model of the RCB is constructed in Abaqus. In this FEM model the RCB is examined under thermal expansion taking into account the measured inhomogeneous temperature distribution along the boundary, its inhomogeneous distribution through the thickness of the RCB wall as well as its linear increase and decrease in time during a loading and unloading cycle respectively.

The finite element model represents one loading and unloading cycle of the RCB. Due to the cosine shape temperature distribution along the inner and outer walls of the cylindrical barrel only one eighth model of it is considered. The barrel is modeled with the help of 2D shell elements of the type S4R.

The temperature distributions are applied along the wall of the barrel at the extent positions given in [1], modeling 297 different temperature regions for the one eighth of the model.

Also the variation of the temperature across the wall thickness has been taken into account in the model. For that purpose 2 different temperature points across the thickness have been defined in the model, at which the measured inner and outer temperature values have been prescribed. The variation has been taken to be linear between these points

The shell of the core barrel is made of stainless steel type 1.4550, for which multi-linear elastic-plastic behavior with kinematic hardening was taken. The temperature-dependent stress-strain curves are shown in Figure 1.

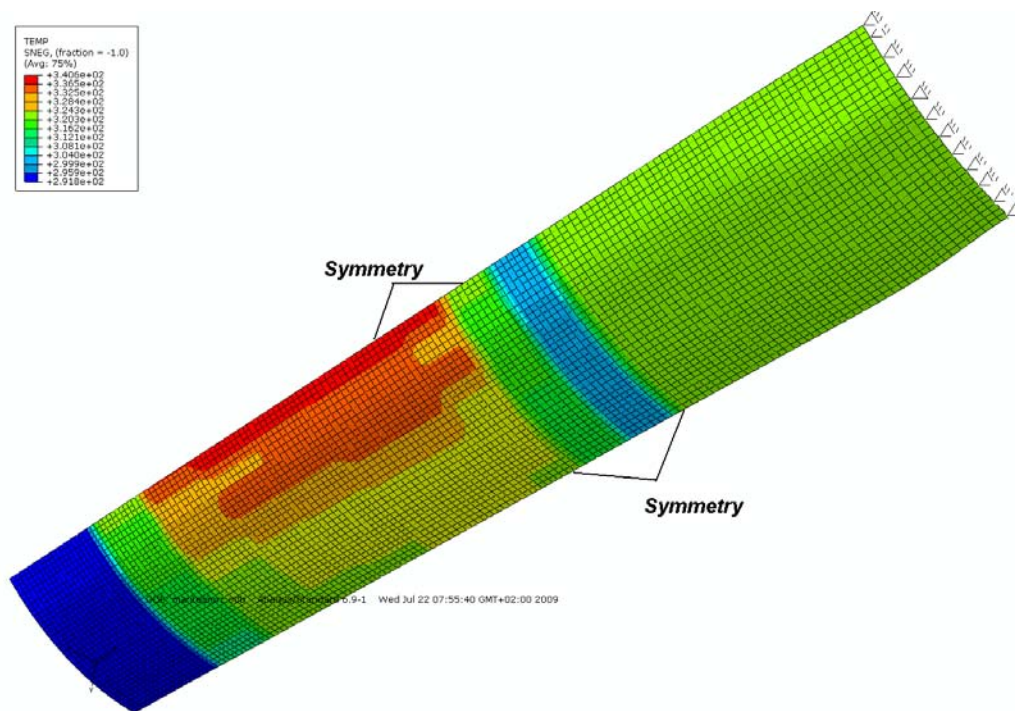


**Figure 1. The temperature-dependent stress-strain curves for stainless steel type 1.4550**

The mentioned curves represent the true stress versus true strain curves, which have been obtained from the experimental curves with the following formulae:

$$\varepsilon = \ln(1 + \varepsilon_e) \quad \sigma = \sigma_e(1 + \varepsilon_e)$$

From the elastic-plastic analysis of the shell with the account of the geometric nonlinearities the thermal stresses and strains during the loading-unloading cycle are computed at the nodes where the bolts of the support brackets are mounted. The temperature distribution along the outer wall of the shell is shown in Figure 2.



**Figure 2. Temperature along the outer wall of the barrel shell.**

The calculation results reveal the expected increase in axial displacement downwards the barrel. The obtained different thermal strains at different mounting brackets may cause fatigue damage and their influence is studied further in the second FEM model.

The second FEM model represents the model of the irradiation channel together with its support brackets (see Figure 4), with the help of which the channel is attached to the RCB.

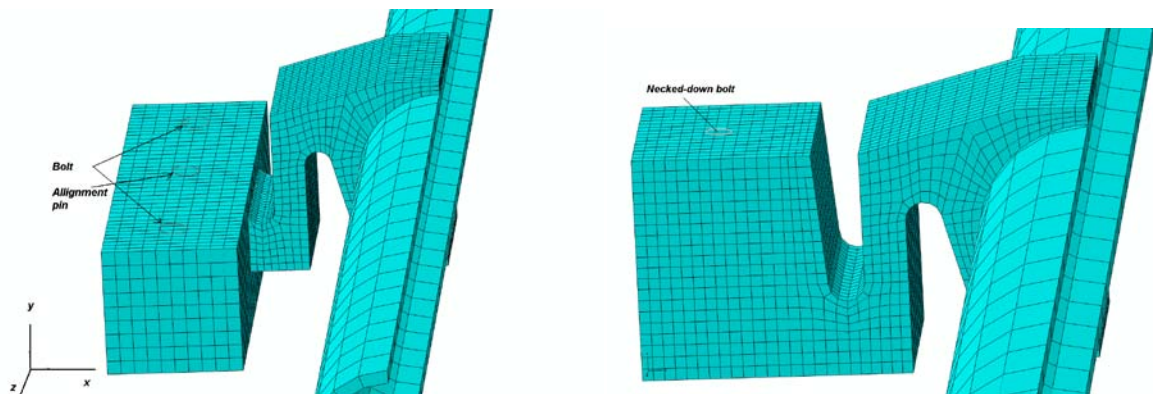
The displacements at the attachment points of the mounting brackets which are obtained from the first FEM model of the RCB, subjected to thermal loads are used further to model the influence of the RCB on the support brackets of the irradiation channel.

### 3 The irradiation channel with 15 support brackets

The irradiation channel essentially consists of a long pipe with 15 support brackets, which are mounted to the shell of the RCB. This is done with the help of cheese head bolts and alignment pins that are screwed into the holes tapped into the RCB shell. The bolts are preloaded and made of Type 1.4571 stainless steel.

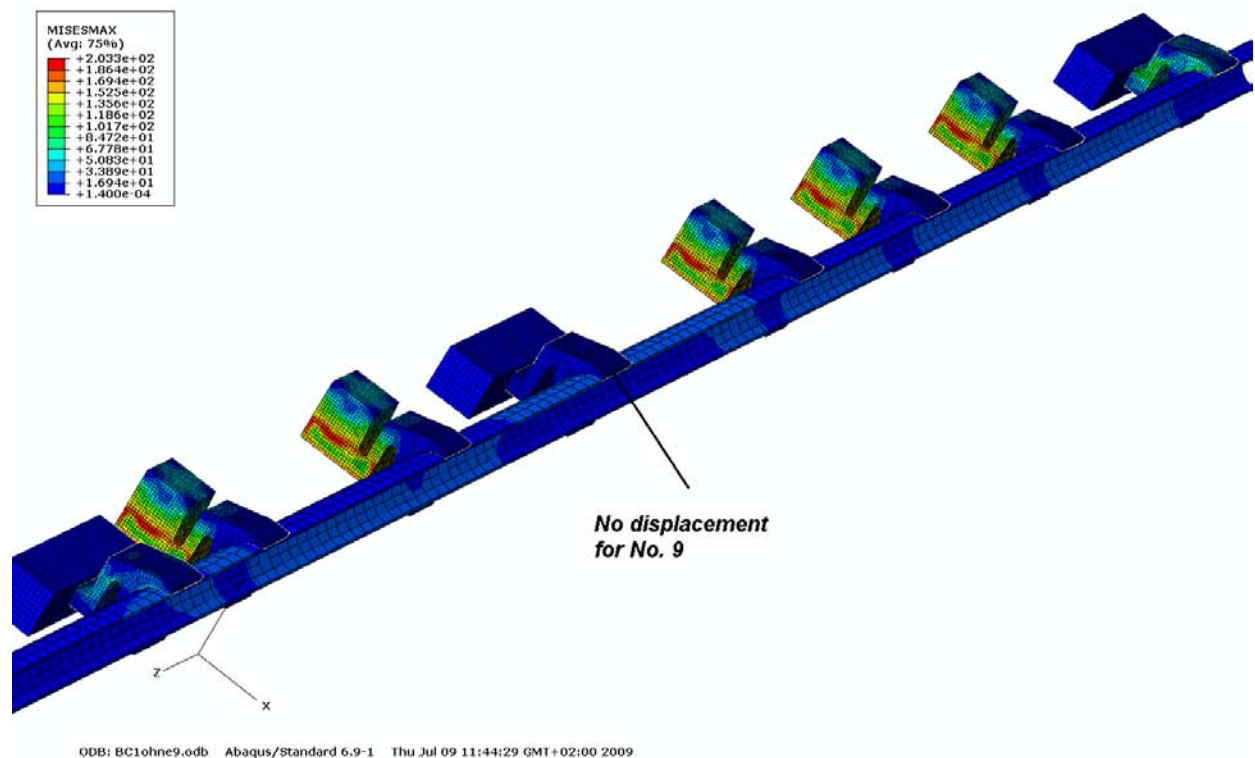
Depending upon geometry of the support brackets two types of mountings are used: loose point (with 2 hold-down bolts) and fixed point (with 4 bolts and 2 alignment pins) mountings. The alignment pins in the fixed point mounting brackets serve for the positive locking of the bolts and the shell.

The bracket, where the broken bolt has been found, represents a loose-point mounting bracket and is the ninth one, counting from the top. The loose-point and fixed-point brackets with different geometries are shown in Figure 3.



**Figure 3. Two types of support brackets, a) fixed-point, b) loose-point**

For determination of the fatigue level of the support brackets, several elastic-plastic nonlinear structural analyses with subsequent fatigue analyses are performed according to the ASME Boiler and Pressure Vessel Code [2] and the corresponding German Nuclear Standard KTA 3204 [5].



**Figure 4. Mises - stresses in the model without boundary condition on the bracket Nr. 9**

For that purpose the 3D- model of the irradiation channel together with its 15 support brackets is constructed in Abaqus. The support brackets as well as the irradiation channel are modeled with solid elements of type C3D8R. Surface-to-surface tied contact is defined between the brackets and the channel.

Due to the symmetry of the geometry and the loading, it is sufficient to model just the half of the channel in the FEM (see in Figure 4).

The material (stainless steel of Type 1.4550) of the support brackets is modeled to have multi-linear elastic-plastic behavior with kinematic hardening. Similarly, as in the above-discussed thermal analysis of the RCB shell, true elastic-plastic stress-strain curves are used to model the behavior of the material. The irradiation channel is assumed to be elastic.

The thermal displacements, determined in the first model, which are transferred to the bolts and the alignment pins, are taken into account in the second model. They are applied linearly in the temperature interval from 20°C to 291.3°C, the latter representing the temperature of the cold loop. One close loading-unloading cycle is modeled.

In order to reveal the influence of the broken hold-down bolt on the neighbouring ones and also on the RCB, 8 different boundary conditions are analyzed and for each of them an elastic-plastic analysis is performed considering a close loading-unloading cycle. The considered four boundary conditions cover all the possible cases when the bolts still hold in different directions.

In the considered boundary conditions the alignment pins of the fix-point bolts are fixed in 'x' and 'y' directions, while being free to move along the 'y' direction. For the bolts of the fix-point brackets as well as for the hold-down bolts of the loose-point bracket four cases are considered. In BC1 the bolts are loose in both directions, BC2: the bolts are loose in 'y' direction, BC3: they are loose in 'x' direction, and BC4: they hold in both directions. These boundary conditions with the account of the damaged bracket are given in Table 1.

**Table 1. The analyzed boundary conditions**

| Boundary Conditions                      | BC1        | BC2        | BC3        | BC4             |
|------------------------------------------|------------|------------|------------|-----------------|
| Alignment pins of the fix-point brackets | $u_x, u_z$ | $u_x, u_z$ | $u_x, u_z$ | $u_x, u_z$      |
| Bolts                                    | $u_y$      | $u_x, u_y$ | $u_y, u_z$ | $u_x, u_y, u_z$ |

The other four boundary conditions are similar to those, given above. The only difference is that they are not valid for the defective support bracket Nr. 9, simulating the case of missing bolts for this bracket.

The finite element analyses show that at different boundary conditions increase in stresses and strains occur when the defective bracket is not mounted to the RCB shell. Especially the brackets, neighboring the defective one are affected and show larger stress values ( increase with 1%) as compared with the case when the bolts of the defective bracket still hold.

For most boundary conditions the maximal strain values are reached in the fourteenth support bracket at boundary condition 1 with missing bolts for bracket No.9

The computed cyclic strains from the FEM model in the worst case are used for the subsequent fatigue analyses. In Figure 5 the maximum principal logarithmic strain in the worst case is shown, for the case when the defective bracket is missing, yielding larger strains in the neighboring ones.

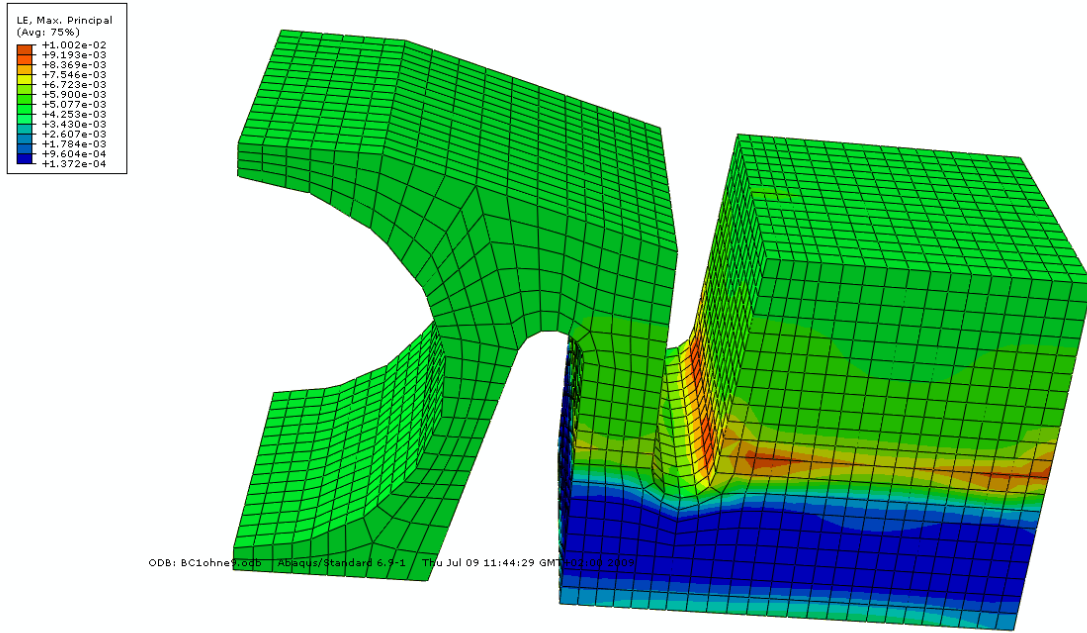


Figure 5. The maximum principal logarithmic strain in support bracket Nr.14

#### 4 Fatigue analyses

In order to evaluate the long-time fatigue life of the core barrel and the irradiation channel with its support brackets, the accumulated fatigue damage is calculated according to ASME standard [4].

If the material is subjected to  $m$  different cycles of frequency  $n_i$  and corresponding to stress ranges  $S_i$  ( $i = 1, 2, \dots, m$ ), then the cumulative usage factor  $D$ , is given by

$$D = \frac{n_1}{\hat{n}_1} + \frac{n_2}{\hat{n}_2} + \dots \frac{n_i}{\hat{n}_i} \leq 1$$

where  $D$  is the cumulative usage factor;  $\hat{n}_i$  - the number of design allowable fatigue cycles at a given total equivalent strain range and temperature,  $n_i$  - the specified number of cycles.

Safety from fatigue requires:  $D \leq 1$ .

The fatigue analysis is based on the determination of the half of the equivalent strain range  $\epsilon_a$  which is defined with respect to changes in the strain components from the starting point to each point  $i$  in the cycle.

From the amount of the considered loading conditions two appropriate ones are chosen in such a way, that the equivalent strain computed from the difference of the corresponding strains, becomes maximum. This maximum represents the range of the equivalent strain (KTA 3201 Section 7.7.3.3 [8]):

$$\varepsilon_a = \frac{\sqrt{2}}{2(1+\nu)} \left[ (\Delta\varepsilon_x - \Delta\varepsilon_y)^2 + (\Delta\varepsilon_y - \Delta\varepsilon_z)^2 + (\Delta\varepsilon_z - \Delta\varepsilon_x)^2 + \frac{3}{2} (\Delta\gamma_{xy}^2 + \Delta\gamma_{yz}^2 + \Delta\gamma_{zx}^2) \right]^{\frac{1}{2}}$$

In our case the Poisson's ratio is assumed to be  $\nu = 0,3$  also in the plastic region.

The half of the effective or equivalent stress range  $S_a$  is obtained by multiplying the equivalent strain range with the elasticity modulus  $E$ :

$$S_a = 0,5 E \varepsilon_a$$

Whereas the elasticity modulus is taken the one for the temperature  $t = 300^\circ\text{C}$ .

The occurring strains in the bracket No.14, which are calculated with the help of elastic-plastic nonlinear analyses, are used in the latter fatigue analysis study. In this bracket the stresses and strains reach their highest values.

For each loading-unloading cycle with the account of the equilibrium temperature, the following ranges of the equivalent strains have been calculated for different boundary conditions (see Table 2).

In the present study the largest value of the equivalent strain range is computed to be at the support bracket Nr. 14 and occurs at the boundary conditions BC1 with missing bolts for the support bracket Nr.9. It has the value 0.0091.

The design fatigue curve for the material 1.4550 is obtained from [5] and the number of the admissible load alternations is determined. In the considered case the maximum equivalent strain value corresponds to an admissible load alternation of 791.

**Table 2. The equivalent strain range and the number of allowable load cycles for different boundary conditions**

| Boundary condition     | $\varepsilon_a$<br>[1] | $S_a$<br>[MPa] | $\hat{n}$<br>[1] | D per<br>Cycle *E-03 |
|------------------------|------------------------|----------------|------------------|----------------------|
| BC1                    | 0,0098                 | 809,41         | 794              | 1,26                 |
| BC2                    | 0,0090                 | 804,16         | 810              | 1,23                 |
| BC3                    | 0,0083                 | 742,18         | 1046             | 0,96                 |
| BC4                    | 0,0079                 | 708,96         | 1223             | 0,82                 |
| BC1 without s. b. Nr.9 | 0,0110                 | 810,53         | 791              | 1,26                 |
| BC2 without s. b. Nr.9 | 0,0090                 | 803,87         | 811              | 1,23                 |
| BC3 without s. b. Nr.9 | 0,0083                 | 742,62         | 1044             | 0,96                 |
| BC4 without s. b. Nr.9 | 0,0079                 | 710,03         | 1217             | 0,82                 |

For the mentioned boundary condition, which appears to be the most critical concerning the number of load alternations among those 8, further fatigue analysis is performed and the cumulative usage factor is computed (See Table 2).

The fatigue cumulative usage factors are computed for the time periods of warming up and down of the core barrel, during operation period from 1985 to 2007 and extrapolated for the period 2007-2025. During the warming up of the RCB 10 loading and unloading cycles are registered. The given 39 loading cycles in the period of 1985-2007 are also considered for the usage factor calculation. Furthermore, these loading cycles are extrapolated for further 17 years and the conservative total fatigue value is computed to have the value: 0.1 which does not appear to be

critical neither for the shell nor for the support brackets. Hence, the failure of the damaged support bracket is not expected.

## 5 Evaluation of the bolts and the alignment pins

Besides the support brackets, also the bolts and the pins require a detailed stress and fatigue evaluation which is out of the scope of the present study.

However, for different boundary conditions for the support brackets of the irradiation channel also the tensile and shear forces were evaluated at the nodes of the bolts and the alignment pins.

The consideration of stresses in the bolts is that thermal loads applied to the bolts are classified as primary loads. The reason for this classification is because these loads, although secondary in the containment, are considered to be applied non-self-limiting loads for the bolts. Hence, the stresses in the bolts are compared to the allowable stress  $S_m$

For the stainless steel Type 1.4571, from which the bolts are fabricated, the allowable stress is  $S_{m,300} = 131,8 \text{ MPa}$  at 300 °C.

The maximum shear stress for the alignment pins is computed for the boundary condition BC2 without support bracket Nr9 and has the value of 39,7 MPa. This value lies below the allowable stress, since

$$\tau_9 = \sqrt{3} \tau_{\max} = 68 \text{ MPa} \leq S_{m,300}$$

The computed tensile stress in the bolts reaches its maximum at the boundary condition 2 and has the value of  $\sigma_z = 30,3 \text{ MPa}$  and in the hold-down bolts:  $\sigma_z = 22,2 \text{ MPa}$ , which still lie within the allowable limits.

The FEM analyses show that the forces in the neighboring brackets increase for the studied cases, when the bolts at bracket are missing. However, the computations show that failure of further bolts and pins is not expected.

## 6 Conclusions

Degradation of core internal structures usually involves increases in clearances due to wear or misalignment, or loss of clamping forces. While these changes are inherently small, significant degradation can occur through thermal fatigue of large structures. This type of degradation may be difficult to detect visually before the structure's function or integrity are compromised.

The elastic-plastic fatigue analyses of the irradiation channel together with its 15 support brackets show that the fatigue induced stresses due to their low values will have no influence on the safe operation of the NPP. No damage of the core barrel shell and its remaining support brackets due to fatigue and thermal ratcheting or failure of the defective bracket itself are expected. The absence of the bolt has an influence on the neighbouring support brackets, leading to increase in stresses in those components. However, the stress evaluations according to the German nuclear safety standard KTA3204 [5] show that those increased stress values lie still within the safety limits.

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