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Design Enhancements of Automatic Depressurization System in a Passive PWR

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피동형 경수로 자동감압계통의 개선에 관한 연구

유성식
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Abstract

In a Passive PWR, the successful actuation of Automatic Depressurization System (ADS) is essentially required so that no core damage is occurred following small LOCA. But it has been shown in the previous studies that Core Damage Frequency (CDF) from small LOCA is significantly caused by unavailability of ADS. In this study, the design vulnerabilities impacting the ADS unavailability have been identified and the design improvement items have been proposed through the system reliability assessment using the fault tree methodology. The impacts on CDF according to the change of system unavailability have also been analyzed. In addition, small LOCA simulation using RELAP5/MOD3 code has been performed to show the thermal-hydraulic feasibility of the suggested design enhancements.

요 약

피동형 원자력발전소의 설계 특성상 소형 냉각재상실사고시 노심손상이 발생되지 않기 위해서는 자동감압계통의 성공적인 작동이 필수적으로 요구된다. 그러나 기수행된 연구들에서 자동감압계통의 비신뢰도가 소형 냉각재상실사고로 부터 기인되는 노심손상빈도에 상당 부분을 기여하고 있음을 알 수 있다. 본 연구에서는 자동감압계통의 불능도에 기여하는 계통의 취약점을 파악함과 함께 계통의 신뢰도를 증대시키기 위한 설계개선 방안들을 제시하고 각 방안에 대한 신뢰도 분석과 함께 열수력학적 타당성 여부를 보기 위한 소형 냉각재상실사고 모의가 RELAP5/MOD3 전산코드를 사용하여 수행되었다. 신뢰도 분석은 고장수목 기법을 이용하여 수행되어졌다.

1. Introduction

The main improvements on the safety of the present water reactor nuclear power plant are to adopt passive safety systems and to enhance the operating reliabilities of equipment and systems to reduce the accident probability, thus making the engineered safety system operate more reliably during accidents and mitigate accident more markedly. Changing the active engineered safety system into the passive one will decrease the core melt frequency, improve its safety properties and not cast aside the original nuclear power industry system[1].

Westinghouse AP-600 plant is a 600 MWe, two-loop plant which incorporates passive safety systems as frontline systems to mitigate the effects of loss of coolant accidents (LOCA) and transients. Its Passive Safety Injection System (PSIS), which protects the plant against leaks and ruptures of the RCS, are comprised of the Core Makeup Tanks (CMT), Accumulators, In-containment Refueling Water Storage Tank (IRWST) and Automatic Depressurization System (ADS)[2]. The preliminary analyses have shown that its safety properties have been improved largely compared with the present PWR nuclear power plants by satisfying the EPRI URD requirement (below ~10E-5/yr) on CDF with 1.3E-6/yr[3] and allowing a grace time of 72 hours without the need for operator interference after accidents. But, these safety facilities still do not eliminate the potential danger of core melting. Rather it has larger CDF than CE System 80+(CDF of System 80+ : 5.8E-7/yr)[4] that is an evolutionary design with still active components. The CDF of current AP600 design is dominated by small LOCA, which provides about 40% of total CDF. A review of the dominant accident sequences has indicated that small LOCA mitigation failure is dominated by the depressurization system failure.

In this study, the design vulnerabilities impacting the ADS unavailability have been identified and the design improvement items have been proposed through the system reliability assessment using the fault tree methodology. The impacts on CDF according to the change of system reliability have also been analyzed. In addition, small LOCA simulation using RELAP5/MOD3 code has been performed to show the thermal-hydraulic feasibility of the suggested design enhancements.

2. System Description

The AP-600 passive safeguards system consists of appropriately separated trains as shown in Fig.1. The CMTs, which provide safety injection (SI) during a LOCA, are located vertically above the reactor coolant loops and its pressure balancing lines from the pressurizer and the cold legs connected to the top of the CMTs maintain the tanks in the pressure equilibrium with the RCS during injection. The accumulators, which provide additional borated injection water to the RV in the event of a LOCA, comprise 85% water and 15% nitrogen by volume at an overpressure of about 700 psig. The ADS provides safety grade means to depressurize the RCS in order to permit passive safety injection following the accidents[5]. It consists of two separate trains each with three stages and a fourth stage which contains redundant valves. The first three stages of each train, located off the pressurizer steam space, discharge through a single sparger located in the IRWST. The fourth stage discharges from the non-pressurizer loop hot leg to the containment vessel. To minimize potential leakage effects, leakage detectors are installed at downstream of the depressurization valves. System actuation valves include the parallel valves in the CMT discharge line and in the cold leg/CMT pressure equalizing line and the depressurization valves in the ADS.

In the event of a LOCA, the AP-600 RCS is depressurized to the pressurizer low-pressure set-

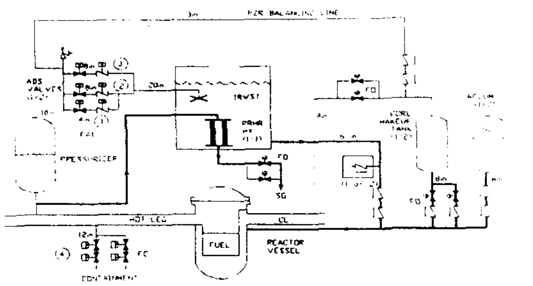

Fig. 1. AP600 Passive Safety Injection System

point, thereby actuating a reactor trip signal. The passive safeguards system is aligned for the delivery with the generation of a S-signal when the pressurizer low-low pressure setpoint is reached. Safety injection is provided initially by the two gravity drain CMTs. The accumulators provide additional SI water when RCS pressure becomes sufficiently low and the IRWST provides the water source for long-term core cooling. For IRWST injection, the RCS pressure must be lowered to about 25 psia[2]. This is accomplished just by the sequential opening of the ADS valves according to the CMT level. The piping sizes and the actuation setpoint involved in the actuation of the ADS are summarized in Tables 1 and 2.

Table 1. AP-600 ADS Related T/H Parameters

Pipe	Nominal Diameter(inch)
Injection Line (to RV)	8
CMT Discharge	8
CMT Vent Line from Cold Leg	8
CMT Vent Line from PZR	3
IRWST Discharge	6
Line from PZR to ADS Valves	10
1st Stage ADS Line	4
2nd and 3rd Stage ADS Lines	8
4th Stage ADS Line	8
ADS Discharge Line	20

Table 2. AP-600 ADS Actuation Setpoint

Parameter	Value
Reactor Trip	
Low RCS Pr. Trip Setpoint	1900 psig
CMT Actuation	
Low PZR Level	10%
Low PZR Pressure	1700 psig
ADS Actuation	
First Stage	75% CMT Level
Second Stage	65% CMT Level
Third Stage	50% CMT Level
Fourth Stage	25% CMT Level
IRWST Water Injection	
RCS Pressure	<25 psia

During a large LOCA, the depressurization is not an issue, since it occurs by itself. In the event of a small LOCA, the successful depressurization requires one of two lines of the first stage to open, two of four of the second and third stage lines to open, and one of two of the fourth stage lines[3]. In reality, the depressurization success criteria depends on the break size and less stringent criteria may be required for larger breaks.

3. System Design Enhancements

The PRA process, which consists of essentially two phase process, is used to derive the design improvements. The first phase involves determining the vulnerabilities of the current design to determine where design efforts should be targeted. The second phase involves integrating the PRA process into the design process to ensure that the design effort meets the targeted goals and resolves the identified vulnerabilities. In the first phase, PRA of current design must be reviewed to determine weaknesses in event mitigation and vulnerabilities in the target system. This assists in directing the system design improvement process to ensure current design weaknesses are addressed. The review should include not only a review of system design but also how it responds to events. The review should concentrate on the dominant initiating event and system failures which cause this event to be important. In this study, those are small LOCA and ADS failure, respectively. In the second phase, the PRA and design processes are integrated. It is an iterative process which requires developing a PRA model of the base design, identifying design weaknesses through quantification of CDF and examination of the results, and evaluating alternate system designs and/or operational strategies to optimize plant safety.

Important information from the PRA analysis for evaluating model adequacy, and system and operational design alternatives are the accident sequences. In this work, the sequences to core damage from small LOCA are constructed using the event tree technique as shown in Fig. 2. On the system level, pertinent information is in terms of cutsets which generally consist of component failures, from the fault tree analysis, and lead to system failures.

3.1. Key Considerations for Reliability Improvement

1) The relaxation of depressurization success criteria : This can be accomplished by the increase of the depressurization capability through the enlargement of the depressurization pipe size in the second and third stages (from current 8 inch to 12 inch). It is assumed that the depressurization success criteria may be relaxed from current "two of four of the second and third stage lines to open" to "only one line of four lines to open."

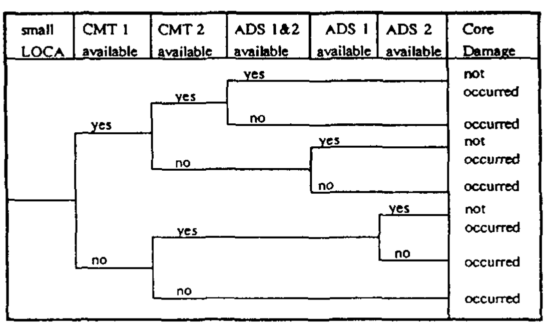

Fig. 2. Event Tree for Small LOCA

2) The reduction of the depressurization stages (in the second and third stages) : With 12 inch-one stage instead of current 8 inch-two stages, it is assumed that the depressurization success criteria is "one of two first stage lines to open," "one of two second stage lines to open," and "one of final depressurization stage lines (the fourth stage in the baseline design) to open". This may reduce the probability of the valve rupture and simplify system operation.
3) The subdivision of the first depressurization stage : As a result of the preliminary reliability analysis, it has been revealed that the main contributor on the ADS unavailability is the failure of the first stage depressurization. The break-down of the first stage is therefore suggested to overcome this aspect and is accomplished by rearranging current "4- inch of two first stage lines" into "3- inch of four first stage lines." It is assumed that the depressurization success criteria on the first stage is relaxed to "two of four first 3 inch lines to open."
4) The allowance of manual operation on the first stage : This item is considered based on the fact that significant portion of the unavailability of the ADS comes from the opening failure of the first stage valves as specified previously. It is assumed that the first stage valves can be opened manually by operator following the failure of the automatic opening.
5) The allowance of the partial depressurization : This item is considered to take the credit for successful partial depressurization. It is

Table 3. Overall Cases for Reliability Quantifications

case	considered items*
case 1	base design
case 2	[a]
case 3	[b]
case 4	[c]
case 5	[d]
case 6	[e]
case 7	[d] + [e]
case 8	[a] + [d]
case 9	[a] + [e]
case 10	[a] + [d] + [e]
case 11	[b] + [d]
case 12	[b] + [e]
case 13	[b] + [d] + [e]
case 14	[a] + [c]
case 15	[b] + [c]
case 16	[c] + [e]
case 17	[a] + [c] + [e]

* [a] : The enlargement of the depressurization pipe size in the second and third stages,
[b] : The reduction of the depressurization stages in the second and third stages,
[c] : Break-down of the first depressurization stage,
[d] : The allowance of the manual operation on the first stage,
[e] : The allowance of the forced depressurization using the NRHR system following the partial depressurization.

assumed that if the partial depressurization up to the third stage is successfully accomplished, the reactor will depressurize sufficiently to enable the Normal Residual Heat Removal (NRHR) system to be aligned and the IRWST water injection by the forced depressurization using NRHR system is possible. Although this runs counter the passive concept, it is considered based on the fact that major concern of this work is the improvement of system reliability.

In this study, the system reliability and the CDF during small LOCA are analyzed for each item mentioned above. Then the effects on the reliability and the CDF are analyzed for possible combinations of the considered items. The overall cases analyzed in this study are presented in Table 3

3.2. Reliability Quantification

3.2.1. Fault Tree Construction

The fault tree technique is used to assess the system reliability. A fault tree consists of a finite number of minimal cut sets, unique for the top event. The first step in fault tree construction is definition of the top event, which in this study, is that the ADS is unavailable. The following are considered as the base events : "Fail to Open," "Signal Generation Failure," "Plugged/Transfer Closed," and Common-Cause Failure of valves. Fault trees are constructed for only cases involved with the hardware change, i.e. case 1, case 2 and case 3 from Table 3. Quantification of system unavailability for the cases related to the manual operation is manually carried out based on the fault trees for cases 1 to 3. The unavailability of various components is based on the mission time of 24 hours. It is assumed that all failures are immediately detectable and there are no undetectable failures and that the test internal is quarterly. In addition, the success probability of manual opening of the first stage valve is assumed to be 0.5.

3.2.2. Method of Analysis

The KIRAP (KAERI Integrated Reliability Analysis Code Package) code[7] is used to generate the minimal cut sets. The code is an integrated probabilistic risk assessment software tool which gives the user an ability to create and analyze fault trees using a PC. Probabilistic importance measures are used to estimate the contribution of a particular basic event or minimal cut set to the system unavailability. In this study, Fussell-Vesely importance measure is used.

3.2.3. Data Base

The main data such as component failure rate and common-cause failure rate are obtained from Appendix A of the EPRI ALWR Requirement Documents[8], which contains a compilation of generic failure rate data from different sources. Data for the signal generation failure invoked in the CMT are also taken from generic data source[9].

3.3. Effects on Core Damage Frequency

The quantified system unavailabilities are used to analyze the effects on the CDF for each suggested items. The CDF analysis is based on the event tree as shown in Fig. 2 and considers only small LOCA as an initiating event. The following are assumed to construct the event tree : 1) The ADS valves in each train are sequentially opened according to the CMT level of the corresponding train ; 2) Negligible is the failure of check valves on the discharge lines of the accumulators and on the discharge lines of the CMT.

3.4. Results and Discussions

Table 4 shows the system unavailabilities due to the failure of each depressurization stage. This calculation was performed to identify how the failure of each stage contributes to the system unavailability and was based on the fact that the system unavailability due to the failure of each stage are combined by OR gate in the fault tree. From the results, it is shown that the main contributor to system unavailability is the first. This fact shows that the first stage may become the major target of the reliability improvement. It has been already suggested to improve system reliability that breaks down the first stage into more lines and/or allows the manual operation following the failure of the automatic opening. In the other hand, from the result, we can expect only slight reliability improvement in the second and third stages. Except the allowance of partial depressurization, the fourth stage has less expectation of reliability improvement.

The results for all analysis cases are summarized in Table 5. The case showing the largest reliability improvement is case 17, which considers concurrently "break-down of the first stage," "enlargement of the second and third stage lines," and "the allowance of partial depressurization using the NRHR system." Next one is case 16. These cases bring also the significant reduction of CDF. In the hardware-change related cases except case 4, case 2 shows insignificant improvement in reliability as expected and rather case 3 shows more system unavailability. But we should recognize that break-down of the first stage may come into conflict with the efforts reducing the inadvertent opening since it accompanies with the increase of the vent valves. If not consider any improvement on the first stage, case 9 is most practical and brings about 5.5% and 6.5% of reduction on the system unavailability and the CDF respectively. The single items showing the largest improvement effect are "break-down of the first stage" and "the allowance of manual opening for stage 1 valves." This fact shows again that the dominant factor on the system unavailability is the failure of the first stage. Finally, it should be recognized that this analysis is performed without any consideration

Table 4. ADS Unreliability According to the Failure of Each Stage

case	Stage 1	Stages 2&3	Stage 4
case 1	2.16 E–4	3.04 E–7	1.78 E–5
case 2	2.16 E–4	4.03 E–8	1.78 E–5
case 3	2.16 E–4	2.16 E–4	1.78 E–5
case 4	3.04 E–7	3.04 E–7	1.78 E–5

Table 5. Reliability Quantification Results

case	ADS* Unavailability	ADS 1 or 2** Unavailability	CDF	Ratio (in CDF)
case 1	2.34 E-4	1.31 E-2	7.13 E-7	1
case 2	2.338E-4	4.08 E-2	7.05 E-7	0.99
case 3	4.50 E-4	8.83 E-3	1.36 E-6	1.91
case 4	1.84 E-4	4.06 E-3	5.85 E-8	0.08
case 5	1.26 E-4	1.12 E-2	3.87 E-7	0.54
case 6	2.22 E-4	1.21 E-2	6.74 E-7	0.95
case 7	1.14 E-4	1.01 E-2	3.49 E-7	0.49
case 8	1.26 E-4	2.06 E-3	3.79 E-7	0.53
case 9	2.21 E-4	4.07 E-3	6.67 E-7	0.94
case 10	1.13 E-4	2.05 E-3	3.42 E-7	0.48
case 11	3.42 E-4	6.81 E-3	1.03 E-6	1.45
case 12	4.37 E-4	8.10 E-3	1.32 E-6	1.85
case 13	3.29 E-4	6.07 E-2	9.92 E-7	1.40
case 14	1.81 E-4	8.13 E-3	6.08 E-8	0.085
case 15	2.34 E-4	1.32 E-2	7.13 E-7	1
case 16	5.94 E-6	4.06 E-2	2.61 E-8	0.036
case 17	5.67 E-6	8.11 E-3	2.36 E-8	0.033

* Unavailability for two trains of ADS,
** Unavailability for ADS train 1 or train 2 respectively.

for the thermal-hydraulic aspects and the detailed design is required to verify the adequacy of manual operation.

4. Small LOCA Analysis for Reliability Improvement Items

The ADS design improvement items in system reliability have been provided in the previous section but have been taken without any consideration of thermal performance. The double-ended rupture of one of the two passive safeguards systems lines is simulated using RELAP5/MOD3 code to verify the feasibility of each enhanced design. The RELAP5/MOD3 code has been developed as a code for the best-estimate transient simulation of PWRs and associated systems. It has been verified in some previous studies[11],[12] that RELAP5 is applicable to SBLOCA analysis of AP600 transient NSSS phenomena. The analysis is limited to the cases related to the hardware change.

4.1. Input Model Nodalization

The RELAP5 input model contains presentative modelling for all major primary, secondary, and PSF components and containment volume. Each of the loops has been modelled separately, including separate hot legs and steam generators and individual RCPs and cold legs. Each of the two PSF component trains has been modelled separately. In the present form the input model is a generalized representation of the AP600 design and consists of 246 node, 280 junctions, and 250 heat structures[4] ; it contains the compromises necessary to construct a complete working model where information was unavailable. In as much as the AP600 design shares much in common with existing Westinghouse NSSS component design, the original model was supplemented with representative off-the shelf RELAP5 component models

Table 6. Boundary and Initial Conditions

Parameters	Value
1. Break opens at DVI-A	instantaneously at t=0 sec
2. Pressurizer low pressure (reactor trip signal)	131.01 bar
3. Reactor trip signal delay	2.0 sec.
4. Turbine stop valve starts to close	at reactor trip signal
5. Turbine stop valve closure time	2.0 sec.
6. Main feedwater isolation	at TBN stop valve closure
7. Reactor coolant pump trip	at PZR low-low level
8. Pressurizer low-low pressure (safety injection signal occur)	117.21 bar
9. Safety injection signal delay	2.0 sec
10. CMT-cold leg pressure balance line valve open	20% of void fraction in the cold leg
11. Accumulator flow begins	48.26 bar
12. IRWST flow begins	containment pr.+static head

where data was lacking. These included existing Westinghouse model F SG models and representative, oversized, 3-loop reactor vessel and pressurizer models. For the ADS, only one train is modeled in compliance with a single failure limit. The main boundary and initial conditions used in this analysis are reported in Table 6.

4.2. Transient Description

In order to investigate possible limitations of the suggested ADS design improvements, five small-break LOCA cases have been run. The base case is based on the current design. The descriptions of each case are presented in Table 7. Among those cases, the case with no the ADS is simulated for the validity of the input model and to discuss the necessity of the ADS in the AP600. As mentioned previously, the double-ended rupture of one of the two passive safeguards systems lines are assumed in this analysis. In this case, only half of the passive SI system flow is available to provide water to the reactor vessel. This rupture results in not only the expulsion of RCS inventory from the RV downcomer out the broken 8-in. piping but also a loss of RCS inventory through the 8-in-diameter cold leg pressure relief line following the spill of the broken loop CMT water to the containment. In all cases except the case with no the ADS, the first- and fourth-stage vent valves open when the CMT level reaches 75% and 25% respectively. For base case and case 3, the second- and third-stage valves actuate as before. But in the cases of case 2 and 4, the second stage vent is bypassed and the third stage valve is opened when the CMT level reaches 50%. This control logic is based on the depressurization success criteria corresponding to each case.

4.3. Transient Results and Discussions

The results about AP600 thermal-hydraulic re-

Table 7. Transient Analysis Cases

case	Description
case 1	current design
case 2	enlargement of the pipe size in the second and third stages
case 3	break-down of the first stage
case 4	case 2+case 3
case 5	without ADS

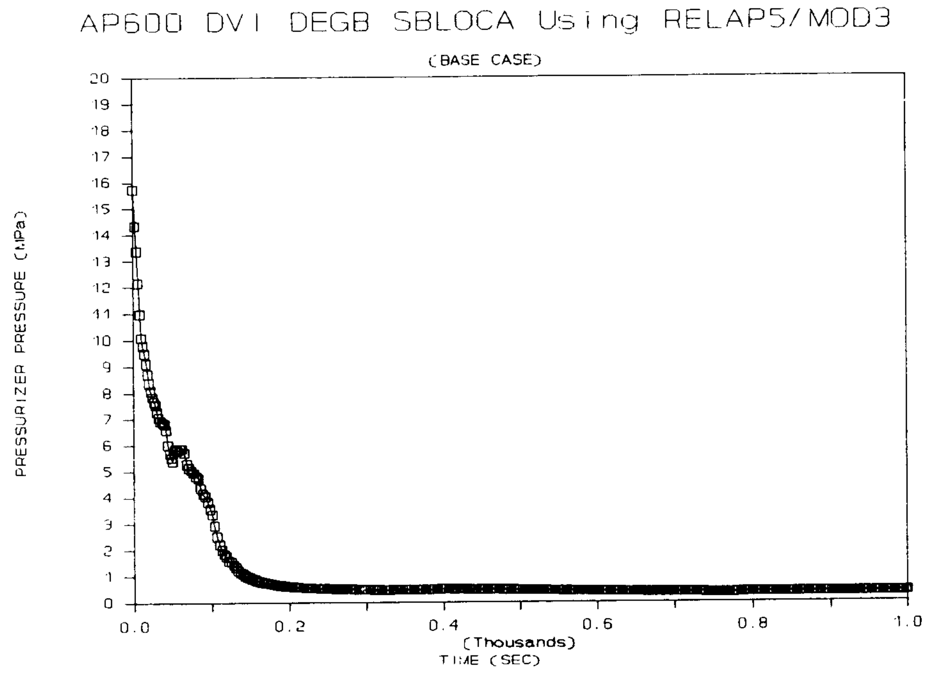

Fig. 3. Pressurizer Pressure for Case 1

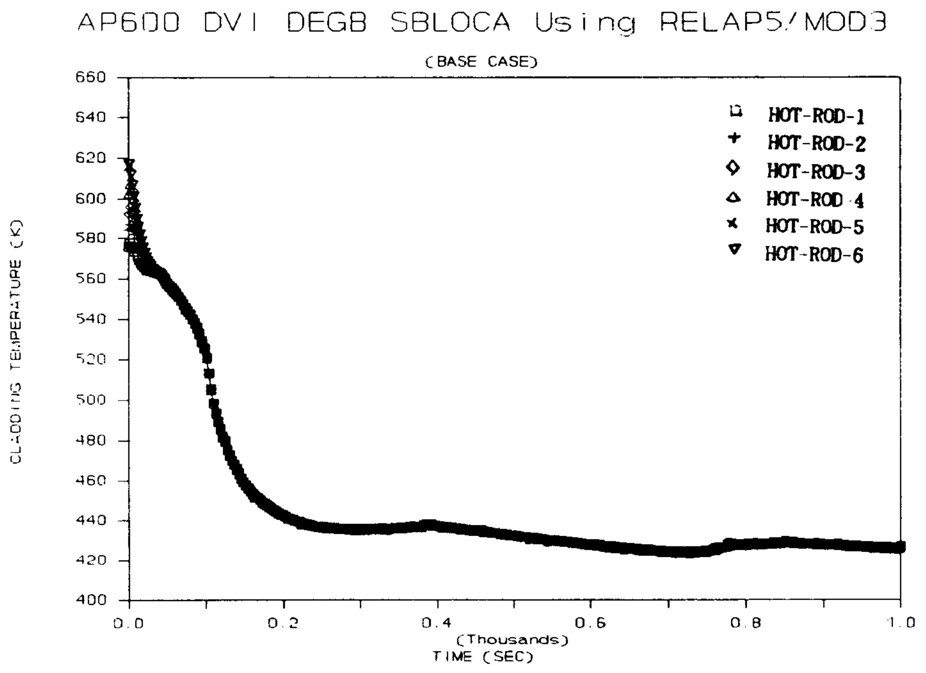

Fig. 4. Cladding Temperature for Case 1

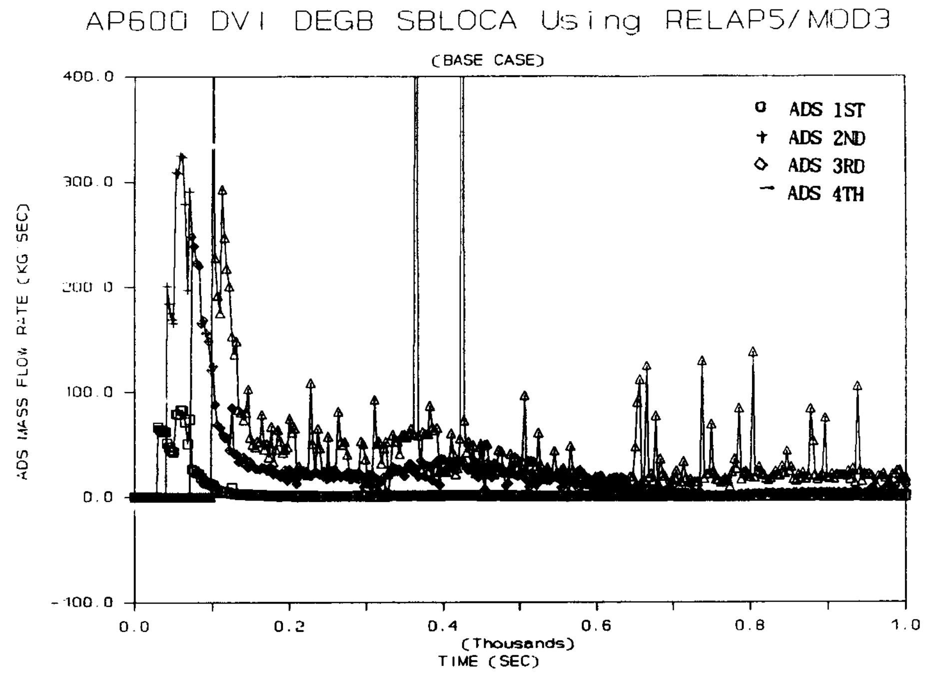

Fig. 5. ADS Mass Flow Rate for Case 1

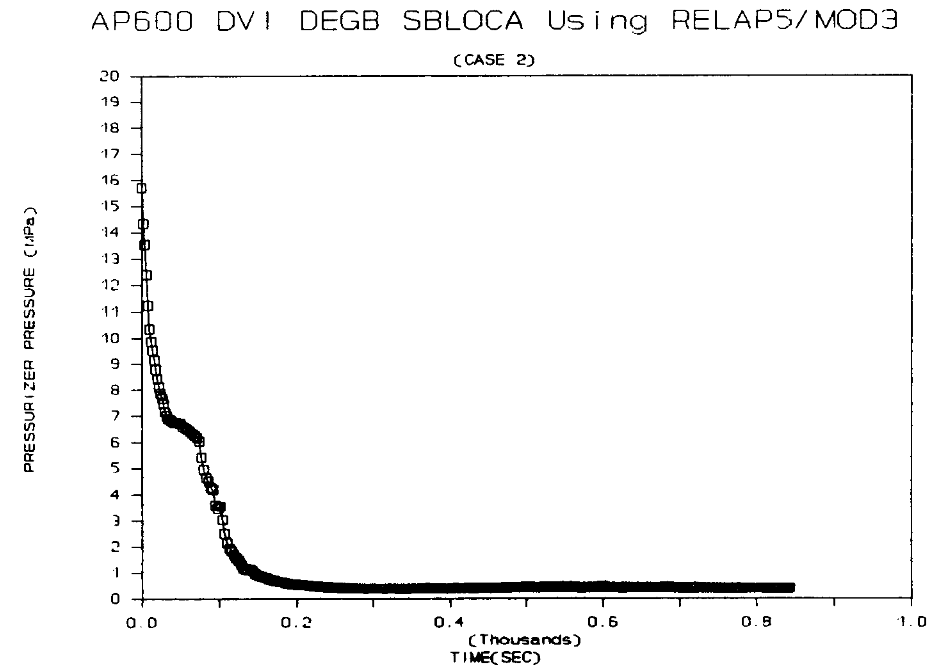

Fig. 6. Pressurizer Pressure for Case 2

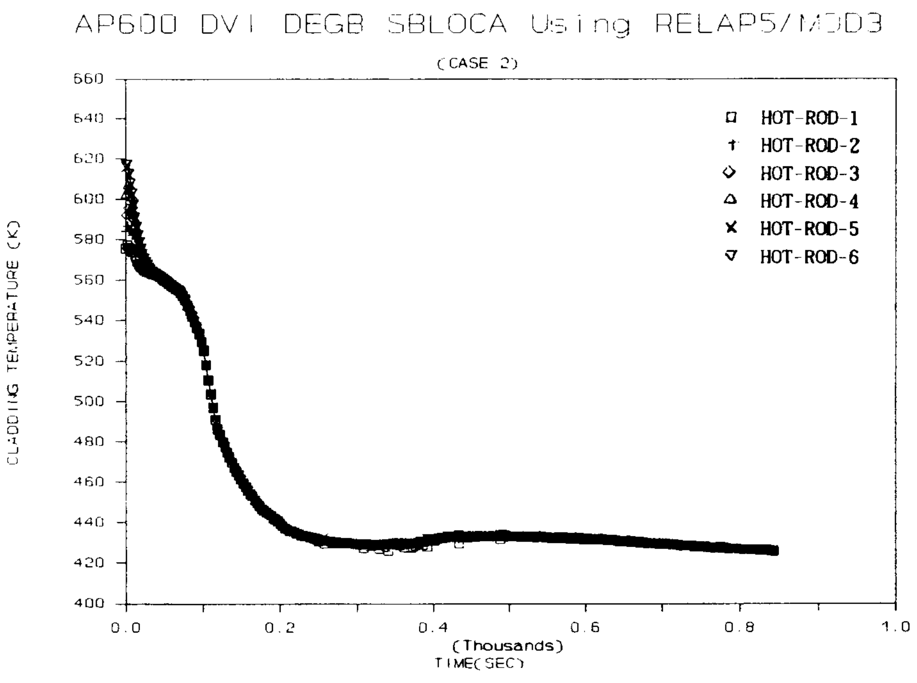

Fig. 7. Cladding Temperature for Case 2

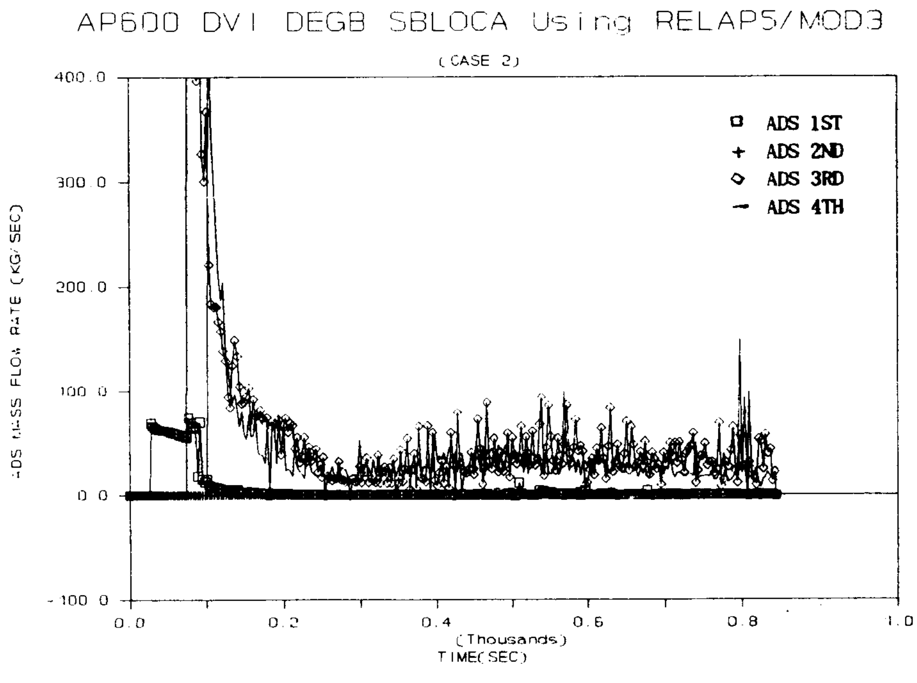

Fig. 8. ADS Mass Flow Rate for Case 2

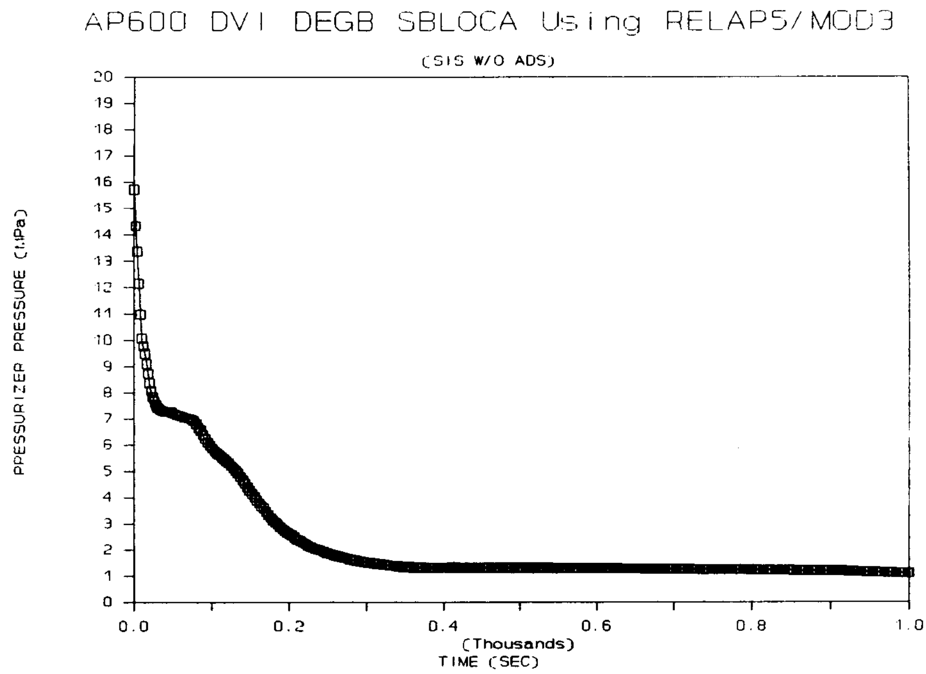

Fig. 9. Pressurizer Pressure for Case 5

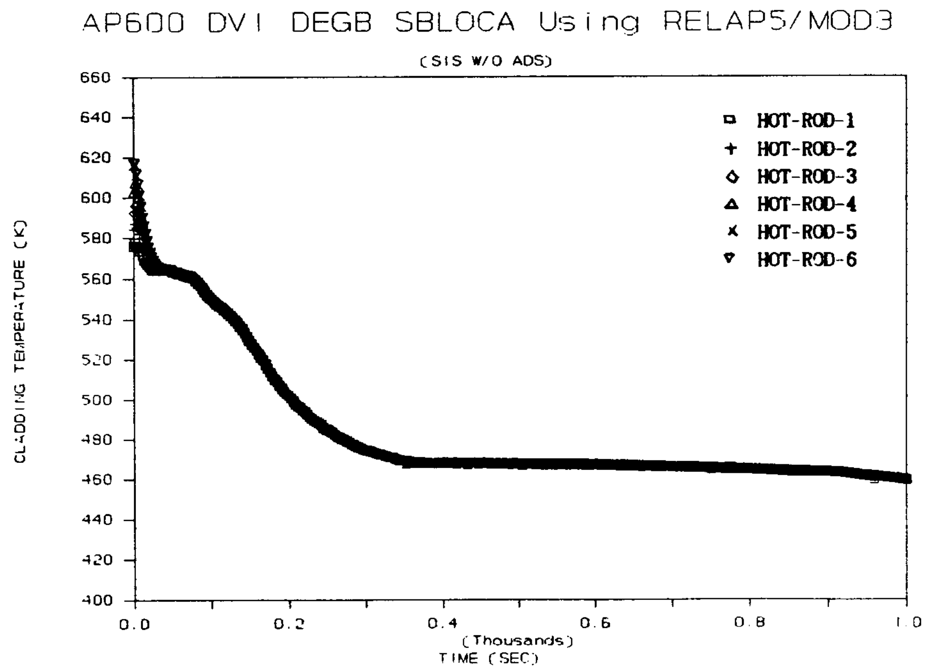

Fig. 10. Cladding Temperature for Case 5

sponse to the small break LOCA have been in Fig.3 to Fig.10. The results for cases 3 and 4 are not shown but have very similar trends compared with case 2. As shown in the figures, the 8-in.-diameter passive safeguards system line break causes a rapid depressurization of the RCS to the reactor trip and SI signal setpoint. In all cases except the case with no ADS, the broken loop CMT level decreases rapidly to the 75% level setpoint in approximately 30 seconds, and the first stage valves open. All valves in the ADS open by 100 seconds; the RCS quickly depressurizes to approximately a pressure at which the IRWST is able to inject. The cladding temperature maintains around 440 K significantly lower than the design criteria. In the case with no ADS, the RCS pressure rapidly depressurizes to about 180 psia (1.2 MPa). But the RCS does not depressurize further and is saturated at 180 psia. Also, the cladding temperature maintains about 470 K higher than other cases and can not be sured to be maintained around that point without any more heatup. Cases 2, 3 and 4 do not show any significant deviation in the depressurization and cooling trends. Exceptionally, it is shown that the RCS pressure is a little bit saturated before the opening of the third stage valve in the case that the pipe size of the second and third stages is enlarged. But this kind of saturation seems to have no impacts on the successful depressurization and cooling. Particularly considering the thermal load on the sparger depending on the ADS mass flow rate, it is shown that the ADS discharges at opening of the third stage valve becomes twice that of the base case. At this point, we can not understand how the vibration in the ADS or the dynamic effects on the IRWST structure are affected by the increased thermal load on the sparger. Finally, it may be considered that the system improvement items suggested above are allowable in the aspect of the depressurization and cooling capabilities.

5. Conclusions

Through the system reliability assessment, the ADS design improvement items have been suggested with the identification of design vulnerabilities impacting the ADS unavailability. Also, the LOCA analyses for small break were performed to study the thermal-hydraulic feasibility of the proposed design enhancements. As a result of the study, the following are recommended to improve the ADS reliability: first, break-down of the first depressurization stage; second, the allowance of the manual operation on the first stage; third, the allowance of the forced depressurization using the NRHR system following the partial depressurization. It may be considered that the system improvement items suggested above are allowable in the aspect of the depressurization and cooling capabilities.

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