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SHUTDOWN HEAT REMOVAL SYSTEM RELIABILITY IN THERMAL REACTORS*

Y. H. Sun and R. A. Bari
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

ABSTRACT

An analysis of the failure probability per year of the shutdown heat removal system (SHRS) at hot standby conditions for two thermal reactor designs is presented. The selected reactor designs are the Pressurized Water Reactor and the Nonproliferation Alternative System Assessment Program Heavy Water Reactor. Failures of the SHRS following the initiating transients of loss of offsite power and loss of main feedwater system are evaluated. Common mode failures between components are incorporated in this analysis via the β-factor method and the sensitivity of the system reliability to common mode failures is investigated parametrically.

I. INTRODUCTION

Whenever a nuclear power plant is shut down, there is the need to remove stored and decay heat from the reactor core. Since there is sufficient heat to cause a meltdown of the reactor core even many days after the initial shutdown, it is important to ensure the reliability of the heat removal capability after reactor shutdown.

This paper contains an analysis for the shutdown heat removal systems (SHRS) for selected thermal reactor designs. Particular attention is given to the Nonproliferation Alternative System Analysis Program (NASAP) heavy-water reactor (HWR) [1]. Section II provides a brief description of the SHRS of the NASAP HWR. A fault tree analysis of the HWR SHRS is contained in Section III. Section IV contains a comparative study of the failure rate of the SHRS at hot standby conditions due to a loss of offsite power (LOSP) and due to a loss of main feedwater in the NASAP HWR and in certain pressurized water reactor (PWR) designs.

II. DESCRIPTION OF NASAP HWR SHUTDOWN HEAT REMOVAL SYSTEM

In the first thirty minutes after reactor shutdown while the temperature, pressure, and decay heat are high, the main heat transport system (MHTS) is used to remove residual heat from the reactor. At thirty minutes after shutdown, the reactor coolant system reaches 350°F and 400 psia and the shutdown cooling system (SCS) is used to cool the reactor to

135°F and atmosphere pressure at 4.5 hours after shutdown. In addition to the MHTS and the SCS, the moderator system may be regarded as a large, passive sink for shutdown heat removal. However, in the subsequent analysis, the reliability of the shutdown heat removal system is based on the reliability of the MHTS and the SCS and no credit is given for shutdown heat removal through the moderator system in either an active or passive mode of operation.

II.1 The Main Heat Transport System (MHTS)

The reactor coolant system (RCS) is comprised of two heat transport loops. Each loop contains 370 reactor pressure tubes, two RCS pumps, two steam generators, two inlet headers, two outlet headers, and interconnecting piping and valving. The two loops are connected to a common pressurizer and purification circuit; however, these loops can be isolated if an emergency condition should occur.

The main steam (MS) system transports steam from steam generators to the high pressure (HP) section of the turbine generator. Steam from four steam generators flows through four pipes to a main steam header. Each MS line includes a flow restrictor, power-operated atmospheric relief valve, safety valves, and main steam isolation valve (MSIV).

The main steam system is capable of removing heat from the reactor coolant system following sudden load rejection by automatically bypassing the main steam to the condenser through the turbine bypass system or by venting to the atmosphere through the main steam safety valves or main steam atmospheric dump valves, if the turbine bypass system is not available.

The condensate and main feedwater system returns condensed steam from the condenser while maintaining the water inventories throughout the system. Condensate pumped from the condenser hotwell will then pass through the high pressure feedwater heater to the steam generators.

II.2 Emergency Feedwater System (EFWS)

Upon loss of main feedwater flow, heat can be removed from the steam generator via the safety and relief valves provided that the coolant inventory is maintained by water makeup from the emergency feedwater system. The emergency feedwater system is designed to operate until the reactor coolant system pressure is reduced to a value below which the shutdown cooling system can be operated. The emergency feedwater system pumps unheated water from the condensate storage tank to the steam generators, and is comprised of one motor-driven pump, one turbine-driven pump and a system of piping, valves, and orifices.

II.3 Shutdown Cooling System (SCS)

This system cools the RCS from 350°F and 400 psia at 30 minutes after shutdown to 135°F and atmospheric pressure at 4.5 hours after shutdown. During shutdown cooling, a portion of the reactor coolant will flow out the shutdown cooling nozzles located on the reactor outlet headers. Coolant will be circulated through the shutdown coolant heat exchangers by the low pressure safety injection pumps and returned to the RCS inlet headers.

III. FAULT TREE CONSTRUCTION

A fault tree for the shutdown heat removal system was constructed and is given in Figure 1. In this construction, it is assumed that, at hot standby conditions, the main heat transport system is used for heat removal. At cold or hot shutdown conditions, both the SCS and the main heat transport system can be used for heat removal. The event that a large break at the inlet header prevents the ECCS water from entering the reactor core has also been included in the fault tree. The possibility of loss-of-core cooling capability due to a large number of simultaneous pressure tube failures is also considered. The heat dissipation through the power-operated atmospheric relief valves is included. The indicated transfers 1-4,9,c have been developed further but space does not permit their display. These fault trees can be obtained from the authors upon request.

IV. COMPARISON OF THE FAILURE RATES OF THE SHUTDOWN HEAT REMOVAL SYSTEMS FOR THE PRESSURIZED HEAVY WATER REACTOR AND THE PRESSURIZED LIGHT WATER REACTOR

A comparison is made of the shutdown heat removal system reliability at hot standby conditions of the NASAP HWR and the pressurized light water reactor (PWR). This comparison is made in order to place the NASAP HWR design in the perspective of the more familiar PWR. The designs are similar in the following aspects: 1) both have a high pressure, single-phase primary cooling system; 2) the HWR EFWS is the counterpart of the auxiliary feedwater system (AFWS) in the PWR; 3) both systems have high-head and low-head emergency core cooling systems; 4) the HWR SCS is the counterpart of the residual heat removal system in the PWR. It is interesting to note that recent studies [4,5] have found that there is great variability in the reliability of AFWS among the PWRs themselves. In this regard, the results of these studies may provide guidance to the optimization of the design of the SHRS of the NASAP HWR.

The reliability of SHRS following the initiating transients of loss of offsite power and loss of main feedwater is calculated. Potential common mode failures are analyzed by the β-factor method [2]. As appropriate, where components are expected to be similar, the same reference data (based on WASH-1400 [3]) are utilized. Table I lists the major differences in the SHRS of the two reactors. In these reactors, the feedwater system is used to maintain the water inventory in the steam generator. An auxiliary feedwater system (AFWS) must be able to supply feedwater following a loss of main feedwater supply. The reliability of AFWS was analyzed separately because this is then used in the analysis of the SHRS reliability.

Because of design differences (see Table I), the failure probability of the AFWS in each reactor is different. As indicated in Table I, the PWR AFWS contains one turbine-driven pump train and two motor-driven pump trains, while the HWR AFWS contains one turbine-driven pump train and one motor-driven pump train. Each pump train includes check valves, motor operated valves, manual valves, and pumps. Either of the HWR pumps have the capacity to supply sufficient feedwater to achieve mission success. In the PWR, however, mission success will depend on the sizing of the two motor-driven pumps. The turbine-driven pump can supply

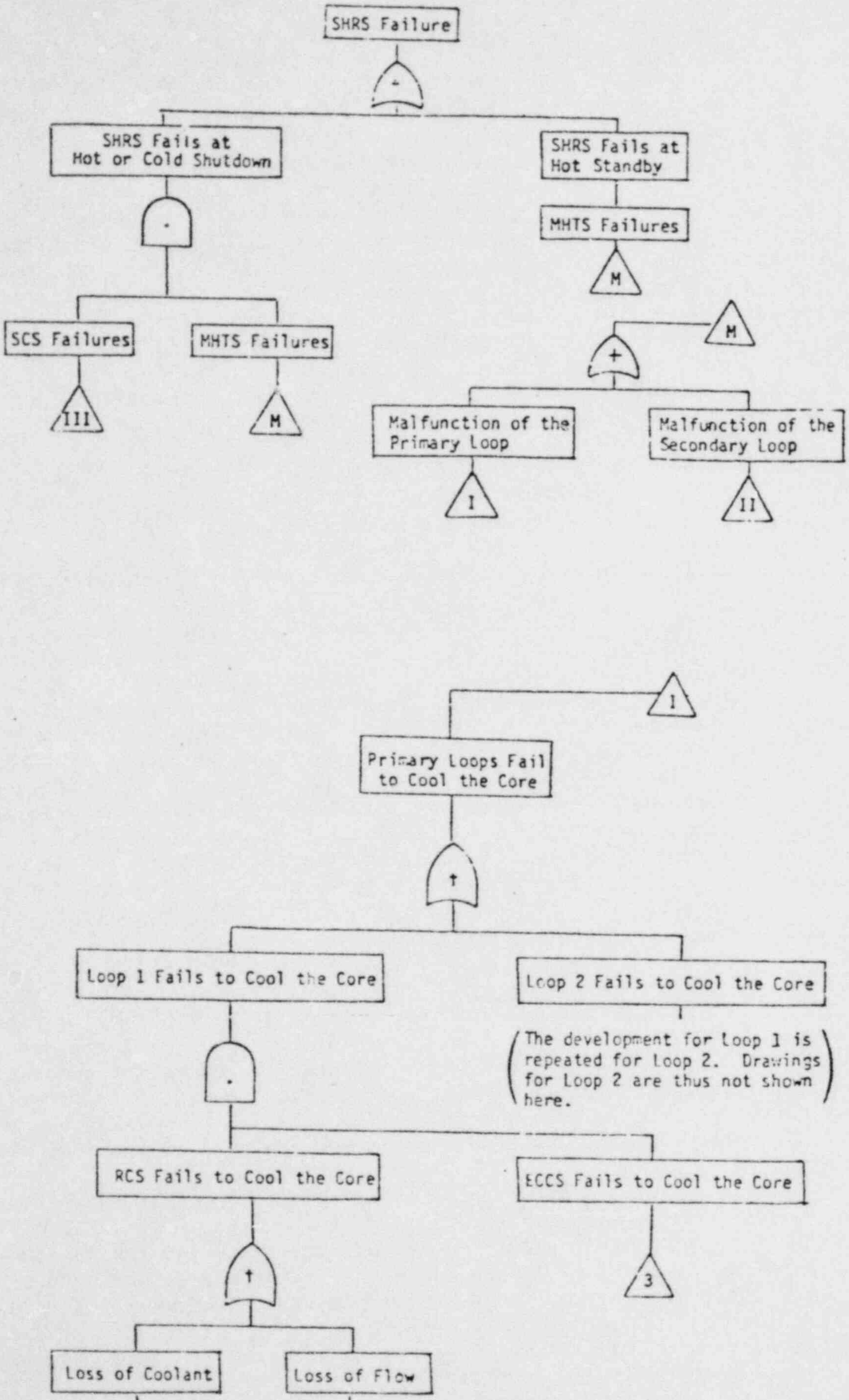
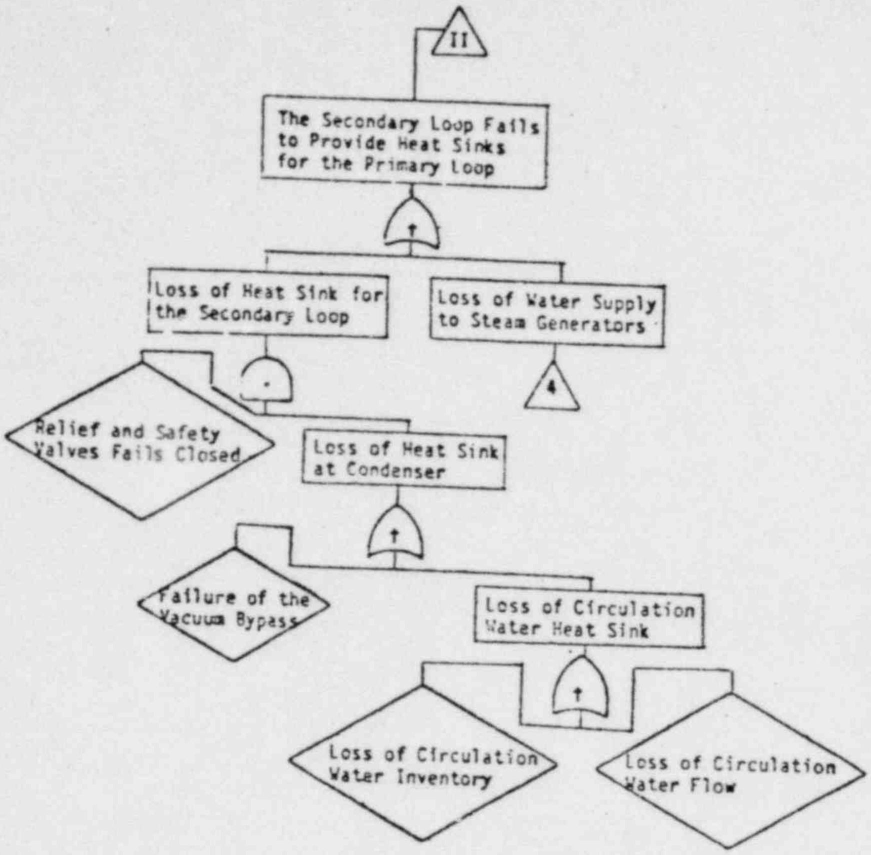
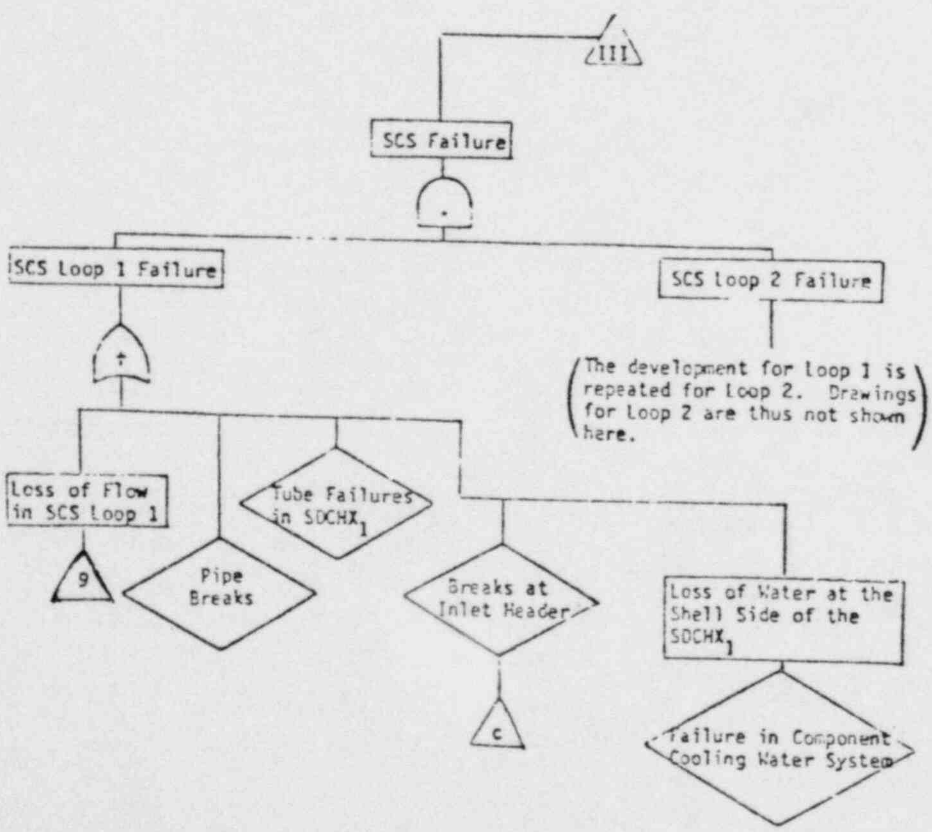


Fig. 1 Fault Tree for the SHRS in NASAP HWR





SDCHX: Shutdown Cooling Heat Exchanger

Fig. 1 (Cont.)

TABLE I

Subsystems of SHRS

	PWR	HWR
Active Heat Removal Systems	PHTS MFS AFWS SGS PCS	PHTS MFS EFWS (AFWS) SGS PCS
Passive Heat Sinks	Water in PHTS and SGS	Water in PHTS and SGS Large inventory of water in moderator system.
Other Heat Removal Capabilities	RHRS (for hot and cold shutdown). Assumed natural circulation. ECCS	SCS (for hot and cold shutdown). Assumed natural circulation.* ECCS
Pumps in AFWS	Two motor-driven pumps. One turbine-driven pump. Each motor-driven pump is half-capacity of the turbine-driven pump.	One motor-driven pump. One turbine-driven pump. Both have the same capacity.
AC Power Sources	Offsite power supply. Two diesel generators. One standby diesel generator.	Offsite power supply. Two diesel generators.

PHTS: Primary Heat Transport System
MFS: Main Feedwater System
AFWS: Auxiliary Feedwater System
SGS: Steam Generator System
PCS: Power Conversion System
EFWS: Emergency Feedwater System
RHRS: Residual Heat Removal System
SCS: Shutdown Cooling System
ECCS: Emergency Core Cooling System

*(less certainty because of the horizontal arrangement of pressure tubes and the large contact surface between the coolant and pressure tubes)

sufficient feedwater but either one or both motor-driven pumps may be required to supply adequate feedwater. For example, for the PWR analyzed in Reference 3, mission success was achieved by requiring only one motor-driven pump to supply water. On the other hand, in a recent study [5] it was assumed that mission success required both motor-driven pumps. Therefore, in the present analysis, the reliability calculations will be done both ways: $(PWR)_1$ - requires both motor-driven pumps for mission success; $(PWR)_2$ - requires either motor-driven pump for mission success.

Hardware failures, human error, and unavailability due to test and maintenance of each pump train have been factored into the analysis. The data were adopted from the Reactor Safety Study [3]. The β-factor method [2] was used to relate the possible common-mode events between the manual valves in the motor-driven train and the turbine-driven train and between the diesel generators.

The results for the AFWS failure probability (per demand) with AC power available are as follows:

HWR: $P_H = 1.4\times10^{-4} + (3.0\times10^{-3})\ \beta_v$

$(PWR)_1$: $P_1 = 2.8\times10^{-4} + (3.0\times10^{-3})\ \beta_v$

$(PWR)_2$: $P_2 = (3.0\times10^{-3})\ \beta_v$

Here, β_v is the β-factor for the manual valves and the above expressions are valid for $\beta_v \geq 0.1$.

From these expressions for the failure probabilities P_H, P_1, and P_2 of the AFWS, the failure rate (per year) of the SHRS can be obtained by multiplying P_H, P_1, or P_2 by the frequency of loss of main feedwater. The NASAP study was not performed for a particular site, but it is generally regarded that between one and three loss-of-main-feedwater events per year will occur at a given plant.

If offsite power is not available, then the failure probability (per demand) of the AFWS in each case is as follows:

HWR: $P_H' = P_H + (4\times10^{-4})\ \beta_D + 10^{-5}\ (1 - \beta_D)^2$

$(PWR)_1$: $P_1' = P_1 + (4\times10^{-4})\ \beta_D$

$(PWR)_2$: $P_2' = P_2 + (4\times10^{-4})\ \beta_D$

Here, β_D is the β-factor which accounts for common mode failures between the diesel generators. Note that for the PWRs a third standby diesel is included in the analysis and only the common mode contribution to the diesel generator failure probability is included (for $\beta_D \geq .01$). For the HWR configuration it is seen that common mode failures dominate the diesel failure probability for $\beta_D \geq .05$. The results of these calculations show that the failure of the manual control valves due to a common mode failure (due mainly to human error) contributes significantly to the AFWS failure probability when β_v is greater than 0.1 in both PWRs and in the HWR. Failures of the diesel generators, together with the turbine-driven pump train, do not significantly contribute to the AFWS failure probability unless $\beta_D \geq 7.5\beta_v$. However, for $\beta_v \geq 0.1$, this would imply an unacceptable high unavailability for onsite AC power.

The expressions P_H', P_1', and P_2' can be used to derive the failure rate of the SHRS due to loss of offsite power by multiplying each by the frequency of loss-of-offsite power. Again, generally recognized values are in the range 0.1-0.3 per plant-year.

V. SUMMARY

An analysis of the shutdown heat removal system reliability of the NASAP HWR has been presented and the results have been compared to two variations of the shutdown heat removal systems of the pressurized light water reactor. The β-factor method for quantifying common mode failure has been utilized to compute the SHRS failure rates which result from loss of main feedwater and from loss of offsite power. It was shown that the failure probability of the AFWS in the HWR is bracketed by that of the two variations of the PWR analyzed in this study.

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REFERENCES

1. Preliminary Safety and Environmental Information Document, Vol. II, HWR, NASAP, Department of Energy, April 1979, and Combustion Engineering NASAP HWR Preliminary Design Document, Chapter 5, (unpublished).

2. K. N. Fleming, "A Reliability Model for Common Mode Failures in Redundant Safety Systems," Proceedings of the 6th Annual Pittsburgh Conference on Modeling and Simulation, April 1975.

3. Reactor Safety Study, WASH-1400 (NUREG-75/014), U. S. NRC, October 1975.

4. M. A. Taylor et al., "An Assessment of Auxiliary Feedwater Systems," Trans. Am. Nucl. Soc. 33, 569 (1979).

5. R. S. Enzinna et al., "Auxiliary Feedwater Reliability Analyses for Plants with B&W Designed NSS's, to appear in ANS Transactions for the 1980 Annual Meeting, Las Vegas, June 8-13, 1980.