Applicability of LBB(Leak-Before-Break) Design for Main Steam Line in Korea Nuclear Power Plant

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ABSTRACT

The Leak-Before-Break(LBB) concept has been applied to the high energy piping such as Reactor coolant piping, Pressurizer Surge line, piping of safety injection system & shutdown cooling system, for Korea nuclear power plants. We had encountered a difficulty when we apply LBB to main steam line of nuclear power plant, due to low safety margin and the lack of the leak detection system. One can obtain additional design margin by removing the conservatism about the fracture mechanics behavior of the high energy piping with a crack for nuclear power plant. The contents of the conservatism related to the LBB analysis methodology consist of the work loading, the effect of the nozzle, and the fracture toughness of piping material. In this study, we have focused on the fracture toughness part. It is recommended that the main steam line is changed from low toughness material to high toughness material to increase the safety margin. In this paper, we reviewed the applicability of LBB(Leak-Before-Break) design for main steam line in nuclear power plant based on the LBB design analysis procedure. It is possible for us to successfully apply LBB for main steam line in nuclear power plant, by changing the material of MSL piping in direction of increasing the toughness and using the additional leak detection facilities. We expect that this result may be helpful in LBB application of MSL for nuclear power plant.

1. INTRODUCTION

The nuclear power plants that were designed piping whip restraints and jet impingement shield before 1983 adopted a double-ended-guillotine-break(DEGB) concept for a hypothetical pipe break situation in the primary coolant and the connected piping system in accordance to the General Design Criteria(GDC)-4 which was then the U.S licensing requirement. In the early 1980's, a study in Lawrence Livemore National Laboratory(LLNL), that was funded by the U.S. NRC and the industry, revealed that a possibility of a DEGB is very Low, and as a result of this, in July 1986 NRC approved the application of LBB concept for all the high energy pipes in Pressure Water Reactors(PWRs). Accordingly, Since 1989, nuclear power plants in Korea have adopted LBB concept for the pipes in the reactor coolant system(RCS) and the pressurizer surge line, shutdown cooling system(SCS) and safety injection system(SIS). But We had encountered a difficulty when we apply LBB to main steam line of nuclear power plant, due to low safety margin and the lack of the leak detection system. In this paper, we reviewed the applicability of LBB(Leak-Before-Break) design for main steam line in the nuclear power plant based on LBB design concept analysis procedure, results and standard design certification document. It is possible for us to apply LBB design for main steam line in nuclear power plant, by changing the material of MSL piping in direction of increasing the toughness and using the additional leak detection facilities.

2. MAIN STEAM LINE

The main steam line is a 30-inch, SA106 Gr.C Seamless for the developing nuclear power plant (see Table.1). the pipe connections are shop and field welded using Gas-Tungsten-Arc-Welding(GTAW) technique. the safe-end of main steam line was shop welded to the steam generator using GTAW method. the System 80+ and AP-600 MSL material of foreign ALWR's is SA672 Gr.C70, SA333 Gr.6.

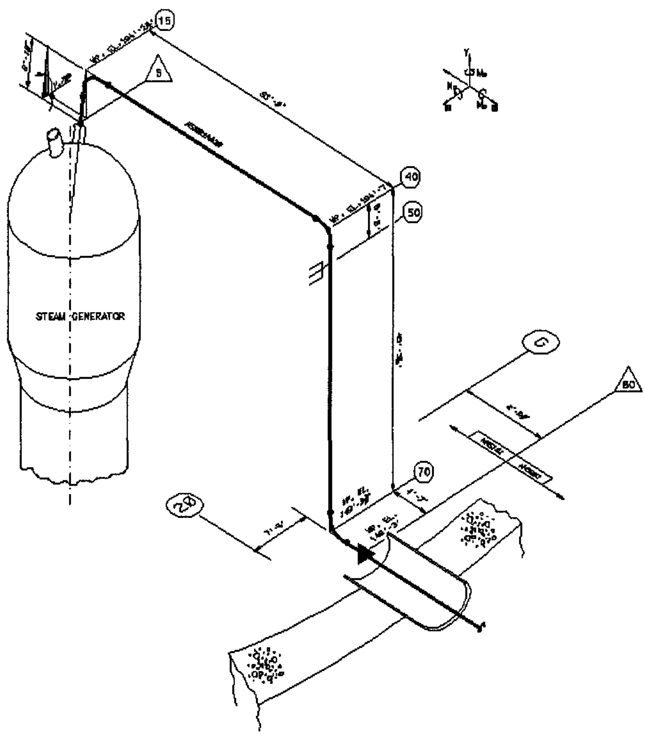


Figure 1. Analytical Isometric Drawing for Main Steam Line in NPP

Table 1. Main Steam System Description

Line No.	Material Spec.	Shop Welding	Field Welding	O.D (in)	Nominal Thickness (in)	Oper. Pr. (psia)	Oper. Temp. (°F)
MS001AA30 MS001AB30 MS001AC30 MS001AD30	SA106 Gr. C	GTAW or consumable insert	GTAW or consumable insert	30.03	1.102	1000	544.6

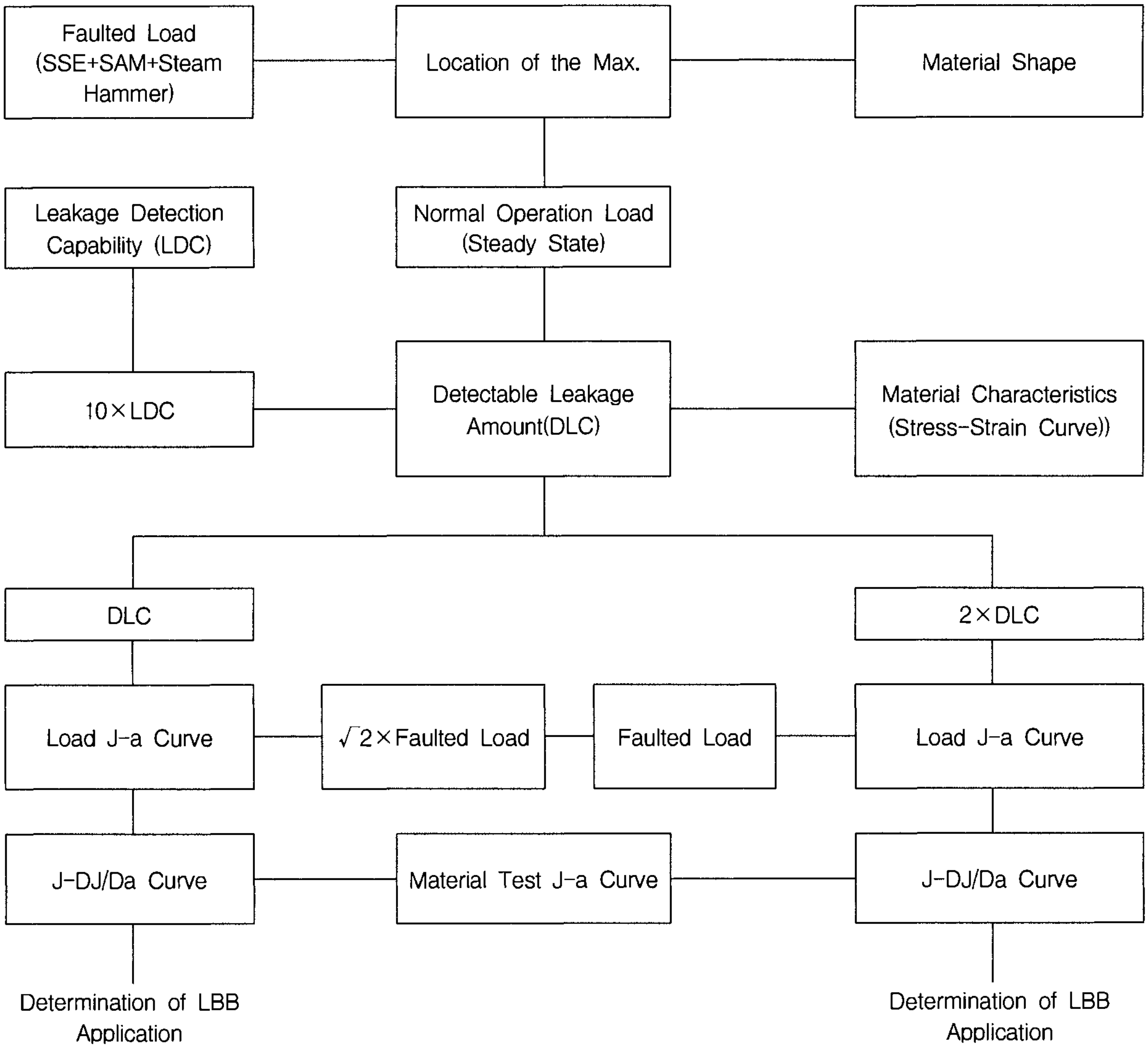
Table 2. Stress-Strain Curve at Upper Temperature

Yield Stress(ksi)	Ultimate Tensile Stress(ksi)	Ramberg-Osgood Parameter alpha	Ramberg-Osgood Parameter n
32	69.4	0.6	4.58

3. REVIEW OF THE TECHNICAL FEASIBILITY

3.1 PROCEDURE OF LBB ANALYSIS

Table 3. Procedure of LBB analysis



3.2 MAJOR PARAMETER OF LBB

3.2.1 Load

- Normal Operation Load
  - Dead Weight + Operation Pressure + Thermal Load + SAM + Steam Hammer
- Safe Shutdown Earthquake(SSE)
  - Specific parameters arisen as the result of the damping changes based on ASME Code Case N-411 are used
- SAM
  - Load by Seismic Anchor Motion

- Load by closing of Main Stop Valve
  - Steam Hammer Load when the Main Stop Valves were closed
  - Steam Hammer Load caused by the closing of Main Steam Stop Valve

3.2.2 Load Combination

After combination the absolute value of (Dead Weight + Operation Pressure + Thermal Load), the absolute Safe Shutdown Earthquake(SSE) load, and the absolute value of SAM load and MS stop valve closing load, it is used in the final safety evaluation.

3.2.3 Location of Maximum Load

We have calculated the working load on the Main Steam Line using PIPSYS Code that is designed for the stress analysis of piping and chose the location of the maximum load as the following.(see Table. 4)

Table 4. Results of crack stability evaluation and leakage size

Node No.	Normal Load ($\times 10^5$ in-lbf)	Applied Faulted Load($\times 10^5$ in-lbf)	Allowable Faulted Load($\times 10^5$ in-lbf)	Leakage Crack Size (in.)
node 5	13.08	154	117	14.9
node 10	16.05	109	115	14.6
node 20	19.75	74	110	14.3
node 25	17.65	90	113	14.5
node 30	19.07	88	113	14.4
node 35	24.01	55	108	14.0
node 45	20.87	34	110	14.2
node 50	17.13	55	113	14.5
node 55	30.21	161	132	16.1
node 60	22.98	145	108	14.1
node 65	43.02	23	90	12.6
node 75	38.29	92	93	12.9
node 80	35.73	104	95	13.1

3.2.4 Leakage Detection Capability(LDC)

One of the major parameters for the LBB analysis is the leakage detection capacity(LDC) that can be detected in the containment. this parameter is multiplied ten times to be used for the LBB analysis as defined in NUREG-1061, Volume 3, and SRP 3.6.3.

3.2.5 Detectable Leaking Crack Length(DLC)

When the LDC, normal operation load, and the material properties for Stress-Strain curve are given, then the Detectable Leaking Crack Length(DLC) can be calculated at each node using Pipe Crack Evaluation Program(PICEP). the information of the material properties is the yield strength, ultimate strength, and Ramberg-Osgood parameter, etc.

4. ANALYSIS OF THE CRACK STABILITY

4.1 Theoretical Aspect(FEM)

- J-a(Stress Intensity Factor-Crack Size) Curve
  - For the cracks a, a+x, a-x and the cracks 2a, 2a+x, 2a-x, the J value was calculated with the FEM by applying the outside loads
  - After calculating the J value with the function of the crack length "a", the dJ/da value was calculated around "a" and "2a"
  - Fitting for J and dJ/dA(multiple equations were used)
    - $J(a) = C_1a^2 + C_2a + C_3$
    - $dJ/da = 2C_1a + C_2$(applies also to the "2a" case)

      where C_1,C_2 and C_3 are polynomial constants
  - Development of the J-T diagram(see Fig.2)

4.2 Experimental Aspect(FEM)

- Tensile Test of the Material
- Creation of the J-a Curve(Curve fitting was used by appling the Power Law)
  - $J = C_1 \triangle a^{C2}$
  - $dJ/da = C_1C_2(\triangle a)^{(C2-1)}$
- Development of the J-dJ/da Diagram(see Fig.2)

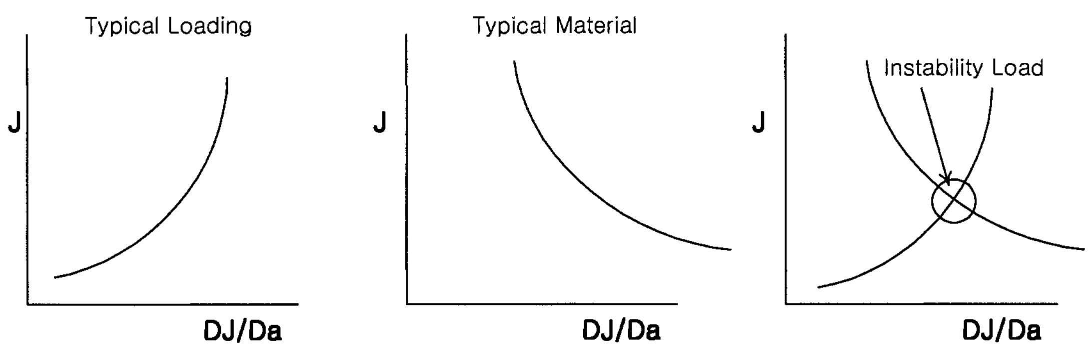


Figure 2. Development of J-R Diagram

4.3 Evaluation of the Crack Stability

At first, the applied loads, detectable leaking crack length and the material properties, etc must be known(the piping load and the bending moment are given in Table 4). Secondly, to evaluate the safety of a through crack in the piping, we must compare the resistant factor(J and dJ/da) of the piping material with those of the external load working on piping. The safety criteria of ductile piping about the crack behavior are as follows :

$$J_{applied} < J_{material}$$

and

$$dJ/da_{\ applied} < dJ/da_{\ material}$$

4.4 Evaluation Result

The J-T evaluation method is introduced for the main steam line of the developing nuclear power plant in Korea. The result of LBB preliminary evaluation was satisfied in excess of about 2%. However the regulatory body was not allowed this result due to the uncertainty of FEM. The modified PED developed by CARR has been applied to the MSL for nuclear power plants. The crack stability has been evaluated for the Allowable Faulted Load and Applied Faulted Load using Low Bound Curve of modified PED. As a result, the crack stability has not been satisfied at node #5(welding point with S/G), nodes #55 and 60, (below pipe support), and node #80(welding point with containment penetration).(see Table. 4)

4.5 Further Studies

4.5.1 Crack Stability of Main Steam Line

There is a little safety margin for MS line material of the developing nuclear power plant in Korea. Thus, it is desirable that the material of MSL is changed from ASME SA106 Gr.C Seamless to SA672 Gr. C70 or SA333 Gr.6 to meet acceptance criteria. Reevaluating the preliminary LBB analysis based on the modified material properties, it is possible to apply the LBB design concept for MS line LBB with the modified material nuclear power plant.

4.5.2 Leakage Detection System

NRC had approved the containment sump, reactor cavity sump and containment aircooler condensate as the leakage detection system of MS line for AP-600 and System 80+. But Korea regulatory body had not accepted the previous primary leakage detection system as the leakage detection system for MS line LBB of nuclear power plant in Korea. Thus, we consider the new leakage detection system including the operating parameter such as moisture or ammonia.

5. CONCLUSION

In this paper, the preliminary LBB analysis was performed for main steam line based on the LBB analysis procedure, as suggested in NUREG 1061, Vol.3 and SRP 3.6.3. but it has a little safety margin for the MS line. It is possible for us to apply LBB design for main steam line of the developing nuclear power plants in Korea, in case of changing the material(SA672 Gr. C70 or SA333 Gr.6) of MSL in direction of increasing the toughness and using the additional leakage detection system which is different from the previous facilities.

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