Journal of the Korean Nuclear Society
Volume 21, Number 1, March, 1989

Risk Model Development for PWR During Shutdown

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(Received June 23, 1988)

원자로 정지 동안의 위해도 모델 개발

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Abstract

Numerous losses of decay heat removal capability have occurred at PWRs during stutodwn while its significance to safety is needless to say. A study is carried out as an attempt to assess what could be done to lower the frequency of these events and to mitigate their consequences in the unlikely event that one occurs. The shutdown risk model is developed and analyzed using Event/Fault Tree for the typical pressurized water reactor. The human cognitive reliability (HCR) model, two-stage bayesian approach and staircase function model are used to estimate human reliability, initiating event frequency and offsite power non-recovery probability given loss of offsite pwer, respectively. The results of this study indicate that the risk of a PWR at shutdown is not much lower than the risk when the plant is operating. By examining the dominant accident sequences obtained, several design deficiencies are identified and it is found that some proposed changes lead to significant reduction in core damage frequency due to loss of cooling events.

요 약

원자로 정지동안에도, 잔열제거계통은 그 기능이 계속 유지 되어야 하나, 실제로 가압 경수로에서 냉각상실고가 많이 발생 되어 있다. 본 논문은 원자로 정지중의 냉각기능상실을 예방 하고, 또한 냉각기능상실로 인한 노심손상의 중대성을 완화 시키기 위한 대책을 강구하기 위한 시도로서, 전형적인 가압경수로에 대한 사고/고장 수목과 운전원실수 확률을 위한 HCR 모델, 초기사상의 빈도를 위한 2단계 bayesian 방법 및 고장난 계통의 회복 활률을 위한 계단함수 모델 등을 이용한 원자로 정지 위해도 모델을 개발하여, 잔열제거계통의 신뢰도를 분석 하였다. 그 결과는 원자로가 정지 중일 때의 위해도가 운전중일때이 것에 비해 별로 낮지 않은 것으로 나타났으며, 몇 가지의 설계개선을 통하여 냉각기능상실로 인한 노심 손상확률을 상당히 낮출 수 있는 것으로 나타났다.

1. Introduction

Numerous losses of decay heat removal capability have occurred at PWRs during shutdown {1-4} while its significance to safety is needless to say. There are relatively few Technical Specification requirements on operability of the safety systems for the plant in cold shutdown condition. In fact, some safety systems are forced to be disabled, e.g.,accumulator, safety injection system, and non-operating charging pumps. And maintenance unavailability of hardware tends to be higher during an outage. For example, 4kv essential buses may be under maintenance during an outage. Another problem is that Reactor *Coolant* System(RCS) may be partially drained and the steam generators may not be available. In the absence of prompt mitigative action by the operator when the loss of DHR occurs, the core may become uncovered. If core damage occurs while the plant is in cold shutdown, the subsequent release of radioactivity and the consequences may be higher than that for core damage resulting from an accident that occurs while the plant is at power, because the containment may be open when the plant is at cold shutdown condition (e.g., equipment hatch open, seal failure of penetrations).

The purpose of this paper is to estimate the improvement in the DHR reliability and risk reduction potential. The shutdown risk model is developed and analyzed using Event/Fault Tree for the typical pressurized water reactor. The human cognitive reliability (HCR) model, two-stage bayesian approach and function model are used to estimate human reliability, initiating event frequency and offsite power non-recovery probability given loss of offsite power, respectively. The benefit of the changes is expressed in terms of the reduction in frequencies of loss of cooling events and core damage.

Section *2* describes the shutdown risk model for the Decay Heat Removal Systems(DHRS) in order to quantify the core damage frequency for a generic PWR. The model includes defining the phases of 3 types of outages and using the Human Cognitive Reliability (HCR) model {5,6} to assess human error probabilities. Section *3* discusses the collection and analysis of the generic data, Section *4* discusses the quantification of the core damage frequencies resulting from the various initating events. Section 5 discusses the benefits of the proposed improvements for a generic plant, and summary and conclusion follow in seotion *6*.

2. Shutdown Risk Model for DHRS

It has been assumed that systems of the generic plant, including frontline systems and support systems, are similar to those of Zion Plant, so that the system models created for the Zion can be used for the generic plant. The analysis is called "generic" mainly because generic component failure data have been used and the frequencies of the initiating events have been estimated using the operational experience of the PWR population The overview of cold shutdown modeling approach is shown in Figure *1*.

1) The Definition of Phases for 3 Types of Outages

The phases of an outage are defined in terms of the time at which a phase starts and the time at which a phase ends, and are characterized by the conditions of the plant such as whether or not the RCS is drained and whether or not the RCS is open. The plant conditions are then used to determine the time available for operator actions and the human error probabilities. Table *1* summarizes the definition of the phases for 3 types of outages. Different phases of an outage occur sequentially. Therefore decay heat is lower for later phases, and the time available for operator actions, given a loss of cooling event, tends to be longer. The most vulnerable condition that a plant may be in is that when the RCS is drained shortly after the shutdown.

(1) Refueling Outage

Phase 1—This phase starts when RHR system is initiated after a shutdown, and ends when the RCS is drained to hotleg midplane. NSAC—84{3} estimated that the mean time at which the RHR system is initiated in a refueling outage is 54 hours after shutdown. This is the starting time of Phase 1. It also

Figure 1 : Overview of Cold Shutdown Modeling Approach

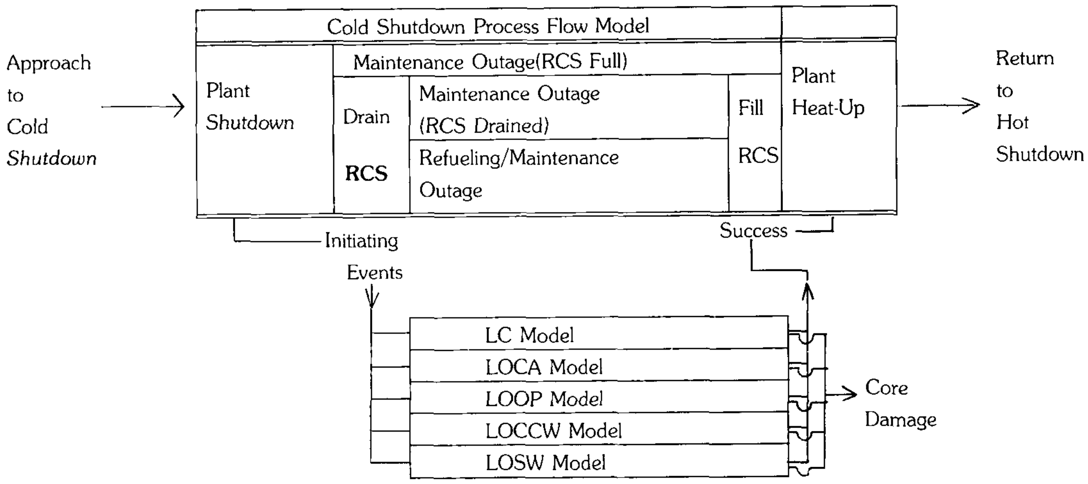

Table 1. Durations and Characterization of Phases of 3 Types of Outages

Outages	Phase	Start*	End*	Duration	Plant Conditions
Refueling	1	54	167	113	RCS Cooling Down, RCS Filled
	2	167	587	420	RCS Drained, SG Eddy Current Test
	3	587	1087	500	Refueling Cavity Filled, Fuel Shuffling,
	4	1087	1996	909	Vessel Head Off RCS Filled, Maintenance
Drained Maintenance	1	21	83	62	RCS Cooling Down, RCS Filled
	2	83	179	96	RCS Drained, Maintenance
	3	179	982	803	RCS Filled, Maintenance
Nondrained Maintenance	1	21	146	125	RCS Filled, Maintenance

Note : *Time(hour) sfter shutdown

showed that RCS draining is initiated at 118 hours after shutdown and takes 49 hours to complete draining. Therefore, drained conditions are reached at 118 + 49 = 167 hours. This is the ending time of Phase 1. The duration of Phase 1 is therefore 167-54 = 113 hours.

For the subsequent analysis, Phase 1 is characterized as a phase with the *RCS* filled and the decay heat is high. *NSAC*{*3*} estimated that 3.8 hours will be available if loss of coolingoccurs at 6 hours after shutdown with *the RCS at 425 psig, 350 °F and a bubble in the* pressurizer. This time, 3.8 hours, is used to estimate the human error probability for failure to diagnose the loss of cooling event.

Phase 2 — In this phase, the RCS is drained to the *hotleg midplane, so that tests and maintenance can* be performend on the steam generators and other components of the primary coolant system. Information taken from several plants (Zion, Prairie Island, Diablo Canýon and Seabrook) {4} indicates that a plant spends about 2 to 3 weeks per refueling outage in the drained *condition.* For the analysis, 2.5 weeks are used as the duration of this phase, i.e., the phase starts at 167 hours after shutdown and ends at 587 hours. With the RCS partially drained, the time to core uncovery for a loss of cooling event can be determined as described in Section *2)* Notice that the steam genrator is not available during this phase as shown in Figure *2.*

Figure 2 : Loss of Colling Event Tree for Phase 1 of Drained Maintenance

Sequence	Core Damage State
1	OK
2	OK
3	OK
4	Core Damage
5	Core Damage
6	OK
7	OK
8	OK
9	Core Damage
10	Core Damage
11	OK
12	OK
13	OK
14	Core Damage
15	Core Damage

* Lower branch of CV represents failure of RHRS due to spurious closure of suction valves or loss of suction due to overdraining. Upper branch represents other failure modes of operating RHRS train.

CV : Failure Modes of Operating RHRS Train
RT : Operator Trips RHR Pumps
DE : Operator Determines Action to Restore Cooling is Required
RH : Normal RHRS Restored
SG : Steam Generator Cooling
BF : Continued Makeup CVCS(Feed and Breed)
SI : Safety Injection System

Phase 3 — In this phase, the refueling cavity is filled and actual fuel shuffling takes place. NSAC-84{3} estimated that the duration with the vessel head off is typically 500 hours. This is used as the duration of this phase. Therefore, the phase starts when the previous phase ends at 587 hours, and ends at 1087 hours. With the refueling cavity filled, a lot of time will be availableif decay heat removal capability is lost. Therefore, the probability of human error for failure to diagnose the situations is very low. As is discussed in *Section 2).2*, a limiting human eror probability of 1.0 E-06 is used for failure to diagnose the problem under these circumstances.

Phase 4 — In this phase, test and maintenance after refueling is performed. The RCS is filled and one-third of the fuel is fresh. Again, a lot of time is available for operator actions to respond to any abnormal event. The duration of this last phase of a refueling outage is determined such that the total duration of the outage is 1996 hours (i.e., 1996-1087 = 909).

② Drained Maintenance Outage

Phase 1 — This phase is similar to Phase 1 of a refueling outage. It starts when the RHR system is initiated and ends when the RCS is drained. Table 3-4 of NSAC-84 {3} lists the times to RHR initiation for maintenance outages at Zion. The mean time is approximately 21 hours. This value has been used as the time at which Phase 1 starts. Figure 3-4 of NSAC-84 {3} estimated that the draining of the RCS is started at 54 hours and the task takes 29 hours to complete. Therefore, the phase ends at 54 + 29 = 83 hours.

Phase 2 — This phase is similar to phase 2 of a refueling outage, except that the RCS is drained sooner and the duration of the phase is shorter. The phase starts at 83 hours after shutdown. The decay heat is relatively high at this time. With minimal amount of coolant inventory in the system, the time available for the operators to respond to any abnormal event is relatively short. As shown in Table 2 and Figure 3, approximately 2.7 hours will be available before core uncovery occurs, if a loss-cooling event occurs at the beginning of this phase. The duration of this phase is estimated to be 4 days based on information obtained

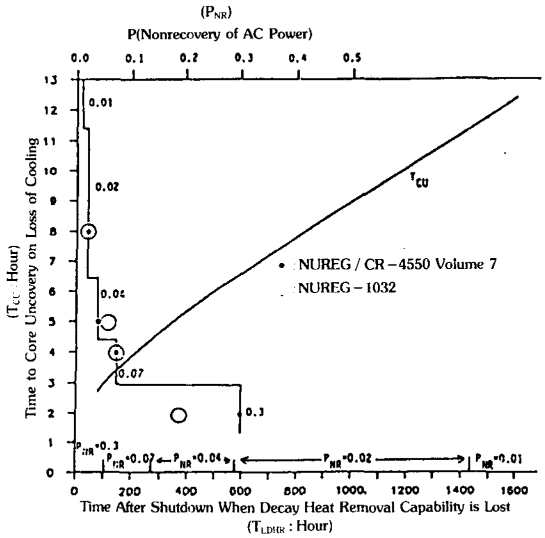

Fig. 3 : Time to Core Uncovery on Loss of Coling and AC Power Nonrecovery as a Function of Time after DHR Capability is lost.

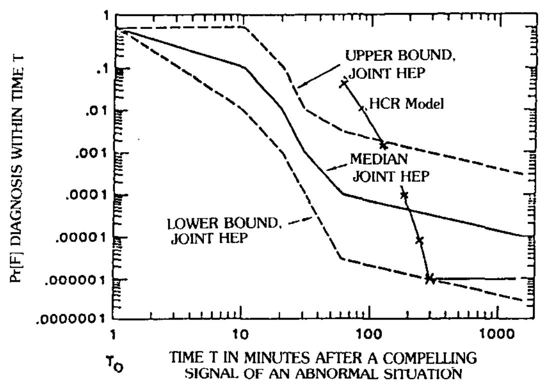

Fig. 4 : A Comparison of the HCR Model used with the Model of the handbook for Human Reliability Analysis

from Oconee PRA{7}.

Phase 3—This phase starts when the maintenance activities that require the RCS to be drained are completed at 179 hours, and the duration is estimated so that the sum of the durations of the three phases is equal to the duration of the drained maintenance. NSAC-84 {3} estimated that the duration of a drained maintenance is 982 hours. Therefore, the duration of this phase is 982-179=803 hours. In this phase, the RCS is filled and test and maintenance is being performed. Due to the large quantity of coolant inventory available, plenty of time is available for operator actions.

(3) Nondrained Maintenance Outage

Only 1 phase is used to model this type of outage. It is similar to phase 1 of a drained maintenance outage. The RHR system is initiated at approximately 21 hours after a shutdown. The duration of a nondrained maintenance at Zion is approximately 146 hours. Therfore, the duration that the RHR system would be operationg is 146-21=125 hours.

2) Determination of Time to Core Uncovery Using the Decay Heat Curve

The following equation (1) expresses the decay power as a function of the time, (sec.) after shutdown and the duration, T_0(sec.) for which the plant had been operating before shutdown{8}

$$P(\tau)=0.1P_0[(\tau-T_0+10)^{-0.2}-0.87(\tau+2\times10^7)^{-0.2}$$
$$-0.87(\tau-T_0+2\times10^7)^{-0.2}]$$

Where P_0 is the power of the reactor, i.e., 3250 MWt for Zion. The energy generated form time T_1 to T_2 is the integral of the equation (1) from T_1 to T_2. If loss of decay heat removal capability occurs at T_1, the time, at which the energy generated from decay heat is equal to what is needed for core uncovery to occur, can thus be determined. The time to core uncovery in Figure 3 is calculated assuming T_0 is one year.

3) Human Error Probability of Failure to Diagnose Abnormal Events

The human cognitive reliability (HCR) model {5, 6} is used to quantify the human error probabilities for failure to diagnose abnormal events. A brief description of the model is provided here, along with a comparison with the model given in the Handbook for Human Reliability Analysis{9}.

The HCR model is expressed in terms of the following formula:

$$P(t)=\exp-\left\{\frac{(t/T_{1/2}-C_{\gamma i})}{C_{\eta i}}\right\}^{\beta i} \qquad (2)$$

where,

t : time available for the operators to diagnose,

$T_{1/2}$: estimated median time taken for the operators to diagnose,

C_{ri}, C_{ni}, β_i : correlation coefficients associated with the i-th type of mental processing, e.g., skill, rule or knowledge which can be calibrated with simulator data, and

P(t) : the non-response probability for a given time t.

The specific application of this model is described as follows. The HEP for operator response to each of the initiating events has been calculated using Equation (2) where HEP(t)=P(t). Section 1) of 3 reviews operational experience of loss of decay heat removal events, and conservatively estimates that on the average it took appoximately 15 minutes for the operator to diagnose the event. Therefore, $T_{1/2}$ is taken to be 15 minutes. The variable "t" in Equation (2) represents the time that is available for operating crew diagnosis This time is estimated based on the specific plant condition at which loss of cooling occurs. For example, Table 2 lists the time to core uncovery as a function of the time at which loss of cooling occurs, assuming that the plant is in a drained condition. Also listed in Table 2 is the human error probabliity calculated using the HCR model. It is assumed that the operators are not formally trained to respond to a loss-of-cooling event during an outage, and no emergency procedure is available. Therefore, the task is considered to be "knowledge based". The following parameters were obtained for this type of tasks in reference {5,6} using the small scale test data:

$C_{ri}=0.527$,

$C_{ni}=0.744$,

$\beta_i=0.81$.

They are used in calculating the human error probabilities listed in Table 2.

Figure 4 is a comparison between the HCR model using Equation (2) and the model taken from the Handbook of Reliability Analysis{9}. The curve marked with crosses is calculated using Equation (2). It is plotted

Table 2 : Core Uncovery Time and Human Error Probability of Failure to Diagnose When the RCS is Drained

Time at Which Loss of Cooling Occurs(Hour)	Core Uncovery Time* (Hour)	Probability ** (Failure to Diagnose)
83	2.7	2.3E-04
103	3.0	1.0E-04
123	3.2	6.2E-05
143	3.4	3.7E-05
167	3.6	2.3E-05
179	3.7	1.8E-05
187	3.8	1.4E-05
207	3.9	1.1E-05
227	4.1	6.6E-06
247	4.3	4.1E-06
267	4.4	3.2E-06
287	4.5	2.5E-06
307	4.7	1.6E-06
327	4.8	1.2E-06
347	5.0	1.0E-06
367	5.1	1.0E-06

Notes : * Calculated using Equation (1)

** Calculated using Equation (2)

for a time greater than 1 hour. It is assumed that 10E-06 is the lower bound of human error probability, no matter how much time is available. This is why the curve becomes a constant line after 300 minutes. The other curves in Figure 4 are taken from the Handbook. The solid curve from the handbook represents the median curve. A very large error factor, 30, is assumed for the time range that is greater than 1 hour.

It can be seen that the HCR model curve decreases more rapidly with time, and falls within the uncertainty bounds of the model taken from the Handbook,

3. Generic Data Collection and Analysis

This section provides discussions on the collection and analysis of the generic data that are needed for the generic assessment. Section 1) discusses the operational experience to loss of DHR, Section 2) uses this operational experience to estimate the frequencies of initiating events that will lead to loss of DHR. Section 3) discusses the estimation of the mean diagnosis time of loss of cooling events. Section 4) discusses the generic data for component failures.

1) Operational Experience of Loss of DHR

Three sources of operational experience of losses of DHR are used in the generic data analysis. They cover different periods of time. There were 177 total events of loss of DHR for the period of 10 years('76-'86). The abstracts of these events are in Appendix C of reference{4}. For the convenience of quantitative analysis, these events have been classified into five types of failures: 1) spurious isolation of RHR suction valves, 2) overdraining the RCS, 3) failure to maintain RCS level, 4) loss of RCS coolant, and 5) other failures. Table 3 summarizes the operational experience according to this classification.

2) Estimation of the Frequencies of Initiating Events Leading to Loss of DHR

The operational experience summarized in Table 3 has been used to estimate the frequencies of loss of DHR due to different causes. It is estimated, using the Gray Book{10}, that all PWRs have accumulated approximately 504 years of operating experience from 1976 to 1986. The number of hours that a plant stays in a shutdown condition is the sum of the numbers of hours that the plant stays in 3 types of outages, i.e.,

0.747 refueling/year $\times$1996 hour/refueling
+1.932 drained maintenance/year $\times$982 hour/drained maintenance
+1.121 nondrained maintenance/year$\times$146hour/nondrained maintenance=3550 hours

Here, the frequency of each phase is taken from the Zion experience {3}. Therefore, the total experienced operationg time of the RHR system is estimated to be

505 year$\times$3550 hours/year=1.79E+06 hours

The freqencies of different initiating events that cause RHR systems to become unavailable are estimated

Table 3 : Classification of Loss of DHR Events and Frequencies of Initiating Events That Lead to Loss of DHR

Initiating Events	NSAC-52 '76-'82	AEOD '82-'83	BNL '84-'86	Total '76-'86	Frequncy/ Probability
Spurious Isolation	23	20	21	64	3.58E-05/h
Overdraining	4	9	8	21	1.21E-02
Inadequate Inventory	8	5	3	13	6.35E-05/h
LOCA	7	0	2	9	5.03E-06/h
Spurious Cont's Spray	2	0	0	2	1.12E-06/h
Others	42	11	12	65	
Total	86	45	46	177	

using two-stage bayseian approach, where applicable, and are shown in Table 3 as mean values.

3) Estimation of Mean Diagnosis Time for Loss of Cooling Events

Descriptions of the 177 loss of DHR events that are categorized in Table 3 were reviewed in an attempt to identify the diagnosis time. The descriptions of more than 50% of the events provided information that is needed to determine the duration that the RHR system is not available. This duration is the diagnosis time plus the time it takes to restore DHR, and, therefore would be a conservative estimate of the diagnosis time. It is observed that loss of cooling events that are caused by overdraining, inadequate inventory and LOCAs tend to have longer recovery time. The reason is that it takes longer to restore the RHR pumps. Review of the event descriptions indicates that the recogntion of the loss of cooling event occurs long before the RHR flow is restored. Therefore, it would be too conservative to use recovery time to estimate the diagnosis time. It was, therefore, decided to use the experienced recovery time of those loss of cooling events that were caused by spurious closure of RHR suction valves and certain other failures to estimate the diagnosis time. The 49 events of spurious suction valve closure events have a total of 675 minutes of RHR recovery time and 32 events due to other failures 523 minutes. Therefore the average recovery time is approximately 15 minutes, which is used as T'_{z} in Equation (2) for modeling cognitive errors.

4) Generic Component Failure Data

The generic data in the Oconee PRA{7} were reviewed and considered reasonable. Therefore, they have been used in the quantative analysis of section 4.

It has been assumed that the maintenance unavailabilities used for Zion are representative of those for all PWRs, and, therefore, are also used in the generic analysis.

4. Quantification of Core Damage Sequences Using the Generic Data

In this section the generic data estimated in Section 3 are used to quantify the core damage sequences of the model described in Section 2. Section 1).1 discusses the quantification of the loss of cooling(LOC)

Table 4. Summary Results for Loss of Cooling Event Trees (CDF per Year)

Outage Type	Phase 1	Phase 2	Phase 3	Phase 4	Total
Refueling	9.80-7	5.83-7	1.57-6	4.96-7	2.75-6
Drained Maintenance	1.40-7	4.27-6	1.16-6	N/A	5.57-6
Nondrained Maintenance	1.62-7	N/A	N/A	N/A	1.62-7
Total					8.48-6

Note : 8.48-6=8.48E-06

Table 5 : Summary of Results for Generic Shutdown Risk

Initiating Event	Frequency(per Year)
Loss of Cooling	8.48E-06
LOCA	2.61E-06
Loss of Offsite Power	5.31E-06
Loss of CCWS	2.34E-05
Loss of SWS	4.04E-06
Total	4.30E-05

event trees. Section 2).2 discusses the loss of coolant accident (LOCA) event trees.

1) Loss of Cooling Event Trees

The loss of cooling event trees are developed and used to model the mitigation of loss of cooling event in each phase of an outage. With the phases and durations defined in section 2, eight event trees are needed for loss of cooling. One of them is shown in Figure 2 as an example.

The generic component failure data are used in calculatung the basic event probabilities used in the fault trees for the core damage suquences. Table 4 summarizes the results of the loss of cooling event trees. The results of generic shutdown risk is summarized in Table 5. The result shown in Table 5, i.e., 4.38E-05/year, indicates that the risk of a PWR at shutdown is not much lower than the risk when the plant is operating i.e., 1.5E-04/year, estimated in NUREG/CR-4550{11}.

2) LOCA Event Trees

Two types of LOCA are considered. The first type can be characterized by a stuck open RHR relief valve, and the leakage rate for this case is relatively low. Therefore, the operator may have a reasonable amount of time in which to isolate the LOCA. However, if the LOCA occurs in the RHR system, isolating the LOCA may require complete isolation of the RHR system. The second type of LOCA, for example, could be due to inadvertent opening of the containment spray header valves. The leakage rate in this case is much higher than the first type of LOCA, and less time would be available before the RHR pump would lose its NPSH. LOCA trees are developed for the two types of LOCA by modifying the loss of cooling event trees. The quantification of LOCA event trees are similar to that of the loss of cooling event trees and sixteen trees are developed for LOCA. Tables 6 summarizes the quantification of the LOCA event tree in case of stuck open RHR relief valve in the refueling outage as an example.

3) Other Initiating Events

It is assumed that the electric power system, component cooling water system, and service water system at Zion and the dependence of other systems on these systems are typical of those other PWRs. Therefore, the analysis done for Zion for loss of these support systems is considered applicable in the generic plant analysis.

The frequency of loss of CCW(LOCCW) was estimated to be 6.92E-09 per hour and loss of SW(LOSW) 1.07E-09 per hour by considering the recovery factor {4} and the Sandia Review of ZPSS{12}. The frequency of loss of offsite power has been taken from

Table 6 : Core Damage Frequency Due to LOCAs in a Refueling Outage-Stuck Open RHR Relief Valve

	Phase 1	Phase 2	Phase 3	Phase 4	Total
Duration(hr)	113	420	500	909	
Frequency (per year)	0.747	0.747	0.747	0.747	
F(LOCA) (per year)	4.25-4	1.58-3	1.88-3	3.42-3	7.31-3
P(CD/LOCA)	6.18-6	5.83-5	8.74-4	5.67-5	
f(CD) (per year)	2.63-9	9.21-8	1.64-6	1.94-7	1.93-6

Notes : 1.93-6=1.93E-06

Frequncy=Frequency of the phase of outage

f(LOCA)= Frequency that a LOCA occurs in the phase

P(CD/LOCA)=Conditional Probability of core damage given a LOCA

f(CD)=Frequency of Core Damge

UNREG-1032{13}, i.e., 1.0E-05 per hour. It has been assumed that the maintenance unavailabilities of components for the generic plant are the same as those for Zion. It has also been assumed that simultaneous maintenance of two diesel generators is not allowed. The offsite power recovery model in NUREG-1032{13} gives similar results to those of the ac power recovery model used in ASEP for Zion{11}. Therefore, the same staircase function inFigure 3. has been used in the generic analysis.

5. Benefits of the Proposed Improm192ents for a Generic Plant

In this section, several design/procedural improvements for the generic design configurations(here, it is assumed again the Zion plant represents the PWR population) are considered based on shutdown risk model described in Section 2. It should be emphasized that the improvements are intended to reduce the frequency of loss of cooling events and to improve the operator's ability to respond to loss of cooling events. Table 7 summarizes the benefits of these improvements when applied to the generic plant. Table 8 provides the detailed informations of the core damage frequency vs. the initiating events. It can be seen that removal of ACI is the most effective way to reduce the frequency of loss of coolingbut the least effective in reducing core darmage frequency. Upgraded instrumentation for the RHR system has no effect on the frequency of cooling but is the most effective way to reduce core damage frequency. The reductions in the frequency of loss of coolingor the frequency of core damage resulting from implementing upgraded vessel level indication and removal of the auto-closure interlock are additive, because these improvements affect the frequency of the initiating events. The benefits of upgrading the instrumentation for the RHR system can not be added to those of the other proposed imp-

Table 7. Summary of Benefits of the Proposed Improvements for a GenericPlant

	Base Case	A1	A2	A3
f(LC) (per year)	3.21E-01	3.21E-01	2.49E-01	1.71E-01
△f(LC) (per year)	N/A	0	7.20E-02	1.19E-01
CDF (per year)	4.38E-05	3.86E-05	4.08E-05	4.26E-05
△CDF (per year)	N/A	5.20E-06	3.00E-06	1.20E-06

Notes: A1=Upgraded Instrumentation for RHR Pumps
A2=Upgraded Vessel Level Indication
A3=Removal of Auto Closure Interlock
f(LC)=Frequency of Loss of Cooling
CDF=Core Damage Frequency

Table 8 : Summary of Core Damage Frequency Results for a Generic Plant at shutdown

Initiating Event	Core Damage Frquency (per year)			
	Base Case	A1	A2	A3
Loss of Cooling	8.48E-06	3.60E-06	5.41E-06	7.28E-06
LOCA	2.61E-06	2.44E-06	2.61E-06	2.61E-06
Loss of Offsite Power	5.31E-06	5.08E-06	5.31E-06	5.31E-06
Loss of CCWs	2.34E-05	2.34E-05	2.34E-05	2.34E-05
Loss of SWS	4.04E-06	4.04E-06	4.04E-06	4.04E-06
Total	4.28E-05	3.86E-05	4.08E-05	4.26E-05

Notes : A1. A2. A3=Same as Defined in Table 7 rovements, because this improvement affects the operator's ability to diagnose and respond to the initiating event.

6. Summary and Conclusion

The shutdown risk model is developed and analyzed using Event/Fault Tree for the typical pressurized water reactor. A dominant cause of core damage during a shutdown is due to the failure of the operator response. Operator performance depends on the information available to him. The improvements are intended to reduce the frequency of loss of cooling events and to improve the operator's ability to respond to loss of cooling events.

The results of this study indicate that the risk of a PWR at shutdown is not much lower than the risk when the plant is operating and several design/procedural changes may lead to significant reduction in core damage frequency due to loss of cooling events.

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