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Evaluation of Direct Vessel Injection Design With Pressurized Thermal Shock Analysis

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가압 열충격해석에 의한 직접용기주입 설계의 평가

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Abstract

The purpose of this paper is to evaluate the direct vessel injection design from a pressurized thermal shock(PTS) viewpoint for the Combustion Engineering System 80+. A break of the main steam line from zero power and a 0.05 ft^2 small break loss-of-coolant accident (LOCA) from full power were selected as the potential PTS events. In order to investigate the stratification effects in the reactor downcomer region, the fluid mixing analysis was performed using the COMMIX-IB code for steam line break and using the REMIX code for 0.05 ft^2 small break LOCA. The stress distributions within the reactor vessel walls experiencing the pressure and the temperature transients were calculated using the OCA-P code for both events. The results of the analysis showed that a small break LOCA without decay heat presented the greatest challenge to the vessel, however, there is no crack initiation through end-of-life of the vessel with consideration of decay heat.

요 약

이 논문의 목적은 C-E System 80+ 원자로에서의 직접용기주입 설계를 가압 열 충격의 견지에서 평가하는 것이다. 영의 출력에서의 주증기관 파단과 0.05ft^2 면적의 소형파단 냉각재상실사고가 가능성있는 가압열충격 사고로 선정되었다. 원자로 다운카머 영역에서의 유체 성층효과를 예측하기 위하여 주증기관 파단사고에 대하여는 COMMIX-IB 전산코드를, 그리고 0.05ft^2 소형파단 냉각재상실사고에 대하여는 REMIX 전산코드를 사용하여 유체혼합해석이 수행되었다. 압력과 온도의 과도변화를 받는 원자로용기 벽내의 응력분포는 두 사고에 대하여 OCA-P 전산코드를 사용하여 계산되었다. 해석결과, 붕괴열의 고려가 없는 소형파단 냉각재 상실사고의 경우 용기내 균열발생의 가능성이 있으나 붕괴열을 고려하면 용기의 수명기간중 균열발생의 가능성은 없다.

1. Introduction

For most existing pressurized water reactor(PWR) plants, the safety injection connects to the cold legs. Thus, for the cold leg breaks, a portion of the safety injection flow may be forfeited by flowing directly to the break. This loss of safety injection fluid can be reduced by having the safety injection lines connect directly to the reactor vessel. In this regard, the EPRI ALWR Requirement Document specifies that the safety injection system(SIS) shall discharge directly into the reactor vessel downcomer region rather than into the cold legs. Use of direct vessel injection(DVI) rather than cold leg injection may reduce the safety injection pump capacity requirements for the cold leg break as well as the safety injection tank volume. In accordance with the EPRI Requirement, the emergency core cooling system(ECCS) in the Combustion Engineering(C–E) System 80+, 3800 MW thermal power of a PWR, employs four trains of high pressure safety injection pumps and the safety injection tanks which inject directly into to vessel annulus instead of cold leg injection as described in the later section.

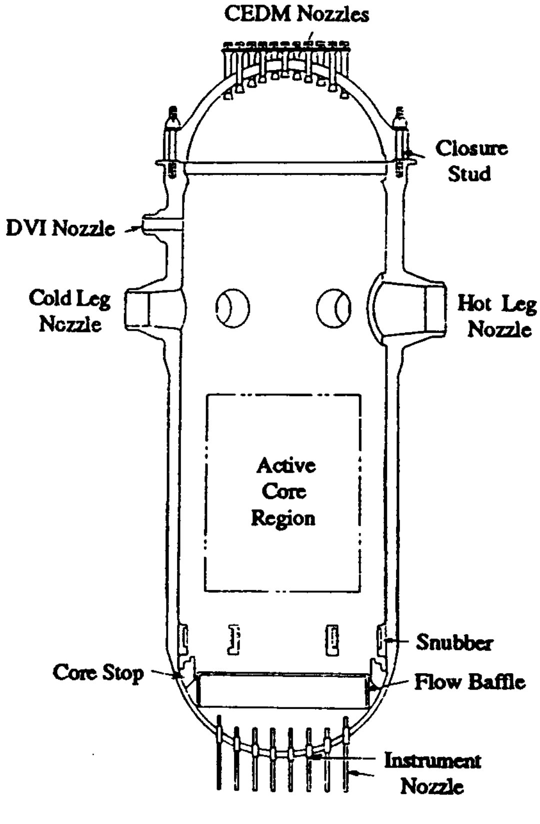
Under certain postulated accident conditions such as small break loss–of–coolant accident(LOCA), main steam line break, and other overcooling scenarios, the reactor pressure vessels of PWR undergo a large cooling rate keeping significantly high internal pressure. A pressurized thermal shock(PTS) event is characterized by injection of cold water into the reactor vessel along with a relatively high reactor coolant system pressure. The injection of cold water induces a thermal stress in the reactor vessel wall when combined with the mechanical stress of a relatively high reactor coolant system pressure. The combination of such thermal stress and internal pressure could pose a serious challenge to the integrity of the reactor vessel. This is particularly true at the elevation corresponding to the top of the core where the neutron fluence of the vessel wall is greatest. In connection with the integrity of the reactor vessel, the design of the direct vessel injection concept should be evaluated with the PTS analysis.

The US Nuclear Regulatory Commission has sponsored a comprehensive program to evaluate the likelihood, consequences and risk of various PTS events for three nuclear power plants, e.g., the Calvert Cliffs Unit 1 [1]. A detailed method of performing such plant specific analyses is being developed in order to provide appropriate guidance to the nuclear industry. The Electric Power Research Institute(EPRI) has also sponsored an extensive program to investigate the PTS issues [2,3]. However, above investigations are mostly concerning cold leg injection . There is no reporting on the PTS analysis relating with DVI.

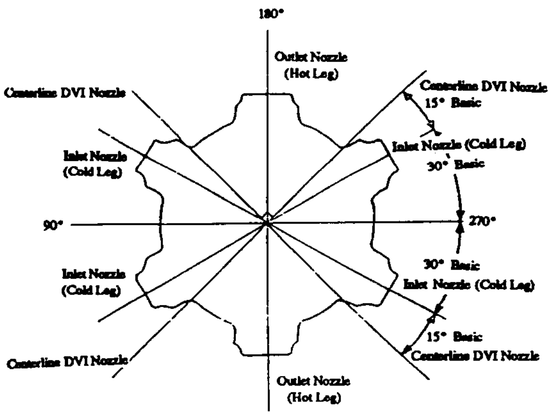
The purpose of this paper is to provide the evaluation result of integrity analysis of DVI design from a PTS viewpoint based on current C–E System 80+ design as an Advanced Light Water Reactor(ALWR) design.

2. DVI Design Description

Figure 1(a) shows the C–E System 80+ reactor vessel with the proposed design feature of DVI nozzles [4]. There are four injection nozzles located around the reactor vessel as shown in Figure 1(b) [5]. These nozzles are located at a 45° offset from the hot leg centerlines and thus are located at a 15° offset from the cold legs. The four DVI nozzles are located equidistantly about the reactor vessel(i.e., at 90° angles), midway between each loop's pair of cold leg nozzles. The nozzle elevation was chosen as close as possible to the reactor vessel flange weld based on structural, fabrication, and in–service inspection considerations. DVI nozzles shall not adversely affect refueling operations, reactor head area maintenance, in–service inspec-



(A) Reactor Vessel with DVI Nozzle



(b) Location of DVI Nozzles

Fig. 1. C–E System 80+ Reactor Vessel with DVI

tion of the reactor vessel or reactor coolant loop nozzles including cavity and component support cooling. This requirment assures the core remains covered with water during maintenance of the process systems, provides the maximum mixing distance between cold injection fluid entering the vessel and the vessel beltline region, and assures adequate space is provided for normal maintenance. Location of DVI nozzles will satisfy minimum distance requirements from existing nozzles for reinforcing and welds. Each nozzle has a 8.5 inch inner diameter(ID) compatible with 10 inch schedule 160 stainless steel pipe. Use of a 8.5 inch ID nozzle results in acceptable shutdown cooling velocities and ECCS performance, and allows use of current leak–before–break methodology.

3. Procedure of Pressurized Thermal Shock Analysis

The PTS analysis consists of the event sequence analysis, thermal hydraulic analysis, and fracture mechanics analysis. The first step of the event sequence analysis is the development of a set of system state trees to describe the potential conditions of important reactor systems. The next step is the identification of specific initiating events which could lead to overcooling transients, and then to examine the system operating states with respect to the initiating events. Finally, expected frequency of each event–tree transient is calculated based on the plant data and generic failure data. The calculated frequencies and engineering judgement are then used to develop a final list of sequences to be considered in subsequent thermal hydraulic and fracture mechanics analyses. In this stage, Combustion Engineering selected a double–ended break of main steam line from zero power and a break size of $0.05ft^2$ cold leg small break LOCA from full power as the potential PTS events for the C–E 80+ with DVI[6].

The thermal–hydraulic analysis for PTS evaluation is divided into two parts. The first part is the system analysis for predicting thermal hydraulic responses of the plant to the events, and the second part is to determine local fluid temperature in the downcomer by fluid mixing analysis with existing engineering models. The transient thermal hydraulic response to selected events are analyzed using the CESEC code [7] for a steam line break, and using the CEFLASH code [8] and the COMPERC–II code [9] for a small break LOCA. Reactor coolant system pressure, fluid flow, and temperature histories obtained from these codes are required as boundary conditions for fluid mixing analysis in the downcomer and fracture mechanics analysis of the reactor vessel wall. The mixing of cold injection water and the warm fluid in the downcomer is of interest in PTS scenarios. Insufficient mixing could result in local thermal stratification in the downcomer. The three–dimensional mixing code COMMIX–1B [10] was used to determine the local temperature in the downcomer for a steam line break, and the REMIX computer program [11] developed on the basis of the regional mixing model was used for a small break LOCA.

The fracture mechanics analysis is to calculate the transient stress distributions within the reactor vessel walls receiving the pressure and the temperature transients from the preceding processes. Following this, fracture mechanics parameters such as the stress intensity factor and fracture toughness are evaluated to examine the behavior of preexisting cracks in this stress field. The OCA–P code [12] was used to perform the fracture mechanics analysis with PTS.

4. Thermal–Hydraulic Analysis

4.1. Steam Line Break

In a steam line break event, the reactor coolant system is rapidly depressurized until the safety injection starts as a result of the blowdown of the ruptured steam generator to atmospheric pressure, and then is repressurized to near the shutoff head of the safety injection pumps. The bulk temperature of the reactor coolant system decreases rapidly. Cold safety injection water is injected into the reactor downcomer. Reactor vessel downcomer annulus region remains full of liquid. The analysis assumed the ECCS configuration which consists of four hydraulic trains and two electrical trains where each train injects the safety injection water directly into the reactor vessel downcomer. Each train is to consist of a high pressure safety injection(HPSI) pump and a safety injection tank. A loss of off–site power and 60 seconds diesel start/load time were assumed. It is assumed that both diesels and all four HPSI pumps start normally. This maximizes the flow rate of cold safety injection water and thus contributes to a cooldown of the fluid in the downcomer annulus. It is also assumed that 1000 GPM of emergency feed water flows into each steam generator without operator action to isolate the ruptured steam generator. This maximizes the reactor coolant system cooldown.

Figure 2 shows the reactor coolant system pressure transient during a steam line break which was analyzed using CESEC code. The pressure drops to about 7.24 MPa but recovers to near the shutoff head of the safety injection pumps (more than 12.24 MPa) in less than 10 minutes. Figure 3 shows the safety injection flow rate for a steam line break event. The safety injection flow rate is about 182 kg/s starting at 60 seconds and gradually decreases to about 22.7 kg/s by 720 seconds due to increase of the primary pressure. The bulk temperature in the right hand downcomer annulus which is connected to the ruptured steam generator stabilizes at 104℃ after about 20 minutes as shown in Figure 4. The decay heat is sufficient to heat the emergency feed water to the atmospheric boiling temeprature, therefore, the liquid temperature in the downcomer would not fall below 100℃ within two hours. Figure 5 shows the reactor coolant flow rate during a steam line break. Because the ruptured steam generator does not boil dry, it continues to steam and remove heat from the reactor coolant system. Therefore, natural circulation is maintained as shown in Figure 5. On the other hand, the intact steam generator is rapidly cooled by emergency feed water flow and pressure never rises to the secondary relief setting. For this reason, the intact steam generator quits steaming which results in a loss of natural circulation flow.

Since system code could not predict the stratification effects in the reactor downcomer region, it was felt that a fluid mixing calculation was necessary to determine wall temperature from a PTS viewpoint. The fluid mixing in the downcomer with DVI has been analyzed using the

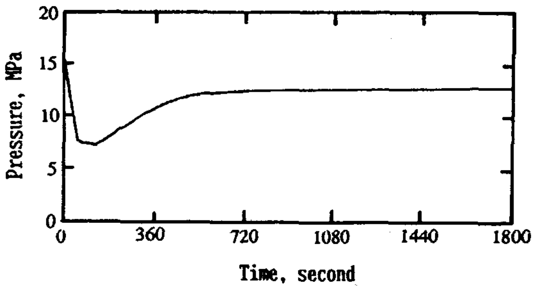


Fig. 2. System Pressure Transient during Steam Line Break

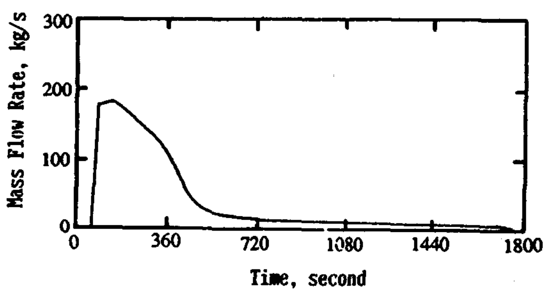


Fig. 3. Safety Injection Flow Rate during Steam Line Break

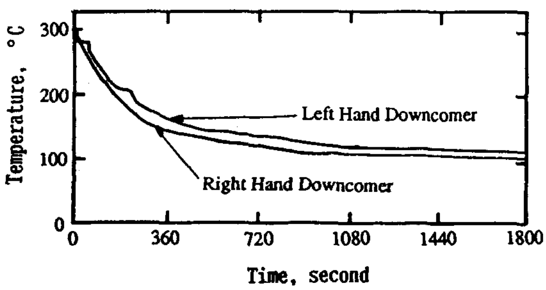


Fig. 4. Downcomer Fluid Temperature Transient during Steam Line Break

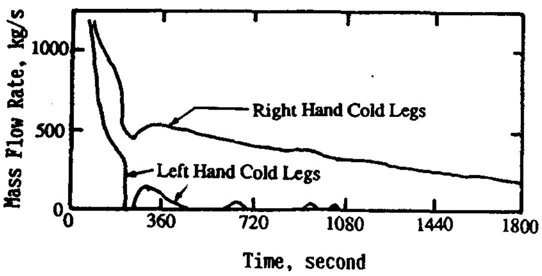


Fig. 5. Loop Flow Rate during Steam Line Break

COMMIX-1B code for a steam line break. The capability of COMMIX-1B includes steady state/transient, three-dimensional, and single-phase heat transfer and fluid flow analysis of nuclear reactor systems under normal and off-normal operating conditions. This code provides detailed local velocity and temperature fields for the problems under consideration. In the calculation, the one-equation turbulence model and the fully implicit scheme were employed.

A nodal model representing a 90° segment was chosen to simulate one of the four regions of symetry in the downcomer flow field. Figure 6 shows the nodal division of the COMMIX-IB model for the C-E System 80+. The nodal grid contains a total of 1070 computational cells. The model has 4 cells in the x(I)-direction, 17 cells in the y(J)-direction, and 23 cells in the z(K)-direction. Boundary conditions for the fluid mixing transient analysis performed were taken from the results of a steam line break transient analyzed using the CESEC code. The temperature of a safety injection water was assumed to be 10℃. The decay heat in a steam line break was examined using the ANS decay heat curve for a hot zero power condition at 100 hours following a reactor trip. The decay power would be about 10 MWt over a two-hour transient period. It was assumed that the majority of the decay heat was dissipated by the emergency feed water added to the steam generator. For the conservative calculation, safety injection flow rate was assumed to be constant of 180 kg/s.

Figure 7 shows the right-hand downcomer temperature transient near the reactor vessel wall at the top level of active core for a steam line break. The rapid temperature decrease from 300℃ to 100℃ was observed for about 5 minutes. After this, the temperatures were stabilized at about 85℃(185℉).

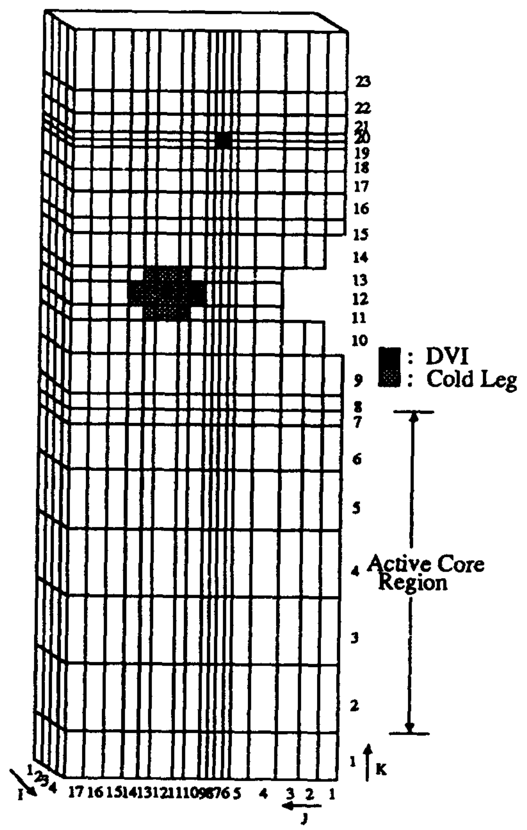


Fig. 6. Nodal Division for COMMIX-1B Model

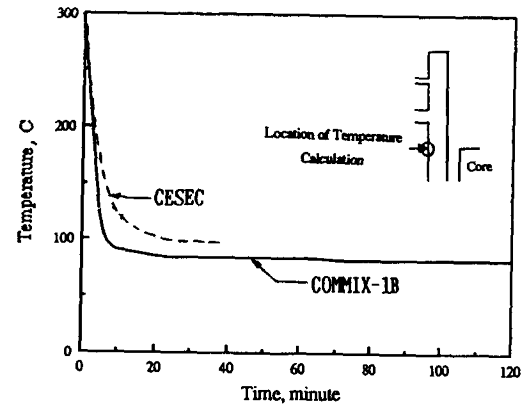


Fig. 7. Downcomer Temperature Transient Calculations for Steam Line Break

4.2. Small Break Loss-of-Coolant Accident

Generally, LOCA conditions may become important because of their potential stagnation loop flow conditions that could produce a significant cooldown. In this transient, the reactor coolant system depressurizes rapidly until the safety injection starts and the pressure is maintained fairly high for a long period. Cold safety injection water is injection into the downcomer. However, the coolant in the downcomer annulus remains at a lower level.

For a break size of $0.05ft^2$ small break LOCA analysis, the CEFLASH-4AS code was used to determine the primary system hydraulic parameters during the blowdown phase, and the COMPERC-II code was used to determine the system hehavior during the reflood phase[13]. According to the results of the system analysis for a $0.05ft^2$ small break LOCA, the primary system pressure is remained at a relatively high value(approximately 8.96 MPa) as shown in Figure 8. The reason that the primary pressure stays high is that the large safety injection flow rate prevents the break from uncovering and the safety injection flow rate at least equals the break flow rates at approximately the secondary pressure level. The coolant in the vessel may be assumed to be at the saturation temperature corresponding to the system pressure. The coolant level in to reactor vessel downcomer is remained at about 1.52 m above the top of the active core as shown in Figure 9. This provides a relatively small height of water with warmer downcomer water before reaching the beltline region. Figure 10 shows the results of the analysis for the break flow rate and the safety injection flow rate, respectively. The safety injection flow rate is 136 kg/s starting at 140 seconds and increases to 150 kg/s by 300 seconds and remains at about that magnitude. After 900 seconds, the safety injection flow rate approximately equals the break flow rate and the two flow rates balance each other from then on.

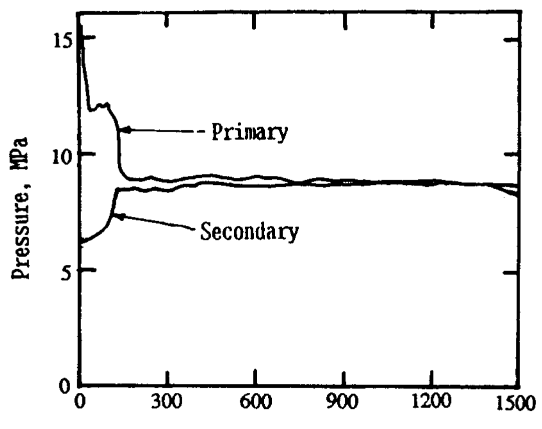


Fig. 8. System Pressure Transient during 0.05ft^2 Small Break LOCA

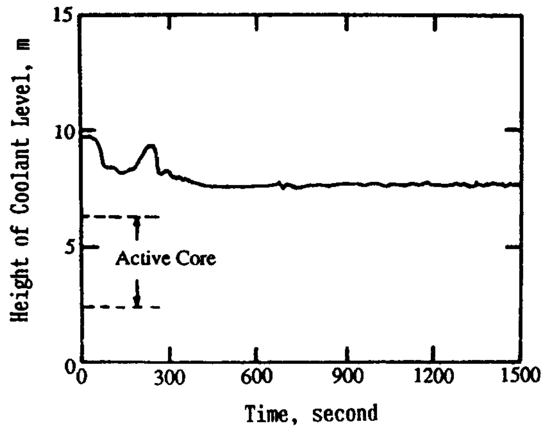


Fig. 9 Coolant Level during 0.05ft^2 Small Break LOCA

In order to observe the physical interaction between DVI water and existing coolant in the downcomer during a cooldown transient, the separate flow visualization tests[14] were conducted in a 1/5-scale transparent model of a C-E system 80+ reactor geometry. The test model was represented by a planar section having width, height and gap comparable to a 90° sector of a reactor downcomer. Dye injection within the test model was used to trace the flow patterns. Flow conditions of DVI were chosen to simulate the Froude number(Fr=1.05~1.50) of prototype reactor coolant with 4.4 wt% salt concentration at room temperature. Figure 11 shows a typical flow pattern of DVI visualization test under the condition of a small break LOCA. It can be seen in this figure that flow patterns exhibit two distinct regions : "plume flow" region, and "mixing developed" region. During the initial part of the flow transient, the flow in the downcomer is fromed from the strong plume flow. Following this the plume flow is developed toward the mixing developed region which has relatively higher mixing rate. It appears that the vessel beltline region was exposed with a relatively well mixed situation.

The mixing pattern associated with DVI into a reactor downcomer was analyzed using the REMIX code which was developed on the basis of the regional mixing model(RMM). RMM provides a phenomenologically-based analytical description of the stratified and temperature fields resulting from high pressure safety injection in the stagnated loops of a PWR. The REMIX code was originally designed for cold leg injection. Therefore, the code was appropriately modified for DVI application. The key of modification was to substitute the DVI nozzle for the cold leg in the calculation. It was used a fictitious DVI line in the prog-

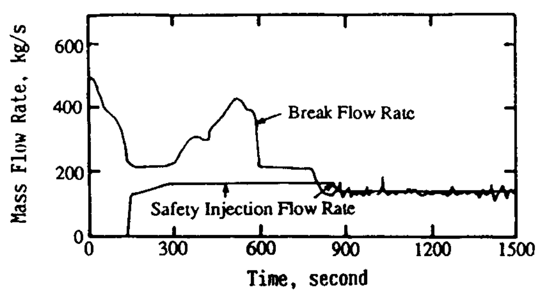


Fig. 10. Break and Safety Injection Flow Rates during 0.05ft² Samll Break LOCA

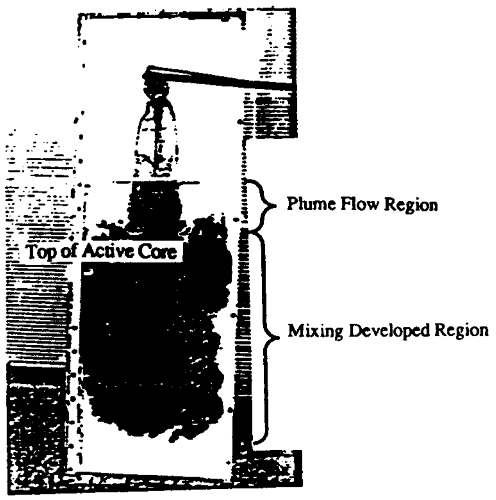


Fig. 11. Typical Flow Pattern of DVI

ram equal in diameter to that of the cold leg. It was also set cold stream temperature equal to DVI temperature, and cold stream hight equal to DVI nozzle diameter.

To verify the REMIX code with experiments, another separate fluid mixing tests were performed in a 1/5–scale steel fabricated model of a C–E System 80+ reactor vessel[15]. All ratios of flow path dimensions and all angles of test model were preserved with prototype. The tests were conducted for loop and DVI flow conditions covering ranges of interest to the issue of PTS at atmospheric pressure. During the tests, Froude number of prototypical DVI flow(about 1.30) was preserved. The temperatures of DVI flow and the initial temperature of coolant in the downcomer were kept at 10℃ and 94℃, respectively. Figure 12 shows a typical temperature transient measurements under the conditions of 0.33 kg/s DVI flow rate (Fr=1.27) and stagnant loop flow. the measurements were agreed with the predictions obtained using the REMIX code. The measured temperature transient compared well with the predictions.

The transient temperature profiles for a 0.05ft² small break LOCA with stagnated loop flow were

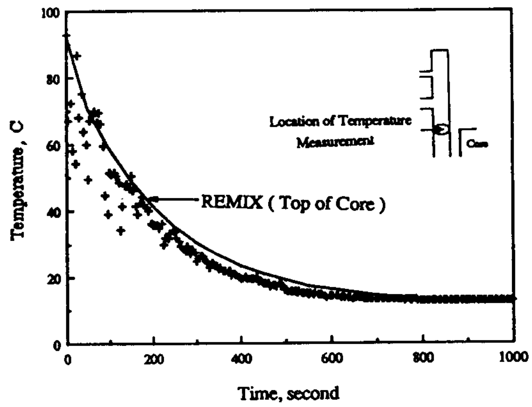


Fig. 12. Typical Temperature Measurements during Cooldown Transient

obtained with the REMIX code under the assumption that stagnation began at time zero. It was assumed that the temperature of safety injection water was 10℃, and that initial temperature of coolant in the downcomer was 300℃. A solid line in Figure 13 shows the downcomer fluid temperature transient at the top elevation of the active core for a 0.05ft² small break LOCA without consideration of the decay heat. A rapid decrease occured from 300℃ to 93℃ during the first 10 minutes and the lowest temperature of 18.3℃ (65°F) was calculated after 120 minutes. This temperature profile is compared with the well mixing curve as shown in the figure. The well mixing temperature curve was generated by the following equation :

$$T=T_{DVI}+(T_0-T_{DVI})\exp(-t/\tau_s)$$

$$\tau_s=V_e/Q_{DVI}$$

and

where T_{DVI}=DVI water temperature, To=initial loop bulk temperature, τ_s=characteristic mixing time, t=time, V_e=effective mixing volume, and Q_{DVI}=total DVI volumetric flow.

A dotted line in Figure 13 shows the transient temperatures at the top elevation of the core for a small beak LOCA with taking into account the core decay heat. It was assumed that about 30 MWt remains as decay heat at the end of two hours after reactor trip. The lowest temperature of 55.5℃(135℉) was achieved after 120 minutes.

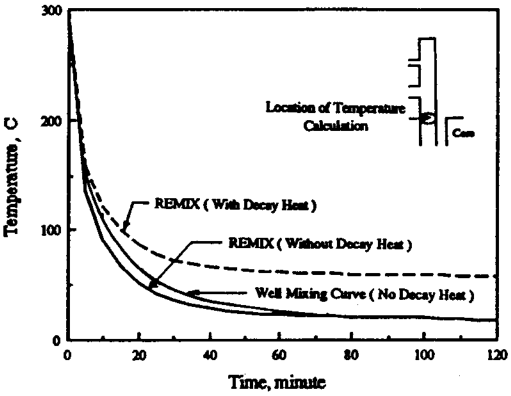


Fig. 13. Downcomer Temperature Transient Calculations for Small Break LOCA

5. Fracture Mechanics Analysis

The fracture mechanics analysis was performed using the OCA-P code. This code is a computer program that performs a linear-elastic fracture mechanics analysis for long axial inner-surface flaws in a cylinder subjected to time-dependent thermal and pressure loadings. The OCA-P calculates the wall temperatures and stresses as a function of time and radial position in the wall. A fracture mechanics analysis is then performed to obtain the stress intensity factors as a function of crack depth and time in the transient. Values of the static crack initiation toughness and the crack arrest toughness are also calculated as a part of the analysis for various crack depths and times in the transient. A comparison of the stress intensity factor with crack initiation toughness and crack arrest toughness permits an evaluation of flaw behavior.

Some assumptions for the analysis have heen made as follows. The assumed chemical composition of the vessel material is 0.05 wt% Cu and 1.0 wt% Ni[16]. The initial nil-ductility temperature(RT_{NDT}) of the vessel material is estimated to be −30℃(−22℉)[16]. It is assumed that the vessel is constructed using a large ring forging, with no welds in the beltline region. The inner diameter of the vessel is 462.91 cm, and the thickness of the vessel wall is 23 cm. The PTS event is assumed to take place at the end-of-life after accumulating 6×10^{19} n/cm^2 (40 years reactor operation). A transient is considered acceptable if it results in no extension of a preexisting crack at any depth less than 50% through the vessel wall.

The output of the results from the fracture mechanics calculation includes : (1) stress intensity factor(K_I) versus time for assumed crack depths, (2) the ratio of stress intensity factor(K_I) to crack initiation toughness(K_{IC}) versus time, and (3) a critical crack depth versus time curve. The stress intensity factor, K_I, is due to contributions from thermal stresses, pressure stresses and other stress that may be present. K_{IC} is the vessel toughness that determines crack initiation. K_{Ia} is the vessel toughness at crack arrest. These values of toughness can be calculated as a function of time in the transient and radial position in the wall. Crack initiation is expected when K_I exceeds K_{IC}. The crack would then grow to a depth where K_I intercepts the crack arrest curve, K_{Ia}. Thus, crack propagation can be evaluated on the basis of the K-ratios at the deepest point of the flaw. The critical crack depth curve depicts the range of critical crack depths for conditions of crack initiation and/or crack arrest at a given fluence level. The curves consist of plots of crack depths corresponding to various events and conditions as a function of the times at which the events and conditions take place. The behavior of flaws can be evaluated with these curves.

In the fracture mechanics calculation, the period of analysis was extended to two hours. Since most of thermal–hydraulic analyses here were terminated at a shorter time period, therefore, these analyses were extrapolated to two hours based on previous calculation. It was felt that a two–hour period is sufficient time to reverse any overcooling trends. The results of fracture mechanics analysis for three cases : a steam line break, a small break LOCA without decay heat, and a small break LOCA with decay heat, for a C–E system 80+ are given in the following.

A stress intensity factor at the depth of 50% through the vessel wall versus time plot for above three cases is shown in Figure 14. Peak stress intensity factors of all above three cases occur at about 20 minutes into the transient and decrease thereafter.

A measure of the margin against crack initiation at the depth of 50% through the wall for above three cases is shown in Figure 15, in terms of the ratio of K_I/K_{IC}. The maximum K_I/K_{IC} ratio for a steam line break at the end–of–life fluence level was calculated to be about 0.30 of the critical level to cause crack initiation. Crack initiation will not occur for a long axial crack until the RT_{NDT} reaches 154°F which corresponds to a fluence level of 6.0×10^{19} n/cm^2. The K_I/K_{IC} ratio for a small break LOCA without decay heat was calculated to exceed 1.0 after 60 minutes. This value is exceeding the critical level to cause crack initiation. The maximum ratio K_I/K_{IC} for a small break LOCA with decay heat is calculated to be 0.78 of the critical level at the time of 110 minutes.

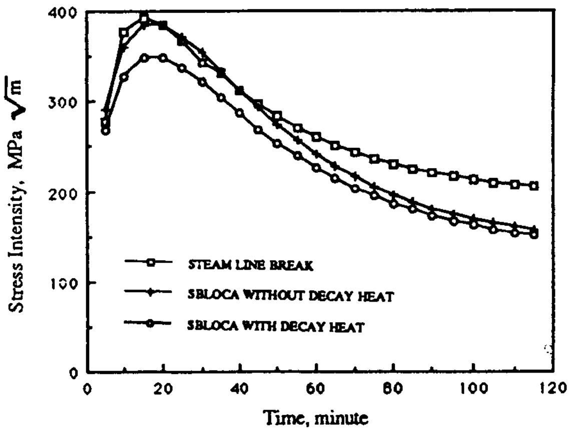


Fig. 14. Stress Intensity Factor vs. Time at 50% Depth of Wall

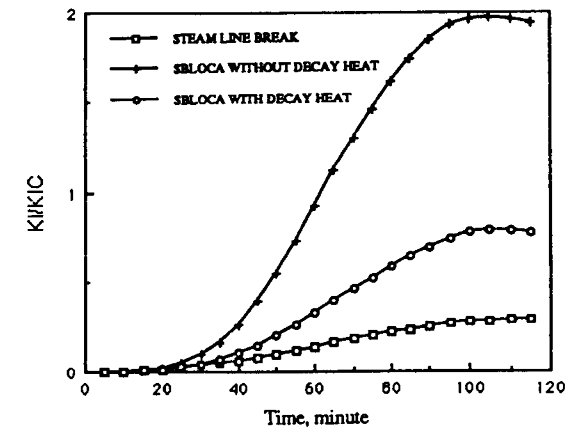
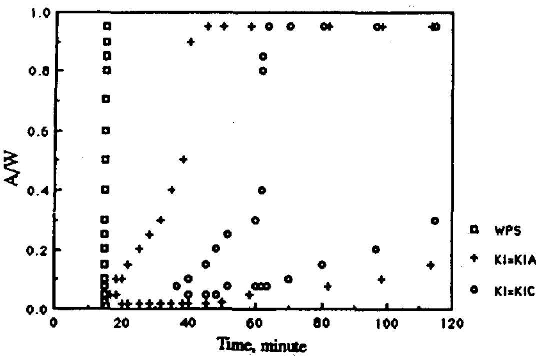
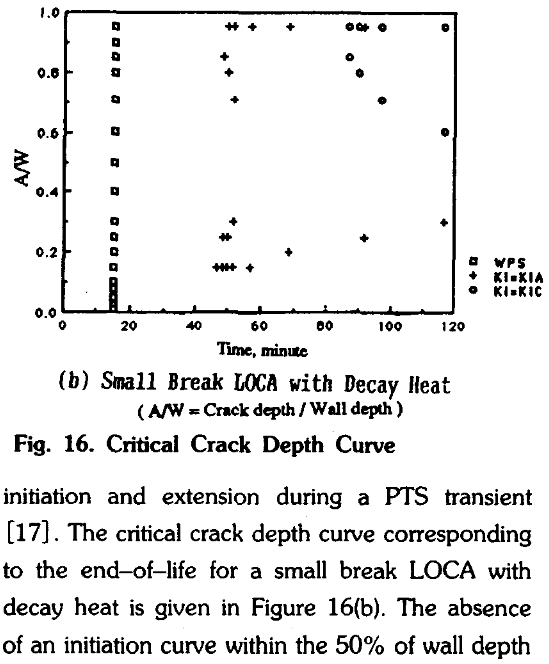


Fig. 15. K_I/K_{IC} vs. Time at 50% Depth of Wall

The critical crack depth curve to the end-of-life for a small break LOCA without decay heat is given in figure 16(a). The shallower flaw would initiate at a time of 37 minutes after the beginning of the transient and would arrest at a point 40% of the way through the wall. And then would reinitiate at a time of 60 minutes and then penetrate the vessel wall. In this transient, two things can be considered. One is that there is sufficient time to take a measure to prevent the crack initiation before the ratio of K_I/K_{IC} exceeds 1.0. The other thing is that warm prestressing(WPS) effects take place about 15 minutes after transient starts. Warm prestressing can theoretically prevent crack



(a) Small Break LOCA without Decay Heat
(A/W = Crack depth / Wall depth)



(b) Small Break LOCA with Decay Heat
(A/W = Crack depth / Wall depth)

Fig. 16. Critical Crack Depth Curve

initiation and extension during a PTS transient [17]. The critical crack depth curve corresponding to the end-of-life for a small break LOCA with decay heat is given in Figure 16(b). The absence of an initiation curve within the 50% of wall depth in this figure indicates that there is no range of crack depths that could produce crack initiation within the depth of 50% through the vessel wall.

6. Conclusions

Through a series of analysis, the following conclusions have been made :

1. There would be no crack initiation throughout the life of the plant due to a steam line break from zero power.

2. A 0.05ft^2 small break LOCA from full power without consideration of decay heat dominates the PTS risk for the plant. This domination is due to high primary pressure and stagnation in the primary loop. However, there is no crack initiation through end-of-life of the plant with consideration of decay heat.

3. The concept of direct vessel injection for the C-E System 80+ is adoptable from a PTS viewpoint. The above PTS events do not pose a significant safety concern to the public for at least 40 years of operation.

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