Three-dimensional Leak before Break Analysis of Nuclear Pipes Containing Cracks

Wenfu Liang[a*], Tong Liu[b] and Minshan Liu[c]

Thermal energy engineering research center of Zhengzhou University, Zhengzhou 450002, China

[a*]lwf314159@163.com, [b]liutung@zzu.edu.cn, [c]msliu@zzu.edu.cn

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Abstract. Three-dimensional crack behavior simulation analysis and anti-fracture design have been a main subject in fracture theory and engineering application. Piping system is a key part of nuclear power engineering. Utilizing the three-dimensional finite element analysis software ANSYS and the specialized crack analysis programs Franc3D, three-dimensional crack behavior and leak before break (LBB) case were simulated and evaluated of a pipe with a crack in waste heat exhaust system of China Experimental Fast Reactor (CEFR). In fast reactor, the piping is working under a high temperature. Therefore, the code RCC-MR.A16 was adopted that is suitable for materials and structural safety design at high temperature. Material used in this article is modified 9Cr1Mo-T91/P91. The analysis model of pipe section was built in three-dimensional entity structure containing a cracks and the high temperature and creep effects were considered. The simulation results show that creep contributes more effect on crack growth than fatigue. The evaluation results on LBB of studied T91 steel pipe with a crack-like defect can satisfy the need of LBB design guidelines. The research results can be referenced in pipe material choose, safety assessment and structural integrity evaluation of a pipe containing defects at high temperature in a fast reactor design.

Introduction

Along with the economic development and raise awareness of environmental protection, the development and utilization of nuclear power and other clean energy have been the main line. With low operating costs, in our country nuclear power has entered a period of rapid development[1]. Operational safety of nuclear power has always been focused on in the realm of construction and development of nuclear power.

China experimental fast reactor is a fourth-generation nuclear energy systems of reactor types. The utilization of uranium resource in fast reactor has been to 60%-70%, which have strategic implications for sustainable development of nuclear power. The accident heat emission system in a fast reactor uses passive system of directly cooling sodium that in the main container, the safety and reliability of this system directly affect the performance of the main container in a fast reactor.To ensure the safety of fast reactor heat emission system, the design concept of leak before break (LBB) was applied to assess the engineering systems. Meanwhile the structural optimization referring to LBB has great significance in practice was also performed. There are more study on LBB assessment in China and abroad in recent years[2,3,4].FENG Xi-qiao and DONG Bi-bo had made LBB analysis on some pipes of nuclear power; The building sodium-cooled Fast Reactor (SFR) in North Korea has applied LBB technologies to the design of thin - walled pipes[5].

Analysis of LBB Assessment Technology

Analysis of LBB assessment process. The purpose of LBB analysis can be used to test the leak amount of coolant that leak from pipe with through-wall crack in quantity. Analysis of crack model generally is based on the stability premise, and consider that the double shear fracture will not occur in pipelines. Therefore, the steady expansion, the amount of crack propagation and calculation of crack size are the main work of LBB analysis. In the design and construction of nuclear reactors, LBB analysis is mainly used in primary loop and auxiliary piping, such as wave pipe, high pressure gas

container and export of waste heat pipelines.The LBB assessment flow can be constructed as in Fig. 1 [6].

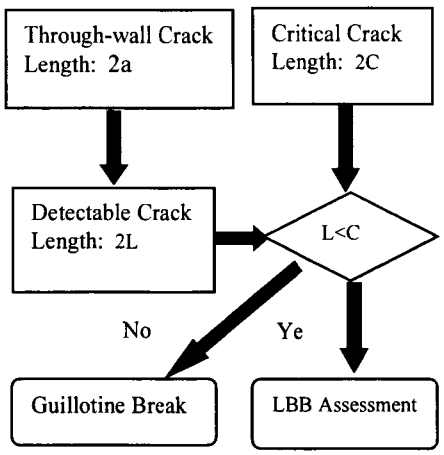

Fig. 1 Flow chart on LBB assessment

Rules of LBB Assessment. When the pipeline with crack was selected,the LBB assessment can be implemented according to steps as follows[7].

(1) Determine load and stress of structure without crack: determine a variety of loading conditions, including internal pressure, axial load, bending moment, heat stress, and so on. Calculate the relevant stress distribution.
(2) Access to the material properties: LBB assessment will use material properties and variation curve of properties. For example: yield limit, stress-strain curves, resistance curve, and so on.
(3) Identification and determination of crack location, shape and sizes: in straight pipe, both circumferential and axial cracks will be considered. As circumferential crack is more dangerous, so some research generally focus on it.
(4) Analysis of leak detection systems: including leak detection methods and sensitivity. Analysis of leak detection system should be based on the experiment and measured data of reactor, in addition, considering the impact of surface roughness, crack opening area, two-phase flow and other factors.
(5) Determine the minimum crack size that can be detected by the leak detection system: typically 10 times the safety margin is chosen.
(6) Determine the extension of the critical crack size at room temperature.
(7) Calculate time that is from the start of leakage to instability of crack.
(8) Stability analysis of the overall: whether the component has plastic instability and failure occurs.
(9) Local stability analysis : determine whether fracture of brittle failure occurs.
(10) Comprehensive evaluation on the structure, and determine whether LBB conditions are met.

Analysis model of nuclear grade pipeline with cracks

Creep-fatigue crack growth of Model with crack. Through structural analysis of the pipeline system, this study choose the most serious part of the fast reactor heat emission system[8].The crack is wall-penetrated and fan-shaped crack, with its sizes changing from 10 to 80 mm along the internal wall. Generally, a fast reactor used liquid sodium as coolant, in which case the tube temperature has been exceeded the pipe creep temperature. So the French instruction A16 standard considering both structural fatigue and creep is adopted here.

In A16, the crack growth of creep-fatigue was calculated using the following equation:

$$a = \mathrm{a}_0 + (\delta a_{fa})_i + (\delta a_{fl})_i \tag{1}$$

where a_0 is half length of initial crack.The crack growth of fatigue was calculated by the following e quation:

$$(\delta a_{fa})_i = C[(\Delta K_{eff})_i]^n \tag{2}$$

where C, n are material constants of P91 steel.ΔK_{eff} is effective stress intensity factor (SIF) range; it can be got by the range of variation ΔJ that is calculated using J-integral of operating model cycle. According to A16, C=3.36×10^{-11}, n=2.63.

For the crack growth of creep:

$$(\delta a_{fl})_i = \int_{t_i}^{t_i+t_{ni}} A(C_i^*)^q dt \tag{3}$$

where A,q are material constants. And C^* is C-integral of crack in steady creep. For conservative, this study set the maximum temperature as 550 degrees centigrade. And we assume that each extension is homeostasis in the operating model cycle, so C^* is constant value. The reduced form is

$$(\delta a_{fl}) = A(C^*)^q t_c \tag{4}$$

where t_c is the time that spent in each power cycle. It is 14.6h/time under condition.

Parameter analysis. The pipeline is a jacketed structure. A simplified approach was applied when we make an analysis. The streamlined pipeline parameters are listed below: outside diameter is 108mm, wall thickness is 4.5mm. The model adopted modified 9Cr1Mo--T91/P91 martensitic stainless steel has already been widely used in a fast reactor at home and abroad. T91 has good mechanical properties. A thinner wall can be designed to increase the heat transfer efficiency and reduce the construction of material, so it has good economic benefits[10]. The LBB analysis of modified 9Cr1Mo material have not seen in report[11]. The density, elastic modulus and Poisson's ratio of T91 are 7770kg/m^3, 2.18E+11Pa and 0.31.

A faster reactor has four operating model: behalf of heap, shutdown, some power and envelope. The parameters of operating model are shown in table 1. In addition, the loads of envelope condition are sum of the load and safe shutdown earthquake (SSE) load under normal operating condition.

Table 1. Parameters in different operating condition.

load \ operating	Behalf of heap	Shutdown	Some power	Envelope
Temperature [℃]	516	250	396	516
Inner pressure [MPa]	0.46	0.4	0.45	0.6
Twisting moment [N·m]	1075.25	239.9	454.497	7524.11
Bending moment [N·m]	2726	477.5	724.8	3127

LBB analysis of Nuclear grade pipeline with cracks

Mechanical analysis of the piping model. After determining associated structure parameters of the mechanical and operating model, the appropriate finite element models were built using ANSYS software considering statics structural analysis and thermal analysis. According to Table 1, we use access technology to apply torsional and bending moment in the course of analysis, and use indirect method for the thermodynamic analysis of the model. In the end, the crack was embed in the model using Franc3D software.

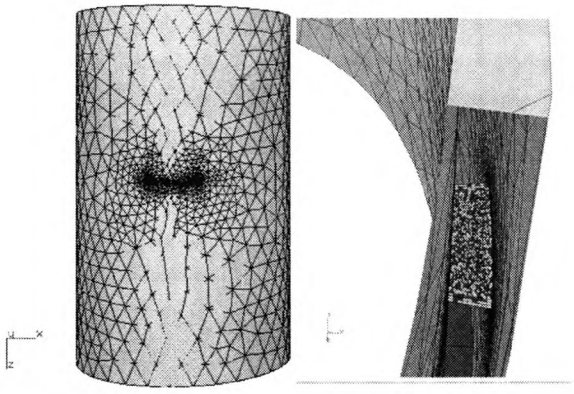

Fig. 2 The finite element model of a pipe with a crack

The model was selected the worst part of the piping system.Using trapezoidal parameters, we determine the length of crack that is on the inner wall. After the introduction of divided grids, overall diagram and a cross-sectional view of the model are shown in Fig. 2.

Analysis on fracture mechanics parameters. For crack propagation of fatigue and creep, Js and C* are needed in analysis. In A16, they can be gtt by the J-integral of crack tip calculated by Franc3D. According to the rules, the J-integral of model under mechanical load and thermal load were calculated. Finally, we can get relation of J_s-a and C^*-a as follow:

$$J_{S1} = -27872.832 + 6.918\times10^6 a - 4.097\times10^8 a^2 + 9.061\times10^9 a^3 \tag{5}$$

$$J_{S2} = 126.446 + 1.994\times10^5 a - 1.060\times10^7 a^2 + 3.523\times10^8 a^3 \tag{6}$$

$$J_{S3} = -3739.2238 + 1.112\times10^6 a - 6.291\times10^7 a^2 + 1.473\times10^9 a^3 \tag{7}$$

$$J_{S4} = -246734 + 9.534\times10^7 a - 9.394\times10^9 a^2 + 3.409\times10^{11} a^3 \tag{8}$$

$$C^* = -27852 + 6.78\times10^6 a - 3.9\times10^8 a^2 + 9.06\times10^9 a^3 \tag{9}$$

where a is the half length of the through-wall crack. Js and C^* is parameter of fracture.

LBB assessment. According to the formula (5) to (9) and using the formula:

$$\Delta J_{S1} = J_{S1} - J_{S2} \tag{10}$$

$$\Delta J_{S2} = J_{S4} - J_{S3} \tag{11}$$

the parameter of start-stop cycle and power cycle can be obtained. And then using programming, the half of crack propagation at the end of life is 12.24mm. But in considering of creep-fatigue cycle, the final half of crack propagation is 23.65mm. So,creep has a great effects on crack growth[12].

Figure 3 was plotted according to J_R resistance curve of the material and dJ_S-da curve got form the four operating model. In Fig.3, the J-integral value of crack tip accelerates with the crack length increases, and it demonstrates that the critical crack size of the pipe is 53.45mm.

When the size of final crack is 23.65mm, according to formula (9), we got appropriate J_S value is J_S=130.85KN/m. And the J_R value is J_R=2024.92 KN/m. Due to $\sqrt{2}\times J_S < J_R$, so the cracks meet the norms of LBB in stability.

On the measurement of the sodium leakage, the sensitivity of sodium concentration that sodium aerosol detector can be detected is 10 ppb[13]. In A16 instructions, sodium leak generally choose the safety factor S_t=10. Based on a conservative estimate, when the size of through-wall crack is 18 mm,

we can detect leakage[8]. According to A16 provision,the length of critical crack must be twice greater than detectability crack. Therefore, LBB assessment of leakage are satisfied.

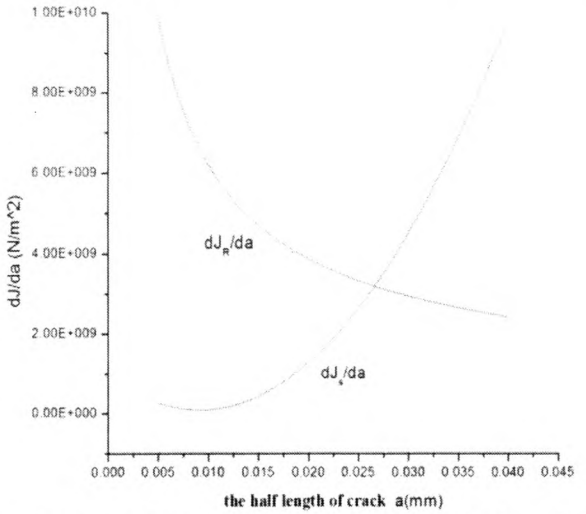

Fig. 3 Method for determination of critical crack length

The response time in relation to detected sodium leak to failure of crack was also calculated. The response time can be estimated according to the following formula:

$$t_{LBB} = (C/2 - L)/V \tag{12}$$

where t_{LBB} ,L, V and C are the response time, the half size of fathomable crack, the rate of crack growth ans the half size of critical crack. According to the upper joint, C=26.7 3mm. the value of V is set to V=2.67×10^{-3} mm/s. Finally,t_{LBB}=1635s was got. In normal operating, the time from detectable leakage to taking measures is T= 45s. So, $t_{LBB} > S_t \times T$, i.e, response time meet the demand of LBB assessment.

Summary

Based on France specification RCC-MRA16 the finite element analysis and LBB assessment of a pipeline containing cracks were accomplished by using ANSY and Franc3D software. The following conclusions can be included here.

Through the three dimensional cracks analysis of nuclear-grade pipings in creep and fracture mechanics,we should consider creep and fatigue effect in the security analysis of defective pipes.and creep effect should be attached attention.The calculation result of modified 9Cr1Mo shows:With high temperature environment in fast reactor, the crack stability, leakage measurability, and response time of this pipeline made of modified 9Cr1Mo steel are in line with the evaluation of LBB assessment.With the analysis of modified 9Cr1Mo steel which is under the condition of nuclear power engineering,this paper provide design reference for the use of modified 9Cr1Mo steel in Fast Reactor.Modified 9Cr1Mo steel can simplify the fast reactor loop structure,and this offers the possibility of high performance, multilevel optimization and economics for fast breeder reactor structures.

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