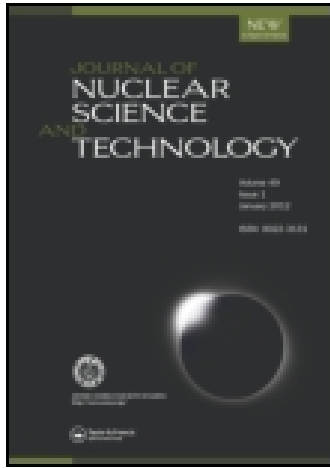


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Reliability Analysis of Emergency Decay Heat Removal System of Nuclear Ship under Various Accident Conditions

Comparison between Nuclear Ship "Mutsu" and Nuclear Ship "Savannah"

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A reliability analysis is given for the emergency decay heat removal system of the Nuclear Ship "Mutsu" and the emergency sea water cooling system of the Nuclear Ship "Savannah", under ten typical nuclear ship accident conditions. Basic event probabilities under these accident conditions are estimated from literature survey.

These systems of Mutsu and Savannah have almost the same reliability under the normal condition. The dispersive arrangement of a system is useful to prevent the reduction of the system reliability under the condition of an accident restricted in one room.

As for the reliability of these two systems under various accident conditions, it is seen that the configuration and the environmental condition of a system are two main factors which determine the reliability of the system. Furthermore, it was found that, for the evaluation of the effectiveness of safety system of a nuclear ship, it is necessary to evaluate its reliability under various accident conditions.

KEYWORDS: *reliability analysis, fault tree, failure data, failure probability, nuclear ship, emergency decay heat removal system, safety system, accident condition, extreme environmental condition, conditional failure rate*

I. INTRODUCTION

Peculiar considerations for safety systems are required for the design of a nuclear ship, as compared with a nuclear power generating plant, because a nuclear ship navigates on the sea far from the land where technical support is hardly available and has a possibility to encounter marine disasters.

If some accident occurs to nuclear ship, the "scram" and "removal of decay heat" are essential to prevent a reactor accident. The nuclear ship "Savannah" has the emergency sea water cooling system for decay heat removal, of which operation is required when all other safety systems have failed.

The reliability of this system under the normal environmental condition has already been analyzed⁽¹⁾. The operation of this system, however, is mainly required when some marine disaster occurs. In this situation, the system is exposed to extreme environmental conditions such as high temperature, humidity, impact, vibration, stress *etc.* The reliability of a system under these environmental conditions is usually different from that under the

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normal condition. Therefore, it is the reliability of a safety system under various extreme environmental conditions that should be evaluated in detail.

The first major application of probabilistic and reliability analysis techniques was the Reactor Safety Study (*WASH-1400*)⁽²⁾. Since the completion of the Reactor Safety Study, much work⁽³⁾⁽⁴⁾ has been performed in the probabilistic analysis to nuclear power plants. In these studies, the effects of extreme environmental conditions were not explicitly considered. Tachikawa *et al.*⁽⁵⁾ have performed the reliability analysis on marine reactor systems, but the accident conditions of nuclear ship were not taken into account to the analysis.

In the present study, the reliability of the emergency decay heat removal system of Mutsu and of the emergency sea water cooling system of Savannah is analyzed under extreme environmental conditions using fault tree analysis technique. Occurrence probabilities of basic events under extreme environmental conditions are estimated based on literature survey.

II. FUNCTIONS AND CONFIGURATIONS OF EMERGENCY DECAY HEAT REMOVAL SYSTEM AND EMERGENCY SEA WATER COOLING SYSTEM

Savannah has the emergency sea water cooling system and Mutsu has the emergency decay heat removal system as a safety system. The main function of these systems is to remove the decay heat from the reactor when normal power sources have failed and there are no other means to remove the decay heat⁽⁶⁾⁽⁷⁾. These systems are designed to prevent excessive rise of fuel cladding temperature and possible damage of core.

Figure 1 shows the main part of the emergency sea water cooling system of Savannah. Almost all the equipments of this system are located inside the containment vessel. Almost all other equipments vital to the reactor operation are also contained in this containment vessel.

This system is composed of three main loops: primary water loop, cooling water loop and make-up water loop. The make-up water loop and the containment cooling coil are not shown in Fig. 1, because they do not relate to the main function of this system. In the primary water loop, water is taken from the primary purification system letdown line at a point upstream from the letdown cooler, and then pumped by the emergency canned pump through the emergency cooler and control valves, and back to the reactor pressure vessel.

The cooling water loop is arranged to circulate either sea water or fresh water by the emergency sea water pump, through the emergency cooler, the containment cooling coil and the cooling coil of the emergency canned pump in parallel. Sea water being used in an emergency is taken from the starboard and/or the port sea chests. All the sea water is discharged overboard through a common discharge line. The sea chests and sea water discharger are located sufficiently below the light load line of the ship.

The emergency diesel generator is located on the navigation deck which is on the upper part of the ship. An automatic control gear starts this generator when the voltage on the main buses drops below 380 V. In addition it can also be started manually from both the main console and the emergency switch board which is located in the emergency generator room. Necessary pumps and motor operated valves are also controlled by switches on this board.

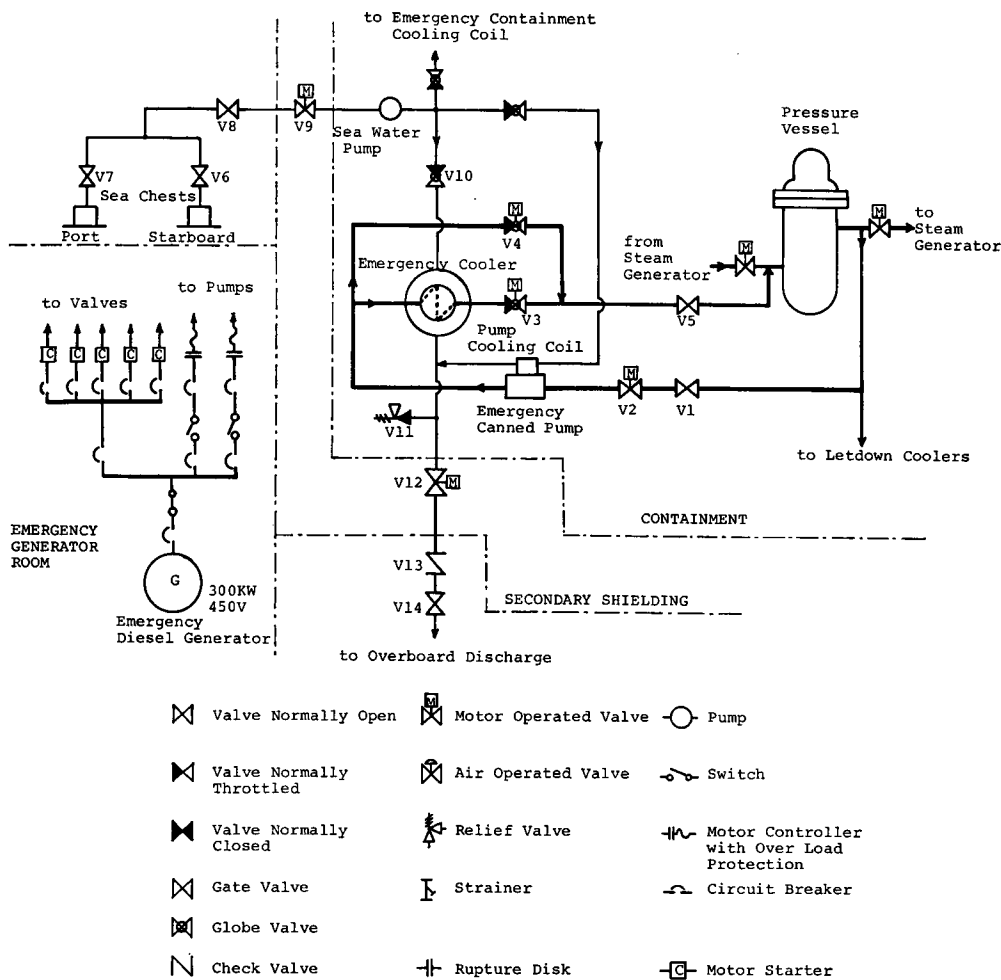


Fig. 1 Emergency sea water cooling system of "Savannah"

Figure 2 shows the emergency decay heat removal system of Mutsu. Equipments of this system are separately placed in many rooms. They are mostly in the containment, auxiliary machine room, machine room and reactor room. Two steam generators are used as the heat exchangers in emergency conditions.

This system is composed of two main loops: one which circulates primary water and one which circulates cooling water. The primary water is circulated through the primary loops which are used in an emergency as well as ordinary operations. The cooling water loop is arranged to circulate the fresh water from the emergency water tank. This water flows through the emergency decay heat removal pump and then splits into two parts and is supplied to steam generators A and B. The decay heat can effectively be removed from the core by one steam generator alone. The water becomes saturated steam at the steam generators by receiving the decay heat from the primary water. These two steam generators have a common discharge line. The saturated steam is discharged overboard through a stack.

The emergency diesel generator is located in the emergency generator room which is

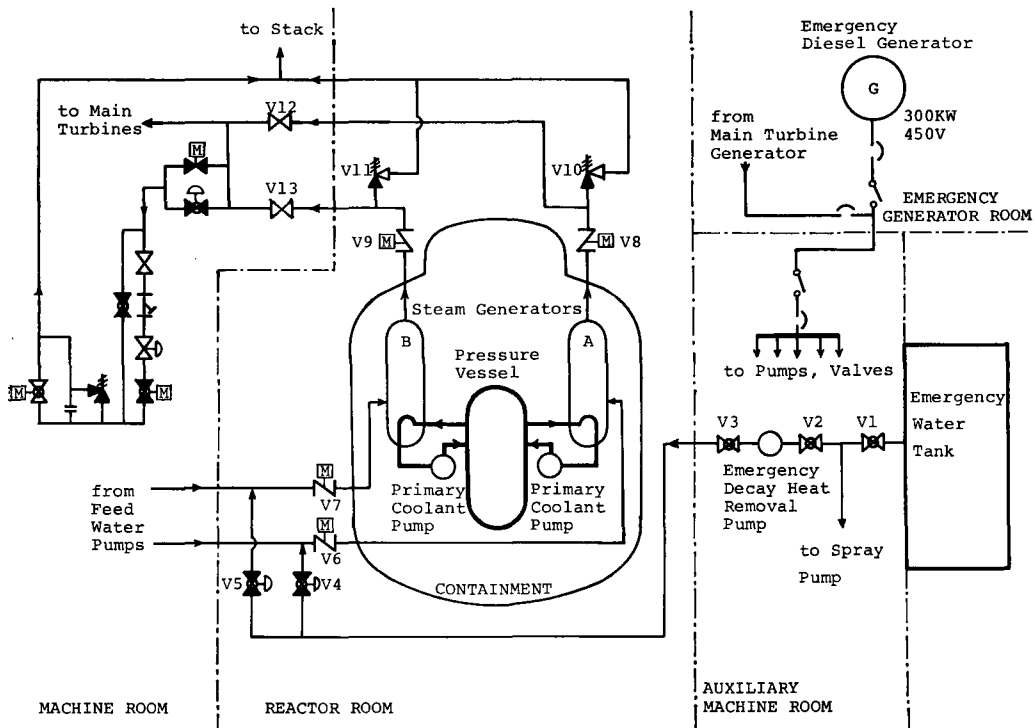


Fig. 2 Emergency decay heat removal system of "Mutsu"

on the upper part of the ship. The emergency diesel engine will be started automatically when both the main turbine and the auxiliary generators are not available. In addition it can be started manually from the main console room and also from the emergency generator room.

III. ANALYSIS METHOD

1. Procedure of Analysis

In the present study, reliability analysis of the systems is performed based on the following procedure:

- (1) Identify all the conceivable failure modes of the components by the failure mode and effects analysis (FMEA)⁽⁸⁾
- (2) Determine an undesired event (top event) of the system
- (3) Construct a fault tree, in which the information developed in the FMEA is arranged as basic events and the critical aspect of the system behavior is pointed out
- (4) Assign probabilities of the basic events under the normal condition
- (5) Construct a reduced fault tree by neglecting events possessing a low probability
- (6) Calculate the probability distribution of the top event by using the computer program SAMPLE⁽⁹⁾, which has been developed in WASH-1400; the median, upper bound and lower bound values of the probability are obtained from this distribution
- (7) Select important accident conditions under which a system reliability should be evaluated
- (8) Estimate probabilities of the basic events under these accident conditions

- (9) Calculate the probability distributions of the top event under these accident conditions.

2. Assumptions Made in Analysis

In this study, the following assumptions were made in order to simplify the analysis: (1) The components in the same room are exposed to the same environmental condition; (2) The structure of the fault tree under the accident condition is the same as the one under the normal condition, and only the occurrence probability of the basic event under accident condition differs from the one under the normal condition; (3) The decay heat can be removed by the natural circulation in the primary loop; and (4) The maintenance interval is 100 days, and at every maintenance, all the components are checked and recovered to the normal condition. The above assumptions were equally applied to the safety systems of the two ships.

3. Top Event

A top event of a fault tree must be clearly defined and its interpretation should not vary during the construction process of the fault tree. In the present study the top event was defined for both systems (the emergency decay heat removal system and the emergency sea water cooling system) as follows: Failure to remove the decay heat from the reactor core within 10 h after the initiation of a demand.

After 10 h from the reactor shutdown, the decay heat decreases below 0.7% of the total thermal power, and therefore, even if all the decay heat removal systems stop at this point, the heat capacity of the water contained in the pressure vessel and the steam generators is enough to prevent reactor core damage for considerable duration of time. This time duration is useful for the repair of the system under many accident conditions. In the worst case, however, the sea water can be poured into the containment through the containment shell flooding valves as a heat sink for the decay heat. Therefore, the requirement that the operation of the emergency decay heat removal system and the emergency sea water cooling system should be guaranteed for 10 h after shutdown is sufficiently reasonable.

4. Accident Conditions

A nuclear ship has a possibility to encounter many accidents such as fire, explosion, collision, flooding, stranding, overthrow and so on. In addition, some accidents produce another accidents and therefore the consequences of nuclear ship accidents are very complicated. In the present study, typical and fundamental accidents were selected in order to treat this complicated situation. They are shown in Table 1. It is assumed that these accidents are not so severe as the operation of the safety systems is of no use in preventing the development of accidents. Fire and explosion accidents in Table 1 are restricted to those occurred in one room. Except collision and flooding, accidents extending to more than one room are not treated in the present study.

Various kinds of collision accidents will occur to a nuclear ship. We select in the present study the one where no components of these safety systems are directly destroyed but they are affected to the same degree. The accident condition "impact" in Table 1, corre-

Table 1 Typical accidents of nuclear ship

1	Explosion in containment
2	Fire in containment
3	Explosion in emergency generator room
4	Fire in emergency generator room
5	Explosion in reactor room
6	Fire in reactor room
7	Explosion in auxiliary machine room
8	Fire in auxiliary machine room
9	Impact
10	Flooding

sponds to this condition.

There are also various kinds of floodings, and the feature of flooding accidents depends on their causes and processes. The following is the condition selected as a flooding accident: the machine room is submerged and rooms other than the containment are partially flooded. The containment is usually more durable than other rooms, and, therefore, the components inside the containment are assumed to be at the normal condition in this flooding accident.

IV. FAILURE DATA

The component failure rate under the normal condition is deduced from WASH-1400, Appendix III "Failure Data"⁽¹⁰⁾, in which comprehensive failure data have been assessed for the components of nuclear power plants. The log-normal distribution has been used to describe the data variability. The median value of the distribution has been given and the assessed 90% range has been expressed by the error factor. Table 2(a),(b) show failure rate and failure probability of each component under the normal condition thus obtained.

Failure rates under the extreme environmental conditions are required to calculate the system failure probabilities under accident conditions. The WASH-1400 has given failure data under extreme conditions for pump and motor as shown in Table 3. It can be seen that 10~100 times greater values have been given to the failure rates at extreme condition as compared with the values under the normal condition.

Table 2(a) Failure rates and failure probabilities (Savannah)

Basic Events		Number of Component	Mission Type	Failure Rates (Error Factor)	Failure Probabilities (Error Factor)
S1	Diesel Generator Fails to Start	1	Q_d	3×10^{-2} (3)	3×10^{-2} (3)
S2	Diesel Generator Fails to Run	1	λ_o	3×10^{-3} (10)	3×10^{-2} (10)
S3	Switch Fails to Operate	1	Q_d	1×10^{-3} (3)	1×10^{-3} (3)
S4	Breakers Fail to Close	6	Q_d	1×10^{-3} (3)	6×10^{-3} (3)
S5	Electric Circuit Open	1	λ_o	3×10^{-6} (3)	3×10^{-5} (3)
S6	Manual Valves V1, V5 Fail to Remain Open	2	Q_d	1×10^{-4} (3)	2×10^{-4} (3)
S7	Motor Operated Valve V3 Fails to Operate	1	Q_d	1×10^{-3} (3)	1×10^{-3} (3)
S8	Motor Operated Valve V3 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
S9	Motor Operated Valve V2 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
S10	Manual Valve V10 Fails to Open	1	Q_d	1×10^{-5} (3)	1×10^{-5} (3)
S11	Manual Valve V10 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
S12	Emergency Cooler Plug	1	λ_o	1×10^{-9} (30)	1×10^{-8} (30)
S13	Emergency Cooler Rupture	1	λ_o	1×10^{-9} (30)	1×10^{-8} (30)
S14	Emergency Sea Water Pump Fails to Start	1	Q_d	1×10^{-3} (3)	1×10^{-3} (3)
S15	Emergency Sea Water Pump Fails to Run	1	λ_o	3×10^{-5} (10)	3×10^{-4} (10)
S16	Relief Valve V11 Fails to Open	1	Q_d	1×10^{-5} (3)	1×10^{-5} (3)
S17	Motor Operated Valve V9 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
S18	Motor Operated Valve V12 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
S19	Manual Valve V8 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
S20	Manual Valve V14 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
S21	Failure in Check Valve V13 ; Plug	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
S22	Pipe (Secondary) Plug	5 sections	λ_o	1×10^{-10} (30)	5×10^{-9} (30)
S23	Pipe (Secondary) Rupture	5 sections	λ_s	1×10^{-10} (30)	6×10^{-7} (30)
S24	Pipe (Primary) Plug	5 sections	λ_o	1×10^{-11} (30)	5×10^{-10} (30)
S25	Pipe (Primary) Rupture	5 sections	λ_s	1×10^{-11} (30)	6×10^{-8} (30)

Q_d : Demand Failure Probability (/d)

λ_o : Operating Failure Rate (/hr), Operating time = 10 hours

λ_s : Stand-by Failure Rate (/hr), Stand-by Time Duration = 1200 hours

Table 2(b) Failure rates and failure probabilities (Mutsu)

Basic Events		Number of Component	Mission Type	Failure Rates (Error Factor)	Failure Probabilities (Error Factor)
M1	Diesel Generator Fails to Start	1	Q_d	3×10^{-2} (3)	3×10^{-2} (3)
M2	Diesel Generator Fails to Run	1	λ_o	3×10^{-3} (10)	3×10^{-2} (10)
M3	Switches Fail to Operate	2	Q_d	1×10^{-3} (3)	2×10^{-3} (3)
M4	Breakers Fail to Close	2	Q_d	1×10^{-3} (3)	2×10^{-3} (3)
M5	Electric Circuit Open	1	λ_o	3×10^{-6} (3)	3×10^{-5} (3)
M6	Pipe (Primary) Rupture	2 sections	λ_s	1×10^{-11} (30)	2.4×10^{-8} (30)
M7	Emergency Tank Leak or Rupture	1	λ_s	1×10^{-10} (30)	1.2×10^{-7} (30)
M8	Pump Fails to Start	1	Q_d	1×10^{-3} (3)	1×10^{-3} (3)
M9	Pump Fails to Run	1	λ_o	3×10^{-5} (10)	3×10^{-4} (10)
M10	Manual Valve V1 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
M11	Manual Valve V2 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
M12	Manual Valve V3 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
M13	Pipe (Secondary) Rupture	6 sections	λ_s	1×10^{-10} (30)	7.2×10^{-7} (30)
M14	Pipe (Secondary) Plug	6 sections	λ_o	1×10^{-10} (30)	6×10^{-9} (30)
M15	Air Operated Valve V4 Fails to Operate	1	Q_d	1×10^{-3} (3)	1×10^{-3} (3)
M16	Air Operated Valve V4 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
M17	Motor Operated Valve V6 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
M18	Motor Operated Valve V8 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)
M19	Relief Valve V10 Fails to Open	1	Q_d	1×10^{-5} (3)	1×10^{-5} (3)
M20	Relief Valve V11 Fails to Open	1	Q_d	1×10^{-5} (3)	1×10^{-5} (3)
M21	Pipe (Secondary) Rupture	3 sections	λ_s	1×10^{-10} (30)	3.6×10^{-7} (30)
M22	Pipe (secondary) Plug	3 sections	λ_o	1×10^{-10} (30)	3×10^{-9} (30)
M23	Manual Valve V12 Fails to Remain Open	1	Q_d	1×10^{-4} (3)	1×10^{-4} (3)

Q_d : Demand Failure Probability (/d)

λ_o : Operating Failure Rate (/hr), Operating Time = 10 hours

λ_s : Stand-by Failure Rate (/hr), Stand-by Time Duration = 1200 hours

Table 3 Failure rates given in WASH-1400

Basic events	Failure rates
Pump fails to run	
Normal environment	3×10^{-5} /h (10)
Extreme, post accident environments, inside containment	1×10^{-3} /h (10)
Post accident, after environmental recovery	3×10^{-4} /h (10)
Motor fails to run	
Normal environment	1×10^{-5} /h (10)
Extreme environment	1×10^{-8} /h (10)

Table 4 Failure rates under extreme conditions (Green & Bourne)

Components	Failure rates (/h)
Circuit breakers	
General (less than 33 kV)	2.0×10^{-6}
415 V-11 kV	1.5×10^{-6}
33 kV	3.0×10^{-6}
132 kV	4.0×10^{-6}
275 kV	7.0×10^{-6}
400 kV	1.0×10^{-5}
Ball bearing	
Heavy duty	2.0×10^{-5}
Light duty	1.0×10^{-5}
Shafts	
Heavily stressed	2.0×10^{-7}
Lightly stressed	2.0×10^{-8}
Springs	
Heavily stressed	1.0×10^{-6}
Lightly stressed	2.0×10^{-7}
Hoses	
Heavily stressed	4.0×10^{-5}
Lightly stressed	4.0×10^{-6}

Green & Bourne⁽¹¹⁾ have given more detailed failure rates under extreme conditions as shown in Tables 4 and 5. “ K_1 ”, “ K_2 ” and “ K_3 ” in Table 5, are multiplying factors which deduce the failure rate under any particular condition from the failure rate under normal condition; the “Rating” indicates the magnitude of the load applied to a component, and the value of 100 corresponds to the rated load; the “temperature”

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means the environmental temperature to which a component is exposed. The failure rates of a factor to two orders of magnitude greater than those under the normal condition have been given in these tables.

Table 5 Multiplying factors to failure rate (Green & Bourne)

Overall environment	K_1	Rating	K_2	Temperature (°C)	K_3
Ideal static condition	0.1	140	4.0	0	1.0
Vibration free, controlled environment	0.5	120	2.0	20	1.0
General purpose, grand-based	1.0	100	1.0	40	1.3
Ship	2.0	80	0.6	60	2.0
Road	3.0	60	0.3	80	4.0
Rail	4.0	40	0.2	100	10.0
Air	10.0	20	0.1	120	30.0
Missile	100.0				

In WASH-1400⁽¹²⁾, the probability of pipe rupture caused by a pump flywheel fracture accident has been estimated with some assumptions. The estimated value is 1.3×10^{-2} . In the present study, this value is taken into consideration for the estimation of the pipe rupture probability at explosion accidents.

Table 6(a),(b) summarize the failure probability of each component under the specified accident conditions, where the mark “*” means that the failure probability is equal to the one under the normal condition.

V. RESULTS AND DISCUSSIONS

Figure 3(a),(b) show the reduced fault trees of the emergency decay heat removal system and the emergency sea water cooling system, respectively. The basic events are explained in Table 2. In Fig. 3(b), events concerning the flow system in the machine room do not appear since the contribution of them to the top event is small as compared with those of other branches.

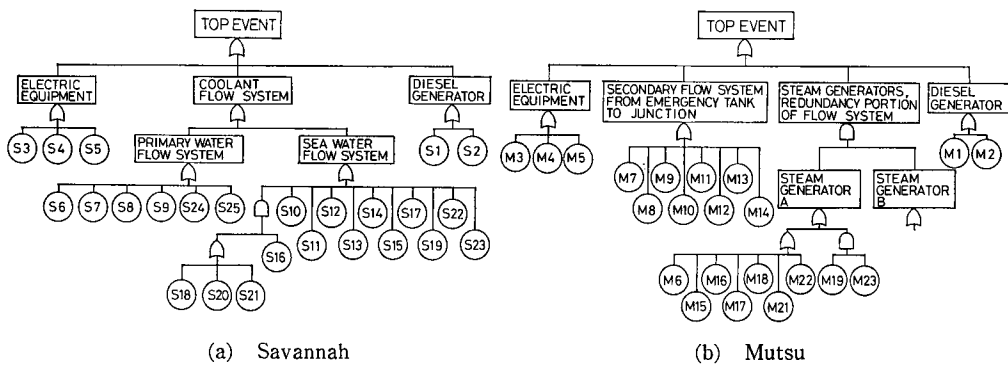


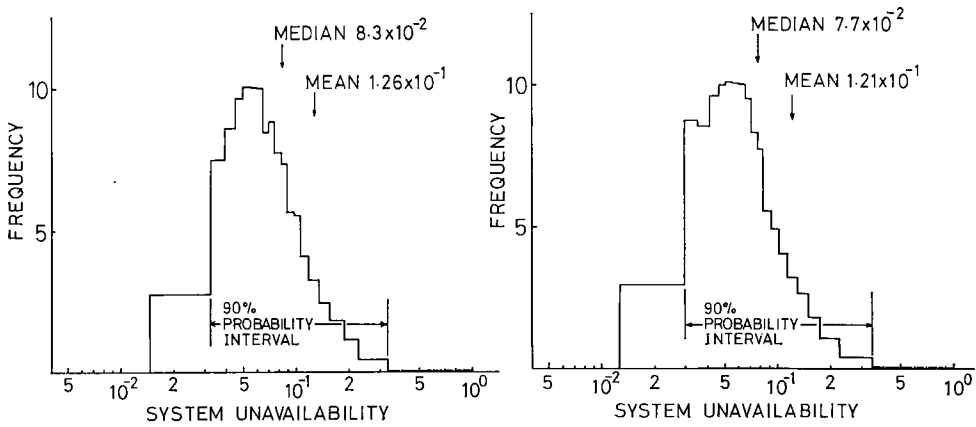
Fig. 3(a),(b) Reduced fault tree for emergency decay heat removal system

The calculated distributions of occurrence probability of top event under the normal condition are shown in **Fig. 4(a),(b)**. The distributions are almost same each other. The median, upper bound and lower bound of them were 7.7×10^{-2} , 3.5×10^{-1} and 2.9×10^{-2} for Mutsu, and 8.3×10^{-2} , 3.3×10^{-1} and 3.3×10^{-2} for Savannah, respectively.

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Table 6(a) Failure probabilities under various accident conditions (Savannah)

Basic Events	Accident Conditions					
	Explosion in Containment	Fire in Containment	Explosion in Emergency Generator Room	Fire in Emergency Generator Room	Impact	Flooding
S1	*	*	6×10^{-2} (3)	4×10^{-2} (3)	4×10^{-2} (3)	4×10^{-2} (3)
S2	*	*	6×10^{-2} (3)	4×10^{-2} (10)	4×10^{-2} (10)	4×10^{-2} (10)
S3	*	*	1×10^{-2} (10)	3×10^{-3} (3)	3×10^{-3} (3)	3×10^{-3} (3)
S4	*	*	6×10^{-2} (10)	1.8×10^{-2} (3)	1.8×10^{-2} (3)	1.8×10^{-2} (3)
S5	*	*	1×10^{-2} (10)	3×10^{-3} (3)	*	*
S6	2×10^{-3} (3)	2×10^{-3} (3)	*	*	*	*
S7	1×10^{-2} (3)	1×10^{-2} (3)	*	*	3×10^{-3} (3)	*
S8	*	1×10^{-3} (3)	*	*	3×10^{-4} (3)	*
S9	1×10^{-3} (3)	1×10^{-3} (3)	*	*	3×10^{-4} (3)	*
S10	1×10^{-3} (3)	1×10^{-3} (3)	*	*	1×10^{-4} (3)	*
S11	1×10^{-3} (3)	1×10^{-3} (3)	*	*	*	*
S12	*	*	*	*	*	*
S13	1.2×10^{-4} (30)	1.2×10^{-5} (30)	*	*	1.2×10^{-5} (30)	*
S14	1×10^{-2} (10)	1×10^{-2} (3)	*	*	1×10^{-3} (3)	*
S15	3×10^{-3} (10)	1×10^{-2} (10)	*	*	*	*
S16	1×10^{-3} (3)	1×10^{-3} (3)	*	*	1×10^{-4} (3)	*
S17	*	*	*	*	3×10^{-4} (3)	*
S18	*	*	*	*	3×10^{-4} (3)	*
S19	*	*	*	*	*	*
S20	*	*	*	*	*	*
S21	*	*	*	*	*	*
S22	*	*	*	*	*	*
S23	5×10^{-2} (10)	6×10^{-5} (30)	*	*	6×10^{-4} (30)	*
S24	*	*	*	*	*	*
S25	5×10^{-2} (10)	6×10^{-6} (30)	*	*	6×10^{-6} (30)	*



(a) Savannah (b) Mutsu

Fig. 4(a), (b) Probability distribution of top event

The median value for Mutsu is a little smaller than that for Savannah. This is due to the redundancy of the heat exchanger portion of the emergency decay heat removal system.

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Table 6(b) Failure probabilities under various accident conditions (Mutsu)

Basic Events	Accident Conditions									
	Explosion in Containment	Fire in Containment	Explosion in Emergency Generator Room	Fire in Emergency Generator Room	Explosion in Reactor Room	Fire in Reactor Room	Explosion in Auxiliary Machine Room	Fire in Auxiliary Machine Room	Impact	Flooding
M1	*	*	$6 \times 10^{-2} (3)$	$4 \times 10^{-2} (3)$	*	*	*	*	$4 \times 10^{-2} (3)$	$4 \times 10^{-2} (3)$
M2	*	*	$6 \times 10^{-2} (3)$	$4 \times 10^{-2} (10)$	*	*	*	*	$4 \times 10^{-2} (10)$	$4 \times 10^{-2} (10)$
M3	*	*	$1 \times 10^{-2} (10)$	$3 \times 10^{-3} (3)$	*	*	$1 \times 10^{-2} (10)$	$3 \times 10^{-3} (3)$	$6 \times 10^{-3} (3)$	$6 \times 10^{-3} (3)$
M4	*	*	$1 \times 10^{-2} (10)$	$3 \times 10^{-3} (3)$	*	*	$1 \times 10^{-2} (10)$	$3 \times 10^{-3} (3)$	$3 \times 10^{-3} (3)$	$6 \times 10^{-3} (3)$
M5	*	*	$1 \times 10^{-2} (10)$	$3 \times 10^{-3} (3)$	*	*	$1 \times 10^{-2} (10)$	$3 \times 10^{-3} (3)$	*	*
M6	$2 \times 10^{-2} (10)$	$2.4 \times 10^{-6} (30)$	*	*	*	*	*	*	$2.4 \times 10^{-6} (30)$	*
M7	*	*	*	*	*	*	*	*	$1.2 \times 10^{-5} (30)$	*
M8	*	*	*	*	*	*	$1 \times 10^{-2} (10)$	$1 \times 10^{-2} (3)$	$1 \times 10^{-2} (3)$	$1 \times 10^{-2} (3)$
M9	*	*	*	*	*	*	$3 \times 10^{-3} (10)$	$1 \times 10^{-2} (10)$	$1 \times 10^{-2} (10)$	$3 \times 10^{-3} (10)$
M10	*	*	*	*	*	*	$1 \times 10^{-3} (10)$	$1 \times 10^{-3} (3)$	*	*
M11	*	*	*	*	*	*	$1 \times 10^{-3} (10)$	$1 \times 10^{-3} (3)$	*	*
M12	*	*	*	*	*	*	$1 \times 10^{-3} (10)$	$1 \times 10^{-3} (3)$	*	*
M13	*	*	*	*	$2 \times 10^{-2} (10)$	$2.4 \times 10^{-5} (10)$	$4 \times 10^{-2} (10)$	$4.8 \times 10^{-5} (30)$	$7.2 \times 10^{-4} (30)$	*
M14	*	*	*	*	*	*	*	*	*	*
M15	*	*	*	*	$1 \times 10^{-2} (3)$	$1 \times 10^{-2} (3)$	*	*	$3 \times 10^{-3} (3)$	*
M16	*	*	*	*	*	$1 \times 10^{-3} (3)$	*	*	$3 \times 10^{-4} (3)$	*
M17	*	*	*	*	$1 \times 10^{-3} (3)$	$1 \times 10^{-3} (3)$	*	*	$3 \times 10^{-4} (3)$	*
M18	*	*	*	*	$1 \times 10^{-3} (3)$	$1 \times 10^{-3} (3)$	*	*	$3 \times 10^{-4} (3)$	*
M19	*	*	*	*	$1 \times 10^{-3} (3)$	$1 \times 10^{-3} (3)$	*	*	$1 \times 10^{-4} (3)$	*
M20	*	*	*	*	$1 \times 10^{-3} (3)$	$1 \times 10^{-3} (3)$	*	*	$1 \times 10^{-4} (3)$	*
M21	$3 \times 10^{-2} (10)$	$3.6 \times 10^{-5} (30)$	*	*	$3 \times 10^{-2} (10)$	$3.6 \times 10^{-5} (30)$	*	*	$3.6 \times 10^{-4} (30)$	*
M22	*	*	*	*	*	*	*	*	*	*
M23	*	*	*	*	*	*	*	*	*	*

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The distributions of occurrence probability of top event under accident conditions based on the values in Table 6, are shown in Fig. 5. The median values are indicated by the circles and 90% confidence bounds are shown by the length of the horizontal lines.

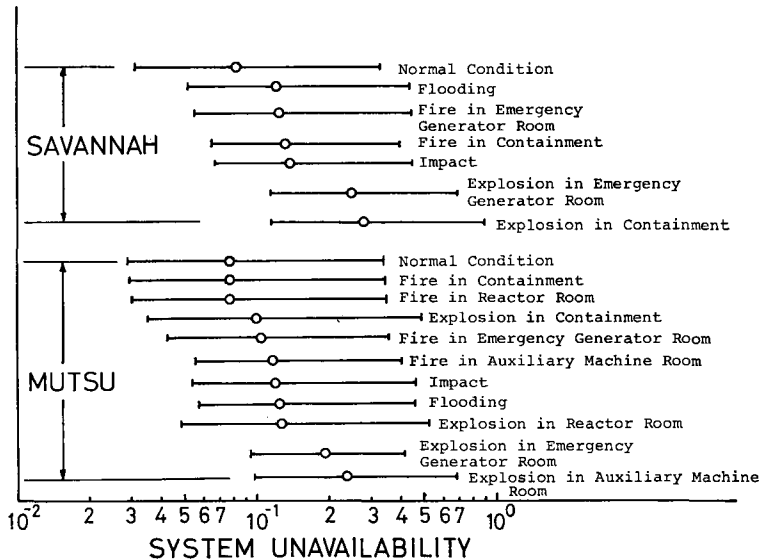


Fig. 5 System failure probabilities under various accident conditions

The largest value obtained among the results is that for the case of an explosion in the containment of Savannah. On the other hand, the top event probability for the same case of Mutsu is small by a factor of 3 as compared with Savannah. The reason of this difference is that, in the case of Savannah, almost all the components of the system are located inside the containment, but in the case of Mutsu these components are separately located in the three places; the containment, the reactor room and the auxiliary machine room.

The probabilities of top event for Mutsu under the conditions "explosion in reactor room" and "explosion in auxiliary machine room", are also smaller than that for Savannah under the condition "explosion in containment". From these results it can be said that the dispersive arrangement is an important factor for the safe cool down of the reactor under the accident in one room.

The effect of the dispersive arrangement is also seen in the result for "fire" accident. The probabilities of top event for Mutsu under the conditions "fire in containment", "fire in reactor room" and "fire in auxiliary machine room" are smaller than that for Savannah under the condition "fire in containment".

In the auxiliary machine room of Mutsu, there are a pump, a switch, a breaker and secondary pipes which failure probabilities are sensitive to accident conditions and large under the condition "explosion". In the case of Mutsu, these failure probabilities prominently contribute to the top event probability under the condition "explosion in auxiliary machine room". Therefore, the dispersive arrangement is not effective and the probability of top event is close to that for Savannah under the condition "explosion in containment". The same thing is seen for the accident "fire in the auxiliary machine room of Mutsu".

In the emergency generator room of Mutsu, there are five sensitive components (emergency generator, two switches, breaker and electric circuit) which have large failure probabilities under the condition "explosion". On the other hand, there are nine sensitive components (emergency generator, switch, six breakers, electric circuit) in the emergency generator room of Savannah. The difference between the number of these sensitive components produces the difference of the top event probabilities between Mutsu and Savannah, in the case of "explosion in emergency generator room". The same thing is also seen for the accident "fire in emergency generator room".

In the case of "impact", all the components are affected by the impact force. Therefore, the dispersive arrangement cannot change the system reliability. The probabilities of top event under the condition "impact" are close each other for Mutsu and Savannah.

Under the condition "flooding", Savannah has the smaller probability of top event than Mutsu. This is due to the fact that, for Savannah, most components are inside the containment and this system is well protected from this flooding accident.

From the above results, it is seen that the reliability of these systems is influenced by extreme environmental conditions. The operation of safety system of a nuclear ship is considered to be mainly required when some accident occurs. Therefore, for the evaluation of the effectiveness of safety system of a nuclear ship, it is necessary to calculate its failure probabilities not only under the normal condition but also under accident conditions.

V. CONCLUSIONS

The reliability of the emergency decay heat removal system of the nuclear ship "Mutsu" and that of the emergency sea water cooling system of the nuclear ship "Savannah" are almost the same under the normal condition.

The dispersive arrangement is useful to prevent the reduction of the system reliability under the condition of an accident in one room.

The system reliability is mainly determined by the reliability of the components which have large failure probabilities.

From the comparison between the reliability of these two systems under many accident conditions, it is seen that the reliability of a system is determined by the configuration and the environmental condition of a system as well as the maintenance interval, the test interval, the repair time *etc.*

For the evaluation of the effectiveness of safety system of a nuclear ship, it is necessary to evaluate its reliability under various accident conditions.

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