Development of an Integrity Evaluation System for Nuclear Power Plants*

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This paper describes the structure and development strategy for integrity evaluation system for nuclear power plants called NPP-KINS/SAFE. NPP-KINS/SAFE consists of three different programs covering the integrity assessment of reactor pressure vessel, pipings, and pressure tubes, respectively. The system has been developed based on currently available codes and standards, and includes a number of databases, expert systems, and numerical analysis schemes. NPP-KINS/SAFE is applicable for various types of nuclear power plants constructed in Korea with the aid of attached database systems including plant specific data. Case studies for the developed system are also provided.

Key Words: Integrity Evaluation, Expert System, Nuclear Power Plant, Reactor Pressure Vessel, Piping, CANDU

1. Introduction

Since the first commercial operation of a KORI nuclear power plant in 1978, 12 nuclear power plants are currently operating in South Korea. The total amount of electricity produced by these nuclear power plants has increased to 10 316 MW accounting for 25% of the overall national electricity production. With an increasing number of nuclear power plants, it becomes increasingly critical to operate them with minimal maintenance over long periods to justify the high initial capital cost.

The assessment of the integrity of nuclear power plants has been studied over last decades, and has resulted in a number of standards specifying failure prevention criteria. In Korea, different vendors such as Westinghouse, Combustion Engineering, Framatome and AECL constructed various types of nuclear power plants. Since these vendors used their own design specifications, different integrity assessment procedures are required for each corresponding nuclear power plant.

ASME and USNRC (USA) provide specific integrity evaluation procedures for flawed components used for PWR type nuclear power plants. AECCB (Canada) also provides regulations and recommendations regarding the integrity assessment procedures for CANDU type nuclear power plants. However, it is time-consuming to assess flawed components, owing to the complicated procedures involved in the assessment. For this reason, several computer programs have been developed⁽¹⁾⁻⁽⁶⁾, but are applicable only for limited components, and require a large amount of input data. In the present paper, an integrity evaluation system covering major components in both PWR and CANDU type nuclear power plants has been developed by integrating application programs and database management system.

The developed integrity evaluation system provides safety assessment for three crucial components; reactor pressure vessels, pressure tubes (CANDU type) and pipings. In this paper, three different integrity assessment procedures were individually pro-

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Fig. 1 Main window for NPP-KINS/SAFE

grammed and integrated into a system named as NPP-KINS/SAFE. NPP-KINS/SAFE consists of three programs as shown in Fig. 1; RVIES (Reactor Vessel Integrity Evaluation System), PTIES (Pressure Tube Integrity Evaluation System) and NPIES (Nuclear Pipings Integrity Evaluation System). Each program has been designed to enhance the integrity evaluation procedure for corresponding components. The structure and development strategy for each of these programs are illustrated, in detail, in this paper.

2. Development of RVIES

ASME and USNRC have proposed the integrity evaluation procedure for an RPV. According to these procedures, four different criteria considering PTS, USE decrease, P-T limit and fatigue must be satisfied to operate an RPV. The developed RVIES is designed to cover these four different analysis procedures by implementing four different sub-modules as follows.

2.1 P-T limit curve module

The ASME Sec. III⁽⁷⁾ provides the pressure temperature limits for RPV operation by postulating a crack. Since ASME Sec. III provides overly conservative results in terms of stress calculation, the finite difference method was adopted for computing primary stress and thermal stress.

The P-T limit curve module is designed with three parts; data input part, analysis part and output part. From the GUI assisted window, a user can input all the required data to generate P-T limit curve generation. In the resulting P-T limit curve diagram, safe and unsafe regions are divided by each limit curve for corresponding operating conditions.

2.2 PTS analysis module

The PTS analysis module was developed (as shown in Fig. 2) based on rules specified in USNRC⁽⁸⁾ in which RT_{PTS} at the end of reactor life must be satisfactory for the given PTS screening criteria.

The PTS analysis module also consists of three

Fig. 2 Flow chart for the PTS analysis module

parts; the data input part, analysis part and output part. Input data such as material properties, flaw shape, temperature, pressure, heat transfer coefficients and the analysis control data can be either provided by the user, or searched from the database. The decrease of RT_{NDT} due to the irradiation embrittlement, ΔRT_{NDT} , is usually obtained from the surveillance test data. If the surveillance data are not available, the ΔRT_{NDT} is automatically calculated by applying tables specified in USNRC Reg. Guide 1.99⁽⁹⁾. The variations of temperature, pressure and heat transfer coefficients with increase in time are required for the time dependent PTS analysis. The 9th order polynomials were applied for simulating each variation, and corresponding coefficients were used for input data.

The distribution of temperature and pressure is computed in the analysis part, and then stress calculations are followed for each time increment. The FDM was used for primary and thermal stress calculations. For the calculation of critical RT_{NDT} , a numerical

scheme which traces the contact point between K_I and K_{IC} curves was applied.

In the output part, the variation of crack depth with increase in time, the variation of stress intensity factors with increase in time, and the critical crack depth diagrams are produced. In addition, information on crack initiation, crack arrest, warm pre-stressing effect, and upper shelf toughness is also provided with resulting diagrams.

2.3 USE evaluation module

The USE evaluation module has been developed to provide USE evaluation based on USNRC regulations⁽⁹⁾. This module also consists of data input part, analysis part and output part.

In the data input part, material properties for RPV, neutron fluence, USE obtained from surveillance test data, and estimated fluence at the end of reactor life are required. The analysis part provides the decrease of USE based on the diagram predicting the decrease rate of USE provided by USNRC regulations⁽⁹⁾. The output part provides evaluation results in tabular and graphical forms.

For determination of USE, the following inference process was programmed into this module. If the first surveillance test data are available, the neutron fluence and the decrease of USE can be determined. For the case that the test data are scattered, the prediction lines for 0.35% copper and 1.0% nickel contents are used. Once the USE is determined, the decrease in USE (which is defined as the difference between the USE of unirradiated specimens and the predicted values) is marked on the USE diagram. The prediction line can be obtained by drawing a line passing the marked point and parallel to the standard prediction line. If second surveillance test data is available, then the decrease of USE can be obtained from two sets of test data, respectively. The prediction line is then obtained by drawing a line passing through these two points marked on the USE diagram. If the second point is lower than the first point, a parallel line is conservatively used. Once the prediction line is determined, the decrease of USE at the end of life is obtained from the USE diagram, and the USE at the end of life can be calculated from the following equation:

$$USE_{\text{end}} = UNE_{\text{unirrad}} - \Delta USE_{\text{predic}} \quad (1)$$

where, USE_{end} : USE at the end of life

UNE_{unirrad} : USE of unirradiated material

$\Delta USE_{\text{predic}}$: predicted USE change

2.4 Fatigue analysis module

The fatigue analysis module has been developed based on ASME Sec. XI⁽¹⁰⁾, and consists of data input part, K -calculation part, crack growth calculation part and output part.

From the data input part, various kinds of data, such as inner radius, thickness, internal pressure, flaw shape parameters, and the number of cycles to be considered at the end of reactor life are provided by the user. The transient data, which is used for K calculation, is prepared in the form of a text file. In this module, two different transient data files were included for corresponding nuclear power plants. This data was prepared based on the stress reports provided by each vendor.

The K -calculation part provides K values required for the fatigue crack growth calculation for surface cracks in an internally pressurized cylinder. In the crack growth calculation part, the final crack growth is obtained by repeating the calculation for the user specified number of cycles. The final crack shape is provided both in graph and text form from the output part.

3. Development of PTIES

AECL and COG (CANDU Owner's Group) published FFSG (Fitness For Service Guidelines) for the flaw evaluation of CANDU pressure tubes⁽¹¹⁾. In this paper, an integrity evaluation system for the pressure tubes named as PTIES was developed based on this FFSG. The developed system consists of modules for crack extension analysis, unstable fracture evaluation, plastic collapse evaluation, LBB (Leak-Before-Break) analysis, and the remaining life estimation.

3.1 Pressure tube integrity assessment procedures

3.1.1 Analysis of flaw extension The evaluation procedures are different depending on the flaw shape, which can be classified as a sharp flaw, blunt notch and blister. The developed program covers the integrity evaluation procedures for a sharp flaw and a blunt notch. While the evaluation for a sharp flaw (crack) is based on the fracture mechanics analysis, the evaluation for a blunt notch is based on the solid mechanics, and thus the flaw extension evaluation is only applicable to a sharp flaw.

For the evaluation of a sharp flaw extension, the DHC (Delayed Hydride Cracking) and the fatigue flaw extension due to operating transients were considered. In the present paper, the empirical equation provided by AECL⁽¹²⁾ was adopted for the calculation of DHC crack extension, and the Paris' law was used for the calculation of fatigue crack extension as specified in the FFSG.

3.1.2 Pressure tube integrity evaluation The FFSG recommends several parameters for integrity evaluation, such as stress intensity factor, fracture toughness, plastic collapse stress, membrane stress, fatigue strength, hydrogen equivalent concentration,

Fig. 3 The flow chart for a sharp flaw evaluation procedure

terminal solid solubility, and LBB evaluation parameters. Figure 3 shows the flowchart for integrity evaluation procedures for a sharp flaw.

3.2 Structure of PTIES

The PTIES consists of four parts; data input part, database/database management part, calculation part, and output part.

3.2.1 Data input part For the pressure tube integrity assessment, various kinds of data such as pressure tube dimensions, flaw dimensions, material properties, and transient data are required. In order to reduce the error during the input process, the automatic data validation checks routine which accepts only reasonable data was included in the data input part. The database can also be used for the data input process.

3.2.2 Database/Database management part In order to manage the input data efficiently, a database and database management module was developed. The developed database provides frequently required data, such as pressure tube dimensions, flaw dimensions, material properties, transient data, sequence of cooling data and finite element analysis data. The database management module provides functions for data insertion, deletion, sorting and searching, and is compatible with other commercial database programs.

The K_I data, which were included in the developed database, were prepared by performing K_I calculation with three-dimensional finite element models for various surface cracks. Varying the crack depth to crack length ratio from 0.2 to 1.0, and the crack

Fig. 4 A typical 3-D mesh used for the finite element analysis

depth to crack wall thickness ratio from 0.2 to 0.8 performed a total of 40 analyses. The loading condition was set to 10.4 MPa of internal pressure considering normal operating condition. Material properties were adopted from a CANDU type nuclear power plant constructed in Korea. Figure 4 shows a typical finite element mesh used for the analysis. The transient data, which are used for the crack extension analysis, were prepared by considering the test loading conditions as specified in the design documents, and added to the database.

3.2.3 Calculation part The calculation part consists of sharp flaw evaluation and blunt notch evaluation. The sharp flaw evaluation part provides the determination of minimum evaluation flaw size, stress intensity factor calculation, fracture toughness calculation, fatigue crack extension calculation, DHC extension calculation, terminal solid solubility calculation, and LBB analysis. For the blunt notch evaluation, the determination of minimum evaluation flaw size, fatigue crack initiation, DHC initiation, terminal solid solubility calculation, plastic collapse analysis, and LBB analysis are provided.

3.2.4 Output part This part provides the resulting output for the entire analysis. The structure of output was developed based on the FFSG so that a user can easily determine the integrity based on standard failure criteria. The output part provides integrity determination, remaining life, and resulting parameters such as stress intensity factors, fracture toughness, plastic collapse stress, the applied stress, crack extension, hydrogen concentration, terminal solid solubility, etc. Several user-friendly tools were

added for graphic display, table display, write in file, summary report, etc. For the LBB analysis, critical crack length, penetrated crack length, velocity of crack extension by DHC, and time for the flaw to reach the critical crack length are also provided.

4. Development of NPIES

The authors in an earlier paper had proposed an integrity evaluation system, which was named as NPIES⁽¹³⁾. In the present paper, NPIES has been modified and extended, and attached to NPP-KINS/SAFE.

4.1 Structure of NPIES

Figure 5 shows the structure of NPIES. NPIES consists of four parts; user interface, database, expert, and pipe integrity assessment parts, respectively. The structure of NPIES was designed to provide efficient integrity evaluation of nuclear pipings.

4.1.1 User interface part In the user interface part, pipe geometries, crack geometries, and loading conditions are provided by the user, or can be chosen from database.

4.1.2 Database part NPIES includes the material property database covering tensile properties and fracture toughness for nuclear piping steels such as SA312 TP304, SA312 TP316, SA106 Gr. C etc. The majority of test data was obtained from the domestic material test program such as YGN 3 & 4 and UCN 3 & 4. The additional data was obtained from a literature survey⁽¹⁴⁾. A module was added to the database for the curve-fitting of raw data in the user specified range.

In order to evaluate a cracked pipe, tensile properties such as σ_y , σ_u , α , n and fracture toughness properties are required. Since this data are not always available, the material property database was developed in a hybrid database format. The devel-

oped hybrid database provides an effective search of material properties by using DBMS (DataBase Management System), KBMS (Knowledge Base Management System), and neural network system. The DBMS is utilized to search for the user specified material properties from the database. If the required material property is not available in the database, the KBMS can be used for the prediction of an unknown material properties based on the knowledge stored in the knowledge base. In the knowledge base part, the following levels of knowledge were stored.

Level 1: Knowledge for inferring σ - ϵ curve from yield strength.

Level 2: Knowledge for inferring σ - ϵ curve from yield strength and tensile strength.

Level 3: Knowledge for inferring J-R curve from fracture strain.

The theories applied on DBMS and KBMS for NPIES were explained in detail in the previous paper⁽¹³⁾.

The J-R curves for various materials are available from the material database. However, the scatter of J-R curves from the test data is usually wide even for the same material. In the present program, the lowest J-R curve is automatically selected for a conservative evaluation. However, this scheme ignores the characteristics of materials. For this reason, a neural network system was developed for J-R curve inference. The J-R curve can be predicted by the developed neural network system based on the toughness level of material which is determined from the tensile properties. The J-R curve can thus be obtained in three different methods.

4.1.3 Expert part In the expert part, an optimal evaluation method for the selected material is recommended depending on either the characteristics of material or the criteria given in the ASME Sec. XI, Appendix C and H. Detailed explanations are given in a previous paper⁽¹³⁾.

4.1.4 Pipe integrity assessment part The pipe integrity assessment part provides three options; integrity evaluation, fracture mechanics analysis based on ASME Sec. XI, and LBB evaluation based on NUREG 1061 Vol. 3⁽¹⁵⁾.

a. Integrity evaluation

The integrity evaluation option provides practical application methods based on LEFM (Linear Elastic Fracture Mechanics), EPFM (Elastic Plastic Fracture Mechanics), and limit load analysis. For the EPFM analysis, DPFAD (Deformation Plasticity Failure Assessment Diagram), CDFD (Crack Driving Force Diagram) and J/T analysis methods are provided.

The ductile fracture handbook⁽¹⁶⁾ was used for

Fig. 5 The structure of NPIES

computing K_I , J -integral, and limit load. The geometry correction factors required for K_I calculation were tabulated and stored in the database, and the linear interpolation scheme was adopted for computing geometry correction factors.

b. Fracture mechanics analysis based on ASME Sec. XI

In ASME Sec. XI, three options for fracture mechanics analysis are available: Appendix A for fatigue crack growth analysis, Appendix C for austenitic steel pipes, and Appendix H for ferritic steel pipes.

In order to obtain the maximum allowable flaw depth, the limit load analysis specified in Appendix C and Appendix H was applied. The final crack depth at the end of the evaluation period is computed by applying the fatigue crack growth analysis as explained in 2.1.4. The integrity evaluation is achieved by comparing the maximum allowable flaw depth with the final crack depth from the fatigue analysis.

c. LBB evaluation based on NUREG 1061 Vol. 3

The LBB evaluation option was developed to check the applicability of LBB design concept based on the conditions specified in NUREG 1061 Vol. 3. There are two sub-modules in the developed LBB module; one is the DLC module, which provides the crack size calculation for the leakage rate of 10 gpm with PICEP⁽¹⁷⁾. The DLC module consists of three parts; pre-process part, post-process part, and PICEP program. The pre-process part prepares input data file for PICEP, and the post-process part provides analysis results out of PICEP in text and graph form. The other is J/T evaluation module determining the instability, and subsequently the applicability of LBB.

The J/T module calculates T_I for the applied loads, and calculates T_{JR} from J_R curve. By comparing T_I with T_{JR} , the instability of structure is determined and subsequently the applicability of LBB design concept.

4.2 Determining unknown material properties

For the determination of unavailable material properties, a knowledge based system and a neural network system were developed.

4.2.1 Knowledge based system Although the tensile properties of nuclear grade steels are easily found in open literature, the material constants C_1 , C_2 of J-R curve in the following equation are seldom found.

$$J_R = C_1(\Delta a)^{C_2} \quad (2)$$

The following empirical equation was obtained by statistically correlating the tensile data and the J-R data of SA312 TP304 and SA106 Gr. C steels⁽¹⁸⁾.

$$C_1 = 1.687 E k (S'_u \times \sqrt{\epsilon_c})^{1.06} \quad (3)$$

Fig. 6 The proposed neural network model for J-R curve prediction

$$C_2 = \begin{cases} 0.38 & \text{for carbon steel} \\ 0.61 & \text{for stainless steel} \end{cases}$$

where E is Young's modulus, k is non-dimensional constant, S'_u is defined as S_u/E and ϵ_c is fracture strain. On the basis of Eq. (3), a knowledge based system for J-R curve prediction was developed by referring domestic material test results of YGN 3 & 4. The theories on developed knowledge based system are illustrated in detail in the previous paper⁽¹³⁾.

4.2.2 Neural network system In the present paper, a neural network system was developed for the prediction of unknown J-R curves. The strain energy density, w_f can be obtained from σ - ϵ curve up to the point of fracture strain, ϵ_c . The J-R curve was observed to be different depending on the strain energy density, and thus the proposed neural network system was designed to classify J-R curves corresponding to the strain energy density. Figure 6 shows the structure of developed neural network model, which employs an ordinary three-layer network. Tensile properties were used for input units of the network. By applying preliminary tests, the neural network parameters were determined as follows;

- learning rate: $\alpha_m = 0.1$,
- momentum factor: $m = 0.9$,
- range of initial weights = $-1 \sim 1$,
- number of hidden layers = 14.

The relationship between the input and output signals was formulated with a sigmoid function which is given by

$$O_j = f(NE_T) = \frac{1}{e^{(-NE_T)}} \quad (4)$$

where,

$$NE_T = \sum_{j=1}^n W_{ji} I_j$$

O_j = output signal of j th unit

f = activation function

W_{ji} = connection weight between the i th and j th units

I_i = input signal from the i th to the j th units

5. Case Studies

In order to verify NPP-KINS/SAFE, several case studies were performed for each program.

5.1 RVIES

5.1.1 P-T limit curve analysis For the verification of P-T limit curve analysis module, a typical reactor pressure vessel in Korea was analyzed for 100 F°/HR HEATUP condition. The input data used for the case study are listed in Table 1, and the resulting P-T limit curves are given in Fig. 7. For the verification of analysis results, the same input conditions were applied for PTLC which was developed by Korea Atomic Energy Research Institute⁽¹⁹⁾. A good agreement was observed between two results as summarized in Table 2.

Table 1 Material properties used for the P-T limit curve analysis

Inner radius	82.0 in
Outer radius	90.0 in
Operating pressure	2250 psi
Initial temp.	70° F
Final temp.	550° F
Eff. flow area	20.0 ft ²
eff. Coolant flow rate	1 × 10 ⁸ lbm/hr
eff. Hydraulic diameter	10.0 in
RT _{NDT} at ° T	200° F
RT _{NDT} at ° T	140° F

Table 2 Comparison of results between RVIES and PTLC

Temperature (° F)	Allowable pressure (psi)		Difference (%)
	PTLC	RVIES	
150	352.82	352.84	0.0043
220	466.80	466.84	0.0075
290	941.76	941.78	0.0024
350	2000.64	2000.64	0.0000

Fig. 7 Resulting P-T limit curve diagram for the verification analysis

5.1.2 PTS analysis module For the verification of PTS analysis module, the input conditions given in Ref.(20) were adopted which are known as Rancho Seco transient. Material properties are listed in Table 3. The crack depth to crack length ratio of postulated crack was set to 1/6 as specified in the ASME⁽⁷⁾. The resulting critical crack depth diagram demonstrated that a crack with a/w of 0.1 will initiate after 1400 seconds, and then penetrate through the wall. A strong similarity was observed in comparison with Jung et al.'s⁽²⁰⁾ results.

5.1.3 USE evaluation module The input data and resulting output for case study are shown in Fig. 8. The resulting USE was verified by comparing with the hand calculation result. The USE evaluation was achieved for the operation period of 24 EFPY (Effective Full Power Year) with the corresponding neutron fluence of 23.9×10^{18} n/cm² on the inner surface. The resulting decrease of USE was -43.9%, and thus, the resulting USE after 24 EFPY was calculated

Table 3 Material properties used for the PTS analysis

Wall thickness	6.0 in
Material type	SA 508 class 2
Cu content	0.22 weight %
Ni content	0.69 weight %
RT _{NDT0}	70° F
Fluence at the end of life	5.6×10^{19} n/cm ²

Fig. 8 The input data and output results for the USE module verification analysis

Fig. 9 The input data and resulting diagram for the fatigue analysis

as 37.2 ft-lb. At the point of $a/w=1/4$, the neutron fluence of 16.18×10^{18} n/cm² was applied, and the corresponding USE was 39.8 ft-lb. A good agreement was observed between hand calculations and program output as shown in Fig. 8.

5.1.4 Fatigue analysis module The input data and corresponding output results are shown in Fig. 9. The input conditions were adopted from a fatigue analysis performed on a nuclear power plant in Korea⁽²¹⁾. The two results were observed to be in agreement.

5.2 PTIES

In order to verify PTIES, a case study was performed on a sharp flaw embedded in the axial direction. The input data were adopted from the test report⁽¹²⁾ prepared by AECL. The analysis results are summarized in Table 4, and showed excellent agreement (within 1%) when compared to the results reported in the Ref.(12).

5.3 NPIES

In order to verify NPIES, case studies were performed for a pipe (outer diameter, $D_o=413.5$ mm, thickness, $t=26.2$ mm) with a through wall crack (crack length $2c=223.9$ mm). The pipe is subject to a bending moment of 565 kN·m, and material properties are listed in Table 5.

A total of 9 different material properties were applied for the same geometry. Resulting maximum allowable loads were compared to those from NRCPIPE⁽³⁾, which was developed by Battelle. The same crack geometries, loading conditions, and material properties were applied for both NPIES and NRCPIPE for each case. Comparisons of analysis results are summarized in Table 6. The maximum difference of 12.5% was observed from 9 cases. This

Table 4 Summary of analysis results from PTIES

Data input		Evaluation parameters		Results	
Wall Thickness	4.2 mm	Fatigue crack extension	0.010 mm	Hydrogen concentration not exceed TSS	
Inner Radius	52.9 mm	DHC crack extension	0.073 mm	Margin on S.I.F.	3.17
Flaw Depth	1.20 mm	S.I.F. for initial crack	8.6mm MPa	Margin on Plastic collapse	4.96
Flaw Length	5.00 mm	Fracture toughness	27.1mm MPa	LBB	Satisfy
Initial Hydrogen	9.0 ppm	Hoop stress	130.2 MPa	Allowable cooldown	3 times
Deuterium pickup	11.3 ppm	Plastic collapse	646.4 MPa		

Table 5 Material properties used for NPIES analysis

M. P.	Tensile Properties							J-R curves	
	E (GPa)	σ_y (MPa)	σ_u (MPa)	σ_f (MPa)	n	f	C_1 (kJ/m ²)	C_2	
YS-A	204.4	221	510	365.5	3.31	4.14	0.00108	1330.7	0.44
YS-B	204.4	248	540	394.0	1.91	4.60	0.00121	1199.0	0.58
YS-C	204.4	269	572	420.5	7.00	3.35	0.00132	1283.9	0.66
YS-D	204.4	276	559	417.5	2.62	4.49	0.00135	945.3	0.75
YS-E	190.3	152	386	269.0	5.74	3.81	0.00080	912.0	0.58
YS-F	190.3	172	448	310.0	3.37	3.92	0.00090	662.8	0.91
YS-G	190.3	179	462	320.5	2.77	4.08	0.00094	668.0	0.87
YS-H	188.5	152	476	314.0	2.82	5.65	0.00080	1092.0	0.55
YS-I	188.5	159	462	310.5	3.97	3.49	0.00084	1223.0	0.47

Table 6 Summary of analysis results from NPIES

Method ID	P _{max} with J-R _{exp.} Curve			P _{max} with J-R _{pred} Curve	
	NRCPIPE KNm	NPIES kNm	Diff ⁽¹⁾ %	NPIES KNm	Diff ⁽²⁾ %
YS-A	980.43	886.19	9.6	945.25	6.2
YS-B	N.A. ⁽³⁾	1197.17	-	1103.81	-8.5
YS-C	1115.17	1063.54	4.6	1010.09	-5.3
YS-D	N.A.	1181.05	-	1131.92	4.3
YS-E	722.43	666.57	7.7	673.50	1.0
YS-F	N.A.	938.82	-	816.08	-15.0
YS-G	N.A.	947.12	-	854.82	-10.8
YS-H	694.34	646.13	6.9	660.97	2.2
YS-I	857.65	750.55	12.5	802.23	6.4

$$^{(1)} \text{ Difference} = \frac{P_{\max \text{ NPIES}} - P_{\max \text{ NRCPIPE}}}{P_{\max \text{ NRCPIPE}}}$$

$$^{(2)} \text{ Difference} = \frac{P_{\max \text{ NPIES}(\text{exp.})} - P_{\max \text{ NRCPIPE}(\text{pred.})}}{P_{\max \text{ NRCPIPE}(\text{exp.})}}$$

⁽³⁾ N.A. = Critical error occurred during the evaluation

difference seems to be caused by difference in J-Integral calculation methods employed in two programs. For all cases analyzed, NPIES provided lower maximum allowable loads than NRCPIPE.

In order to verify the developed neural network system, 9 J-R curves were predicted based on the strain energy density and other tensile properties. The obtained J-R curves were applied for the same geometry, and resulting maximum allowable loads were compared to those from test data in Table 6. The maximum difference of 15% was observed from 9 cases. In comparison with 30% of difference resulted from KBMS⁽¹³⁾, the proposed neural network system produced much improved results for the cases analyzed.

6. Conclusions

In the present paper, an integrity evaluation system for nuclear power plants named as NPP-KINS/SAFE was developed. NPP-KINS/SAFE provides efficient and accurate integrity evaluations for RPV, pipings and pressure tubes with the aid of attached databases, expert systems and numerical schemes. Case studies were successfully performed to demonstrate the usefulness of NPP-KINS/SAFE.

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