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# Expert System Approach for Reactor Vessel Integrity

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This paper describes the development strategy of a prototype expert system, called RViES, for the reactor vessel integrity. The main objectives of the system are to assist engineers to perform fatigue and fracture mechanics analyses of reactor vessels quickly and accurately. The system consists of three parts; user interface, knowledge base and inference engine. Various rules recommended in codes and standards are stored in the knowledge base. Several case studies were performed to check the usefulness of the system.

**Key Words:** Expert System, Reactor Vessel, Fracture Mechanics, Fatigue

## 1. Introduction

The expert system, as part of artificial intelligence (AI), is being applied for the purpose of design and diagnostic analysis in a variety of engineering fields. In power generation industry, applications of AI technology to the integrity assessment of power plant facilities have been progressed from early 80's and several prototype softwares such as ESR(Jovanovic, 1989), RAMINO(Luci a and Volta, 1991) and DIAS (Okamoto, et al., 1987) have been developed. The ESR system aims at providing support for assessment and management of their remaining life of the high temperature pressurized components (mainly pipings) in power plants. The RAMINO system deals with reliability assessment for maintenance and inspection optimization. The DIAS system can be used for defect identification and its assessment in conjunction with non-destructive test equipments. However no specific system applicable for the reactor vessel integrity has been developed.

With aging of nuclear power plants in Korea, there is a demand to establish the structural integrity evaluation method for power plant facilities. A prototype expert system, called RViES (Reactor Vessel Integrity Evaluation System), has been developed for the integrity evaluation of the reactor vessel. The main objectives of the system are to assist engineers to perform fatigue and fracture mechanics analyses quickly and accurately. The system consists of three parts; user interface, knowledge base and inference engine. Various fatigue and fracture mechanics analysis rules recommended in codes and standards, for radiation embrittlement, pressurized thermal shock (PTS), pressure-temperature (P-T) limit curve and fatigue analysis are stored in the knowledge base. Several case studies were performed to check the usefulness of the system.

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## 2. Integrity Assurance of Reactor Vessels

In order to ensure the structural integrity of reactor coolant pressure boundary, it should be designed, manufactured and operated in accordance with the following codes and standards.

### 2.1 Radiation embrittlement

The probability of fracture in a reactor vessel increases with the decrease in ductility and the increase in brittleness. Therefore the radiation embrittlement should be controlled properly. It is specified in 10 CFR 50(1987), Appendix $G$ that the reactor vessel beltline materials must maintain

Charpy upper shelf energy (USE) of no less than 50 ft-lb(68 Joule) throughout the life of the vessel. Otherwise the reactor vessel may be operated provided that the following requirements are satisfied :

1) A volumetric examination of 100 percent of the beltline materials.

2) Additional evidence of the fracture toughness of the beltline materials by performing supplemental tests.

3) Fracture mechanics analysis which provides the adequate margin of safety.

**2.2 Pressurized thermal shock(PTS)**

In 10 CFR 50 (1987), § 50.61, the PTS event is defined as an event or transient in pressurized water reactors causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. If the wall temperature of operating reactor vessel is lowered below the nil-ductility transition reference temperature ($RT_{NDT}$), the material fracture toughness is decreased significantly and the PTS may occur. The reference temperature $RT_{PTS}$ for use as a PTS screening criterion must be calculated as follows ;

$$\Delta RT_{PTS} = CF \cdot f^{(0.28-0.10\log f)} \tag{1}$$

where $CF$ is the chemistry factor, a function of copper and nickel content and "f" means the best estimate neutron fluence, in unit of $10^{19}$ n/cm$^2$ ($E \geq 1$ Mev), at any depth in the vessel.

In addition, $RT_{PTS}$ including the effect of fluence can be calculated as follows ;

$$RT_{PTS} = \text{Initial } RT_{PTS} + \Delta RT_{PTS} + \text{Margin} \tag{2}$$

where initial $RT_{PTS}$ is reference temperature for unirradiated material and margin is 66°F(18.9℃) for welds and 48°F(8.9℃) for base metal if generic values of initial $RT_{PTS}$ are used, and margin is 56°F(13.3℃) for welds and 34°F(1.1℃) for base metal if measured values of initial $RT_{PTS}$ are used.

**2.3 Pressure-temperature limit curve**

In order to prevent catastrophic failure of reactor vessels, design criteria are provided in ASME Section III (1992). The P-T limit curve, upper limit for pressure as a function of temperature during heatup and cooldown for a given service period, is determined based on the following relation

$$2K_{IM} + K_{IT} < K_{IR} \tag{3}$$

where $K_{IM}$ and $K_{IT}$ are membrane and thermal stress intensity factors, respectively, and $K_{IR}$ is the reference stress intensity factor which is defined as a function of $RT_{NDT}$ obtained from surveillance tests.

**2.4 Fatigue**

There is a possibility of fatigue failure of reactor vessels due to various operating transients. The fatigue design curves are provided in ASME Section III (1992) to design against fatigue failure, and allowable number of cycles are determined from the fatigue curves and the linear cumulative damage rule(Miner, 1945). Therefore the integrity of reactor vessel is assured through comparison between the design fatigue life and the actual number of operating transients.

If a crack is detected during an in-service inspection, following ASME Section XI (1992) acceptance criteria based on flaw size should be satisfied.

$$a_f < 0.1\ a_c \tag{4a}$$
$$a_f < 0.5\ a_i \tag{4b}$$

where

$a_f$ = the maximum size to which the detected flaw is calculated to grow in a specified time period

$a_c$ = the minimum critical flaw size of the flaw under normal operating conditions

$a_i$ = the minimum critical flaw size of the flaw for initiation of nonarresting growth under postulated emergency and faulted conditions.

## 3. Development of RViES System

In order to evaluate the integrity of reactor vessels more efficiently, RViES system has been developed. Figure 1 shows the structure of RViES system and Fig. 2(a) shows the sample screen of the system. Figures 2(b) and (c) show the sample screens for selection of PTS analysis and PT analysis respectively. RViES system consists of following three parts.

— User interface which provides the required

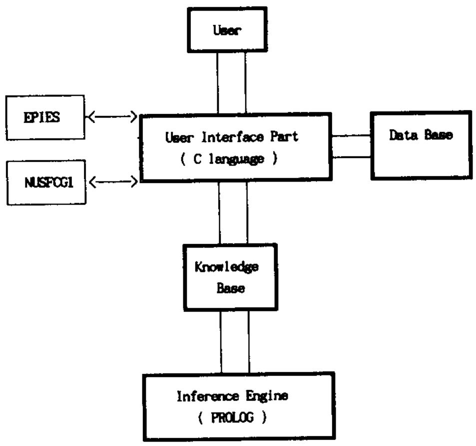


**Fig. 1** Structure of RViES system

knowledge through communication with users.
— Knowledge base which stores various rules for the integrity evaluation of reactor vessels.
— Inference engine which operates the knowledge base.

RViES system can be used in conjunction with EPIES(Kim et al., 1993), elastic-plastic fracture mechanics program and also with NUSFCG1 (Eun, 1991), fatigue analysis program.

### 3.1 User interface and knowledge base parts

In the user interface part, minimum amount of input data required for the analysis is received. In order to utilize the graphics variety, the program was developed in graphics mode using C-language(Schildt, 1986). For user's convenience, a mouse is connected with a keyboard during correction process and DOS shell(Schidlt, 1986) is added for multiple processes. In case the user does not know the input values, a modified knowledge is installed in the database by providing an empirical knowledge through selecting "Undo" function. The knowledge base is programmed by using PROLOG (Marcus, 1986) with a IF ~THEN~formula. It consists of four parts; USE, fatigue, PTS and P-T analysis.

### 3.2 Inference engine

Backward chaining method(Badiru,1992) was adopted in the inference engine. In backward chaining, the inference engine tries to verify a fact by finding rules that can prove the fact and then attempting to verify their premises. This enables the four analysis parts(USE, fatigue, PTS and P-T) to select the most appropriate rule by using existing and new knowledge bases and to infer from that rule. Computational part which is beyond the scope of PROLOG is processed in calculation part and stored in the knowledge base part.

1) USE Inference: Life prediction of a reactor vessel is performed in accordance with the rules in 10 CFR 50 (1987) Appendix $G$ as follows.

Rule 1 (USE, decrease ratio at $(1/4)t$($t$ is thickness of reactor vessel) and uncertainty are available): If the USE value is greater than 50 ft-lb(68 Joule), the fluence value corresponding to the decrease in the USE value is inferred. Subsequently, by considering the decrease ratio at $(1/4)t$ and the uncertainty in the measurement of fluence, the adjusted fluence value is obtained and the corresponding life is predicted.

Rule 2 (USE and decrease ratio at $(1/4)t$ are available): The uncertainty of fluence is taken as 25%, which is the maximum tolerable value, and the fluence and the predicted life of a reactor vessel are inferred.

Rule 3 (USE and uncertainty are available): The decrease ratio at $(1/4)t$ is stored in the knowledge base as 70%, and the fluence and the predicted life of a reactor vessel are inferred.

Rule 4 (USE is only available): The decrease ratio at $(1/4)t$ is stored in the knowledge base as 70%. The maximum tolerable uncertainty, 25%, is stored in the knowledge base by rule, and the fluence and the predicted life are inferred.

Rule 5 (USE is less than 50 ft-lb(68 Joule)): The decrease ratio at $(1/4)t$ and the uncertainty of the fluence are determined again and the predicted life is inferred. When the predicted life is larger than that corresponding to 50 ft-lb(68 Joule), an elastic-plastic fracture mechanics analysis is performed by EPIES(Kim et al., 1993).

2) Fatigue Inference: Since the fatigue life of a reactor vessel does not depend upon $RT_{NDT}$, the integrity of the reactor vessel can be assured through the comparison between the design fatigue life and the actual number of operating transients. The plant specific transient conditions are stored in the knowledge base and providing

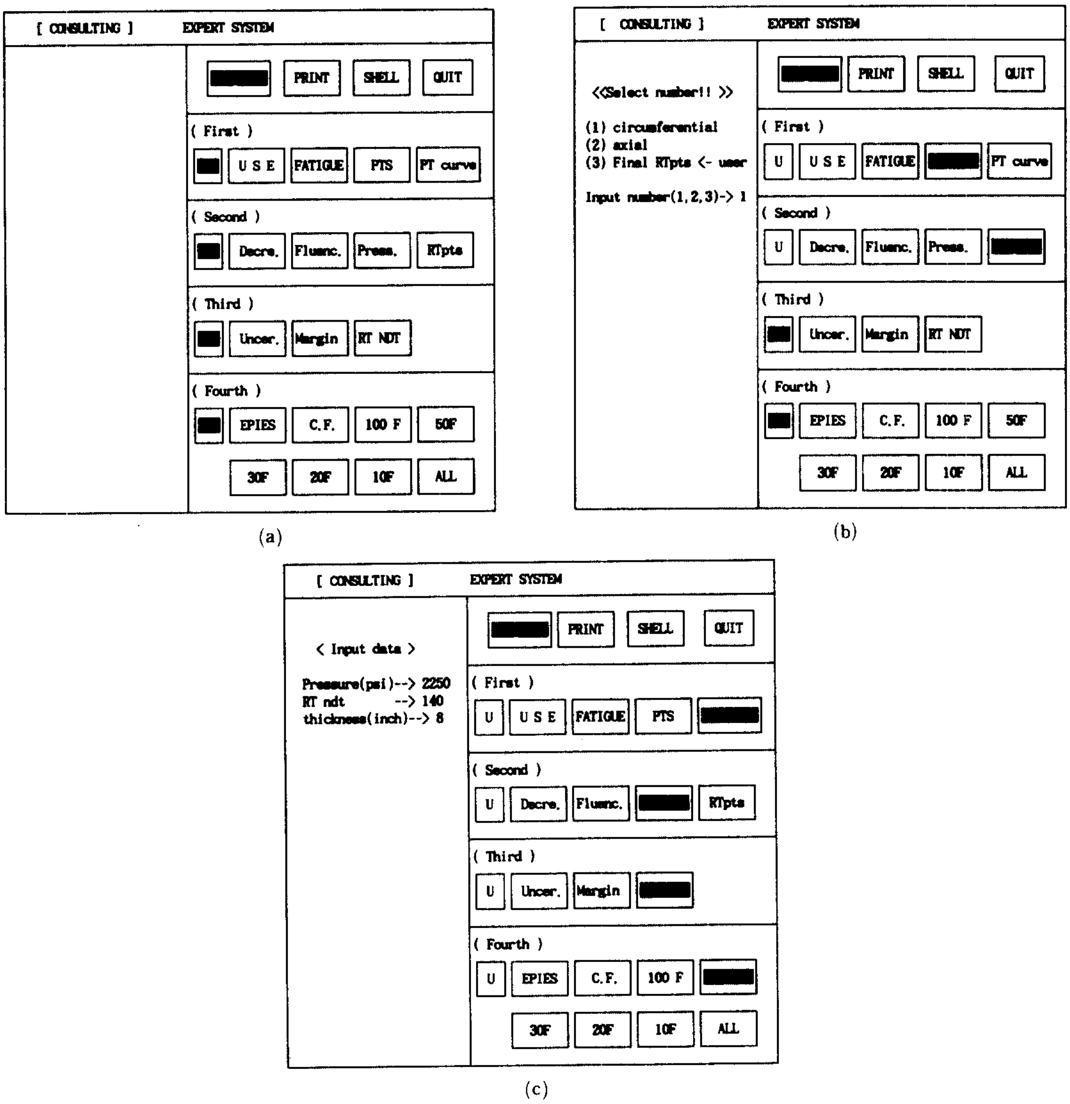


**Fig. 2** (a) Initial screen of RViES system
(b) Sample screen of PTS analysis part (circumferential weld)
(c) Sample screen of PT analysis part

that the cumulative fatigue damage($D$) is less than 1, the program is continued. If the $D$ value is larger than 1, then the program is stopped irrespective of other three inference parts and it is notified to the user.

3) PTS Inference : The PTS inference consists of following three forms :

— circumferential weld
— axial weld
— user supplied screening criteria

Life corresponding to 300°F(148.9℃) (circumferential weld) and 270°F(132.2℃) (axial weld) can be predicted from the following rules.

Rule 1 (PTS, fluence, margin and chemistry factor are available) : Initially the PTS criteria are determined corresponding to the weld form. Subsequently, $RT_{PTS}$ in Eq. (2) is determined and the corresponding life is predicted.

Rule 2 (Fluence is only available) : Values for margin(°F) and chemistry factor(°F) are determined by inferring the values stored in the knowledge base. Subsequently $RT_{PTS}$ in Eq. (2) is determined and the corresponding life is predicted.

Rule 3 (Fluence and margin are available) : Chemistry Factor(°F) value is determined by inferring the CF value(Table 1 and 2 in 10 CFR 50.61) stored in the knowledge base. Subsequently $RT_{PTS}$ in Eq. (2) is determined and the corresponding life is predicted.

Rule 4 (Fluence and chemistry factor are available) : The standard deviation value of initial $RT_{NDT}$ or $\Delta RT_{NDT}$ is added to the knowledge base. Subsequently, $RT_{PTS}$ in Eq. (2) is determined and the corresponding life is predicted.

Rule 5 (Fluence is unknown) : The user-supplied screening criteria are stored in the knowledge base and the fluence value is determined. Subsequently, $RT_{PTS}$ in Eq. (2) is determined and the corresponding life is predicted.

4) P-T Inference. : The P-T limit curve is generated for the cooling rate temperature of 10, 20, 30, 50, 100°F(−12.2, −6.7, −1.1, 10.0, 37.8℃). After providing input data such as pressure, thickness of the reactor vessel and $RT_{NDT}$, the $K_{IM}$ and $K_{IT}$ values are calculated and stored in the knowledge base. Finally P-T limit curve which satisfies Eq. (3) will then be drawn.

## 4. Case Studies

In order to check that the RViES system works properly, several case studies were performed.

### 4.1 USE problem

The objective of this case study was to verify the USE part of the RViES system. Initially, a life prediction was made for the unirradiated USE value of 66.3 ft-lb(90.17 Joule) to reach 50 ft-lb(68 Joule). By recognizing the uncertainties in fluence and USE measurement, the initial USE value can be regarded as 67.3 ft-lb(91.53 Joule). Figure 3(a) shows the predicted life of 8.37 effective full power year (EFPY)*, in good agreement with previously published KAERI report(Hong, 1988).

* EFPY is defined as actual power year for nuclear generating station.

Inference Engine

* Unirrdiated USE : 66.3(ft-lb)<br>
* Upper shelf Energy : 50 (ft-lb)<br>
* Decrease ratio at 1/4t : 0.7<br>
* Fluence at ID : 4.86*E18 (n/cm**2)<br>
* Uncertainty : 0.25<br>
* Predicted Life 8.37 EFPY

⟨Diagnosis⟩

The predicted life satifies 10 CFR 50 Appendix *G* fracture toughness requirements.

(a) USE=50 ft-lb

Inference Engine

* Unirrdiated USE : 66.3(ft-lb)<br>
* Upper shelf Energy : 49 (ft-lb)<br>
* Decrease ratio at 1/4t : 0.7<br>
* Fluence at ID : 4.86×E18 (n/cm**2)<br>
* Uncertainty : 0.25<br>
* Predicted Life 10.34 EFPY

⟨Diagnosis⟩

The USE value is less than 50 ft-lb, therefore an EPFM analysis is recommended.

(b) USE=49 ft-lb

**Fig. 3** USE case study

Inference Engine

* Delta $RT_{PTS}$ : 246.88°F<br>
* $RT_{PTS}$ : 299.88°F<br>
* The neutron fluence at ID<br>
→ 2.5091 (n/cm**2)×10**19<br>
* The neutron fluence at (1/4)$t$<br>
→ 1.7102 (n/cm**2)×10**19<br>
* Predicted Life : 25.63 EFPY

⟨Diagnosis⟩

$RT_{PTS}$ is less than screening criteria(300°F). Therefore the KORI Unit 1 is safe.

(a) Circumferential weld

Inference Engine

* Delta $RT_{PTS}$ : 246.88°F<br>
* $RT_{PTS}$ : 299.88°F<br>
* The neutron fluence at ID<br>
→ 1.4050 (n/cm**2)×10**19<br>
⟨The neutron fluence at (1/4)T⟩<br>
0.9495 (n/cm**2)×10**19<br>
* Predicted Life : 13.72 EFPY

⟨Diagnosis⟩

$RT_{PTS}$ is less than screening criteria(270°F). Therefore the KORI Unit 1 is safe.

(b) Axial weld

**Fig. 4** PTS case study

This case study was repeated for unirradiated USE value to reach 49 ft-lb(66.64 Joule). As shown in Fig. 3(b), the predicted life is increased from 8.37 EFPY to 10.34 EFPY.

**4.2 PTS problem**

The PTS part of RViES system was verified for the circumferential and axial welds. For the case of circumferential weld, the fluence value corresponding to the $RT_{PTS}$ of 300°F(148.9℃) is obtained, and the life prediction is made as shown in Fig. 4(a). For the case of axial weld, the fluence value corresponding to the $RT_{PTS}$ of 270°F(132.2℃) is obtained, and the life prediction is made as shown in Fig. 4(b).

**Table 1** P-T curve analysis results

| pressure (psi) | Temperature(°F) | |
|---|---|---|
| | RViES | USNRC |
| 450 | 140 | 162.7 |
| 1150 | 240 | 238.6 |
| 2250 | 298 | 296.4 |

**Table 2** Stress distribution for fatigue analysis

| Distance into wall (inches) | Group #2 | Primary leak | Primary hydro | Heatup cooldown | 100% load | Inadvert Aux.Spray | Group #1 | Reactor trip C |
|---|---|---|---|---|---|---|---|---|
| (Maximum stress distribution) | | | | | | | | |
| 0.0 | 42.6 | 29.2 | 36.8 | 38.6 | 26.5 | 73.2 | 48.6 | 47.9 |
| 0.15 | 39.7 | 29.1 | 36.6 | 32.9 | 26.5 | 63.2 | 45.0 | 44.1 |
| 0.47 | 33.8 | 28.9 | 36.3 | 21.4 | 26.3 | 43.4 | 37.7 | 36.3 |
| 0.78 | 29.3 | 28.7 | 36.1 | 11.9 | 26.1 | 27.3 | 31.8 | 30.3 |
| 1.16 | 25.9 | 28.4 | 35.7 | 4.2 | 25.8 | 15.5 | 27.0 | 25.8 |
| 1.63 | 23.6 | 28.1 | 35.3 | - 1.3 | 25.5 | 6.8 | 23.7 | 22.8 |
| 2.19 | 22.5 | 27.7 | 34.8 | - 3.8 | 25.2 | 3.2 | 21.9 | 21.4 |
| 2.81 | 21.9 | 27.3 | 34.3 | - 4.6 | 24.8 | 2.0 | 21.2 | 20.8 |
| 3.44 | 21.5 | 26.9 | 33.8 | - 4.8 | 24.5 | 1.6 | 20.8 | 20.4 |
| 3.75 | 21.3 | 26.7 | 33.6 | - 4.9 | 24.3 | 1.5 | 20.6 | 20.2 |
| (Minimum stress distribution) | | | | | | | | |
| 0.0 | 2.3 | 0.0 | 0.0 | -21.2 | 26.0 | 8.9 | 19.2 | 17.0 |
| 0.15 | 2.0 | | | -16.6 | 25.9 | 10.5 | 20.4 | 17.5 |
| 0.47 | 23.4 | | | - 7.4 | 25.7 | 13.4 | 22.9 | 18.6 |
| 0.78 | 24.3 | | | - 0.5 | 25.5 | 15.7 | 24.6 | 19.4 |
| 1.16 | 25.0 | | | - 4.4 | 25.2 | 17.1 | 25.5 | 20.0 |
| 1.63 | 25.2 | | | 7.2 | 24.9 | 17.3 | 25.2 | 20.2 |
| 2.19 | 25.0 | | | 7.7 | 24.6 | 15.8 | 23.7 | 20.1 |
| 2.81 | 24.7 | | | 7.2 | 24.2 | 13.4 | 21.7 | 19.7 |
| 3.44 | 24.4 | | | 6.7 | 23.3 | 11.8 | 20.4 | 19.4 |
| 3.75 | 24.2 | | | 6.4 | 23.7 | 11.0 | 19.7 | 19.2 |

**Table 3** Number of operating transients

| | No. of transients |
|---|---|
| Group #2 | 11874 |
| Primary leak | 60 |
| Primary hydro | 2 |
| Heatup & cooldown | 240 |
| IAS* | 2 |
| Group #1 | 920 |
| Reactor trip C | 2 |

* Inadvert Aux. Spray

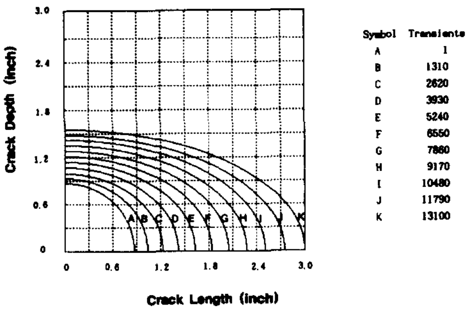


**Fig. 5** Behavior of crack extension in fatigue analysis

### 4.3 P-T limit curve problem

The P-T curve part of RViES system was verified using the following input data in USNRC report(1981).

- operating pressure : 2250 psi(15.52 MPa)
- vessel thickness : 9″
- $RT_{NDT}$ : 140°F(60℃)

The $K_{IM}$ and $K_{IT}$ values were obtained by RViES system and Table 1 shows the calculated temperature corresponding to the cooling rate of 50°F(10℃), in good agreement with USNRC report(1981).

### 4.4 Fatigue problem

It was not possible to obtain information related to fatigue crack growth analysis of operating reactors. Therefore, in order to verify the fatigue part of RViES system, fatigue analysis of pressurizer weld defect in Yonggwang nuclear power plant(Eun, 1991) was used. Table 2 shows the circumferential stress distribution and Table 3 shows the number of transients expected during 8 years in service. These numbers were extracted from the Yonggwang fracture analysis report (Bamford, et al., 1989).

Initial flaw geometry(semi-elliptic shape) was assumed as a(crack depth)=0.84″(21.43 mm) and l(crack length)=1.69″(42.86 mm). Fatigue analyses results show that, after 8 years in service, the initial flaw grew to $a=1.51''$ (38.33 mm) and $l=2.86''$ (72.69 mm) which still satisfy Eq. (4). As shown in Fig. 5, the flaw geometry is maintained as semi-elliptic shape which can also be observed experimentally.

## 5. Conclusion

A prototype expert system called RViES was developed for the reactor vessel integrity evaluation. The knowledge base was programmed to deal with various problems such as radiation embrittlement, pressurized thermal shock and fatigue for the reactor vessel. The knowledge base and inference engine were programmed by using PROLOG and *C* languages. The usefulness of RViES system was verified by several case studies.

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