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Printed June 1983

Study of the Value and Impact of Alternative Decay Heat Removal Concepts for Light Water Reactors

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Prepared by
Sandia National Laboratories
Albuquerque, New Mexico 87185 and Livermore, California 94550
for the United States Department of Energy
under Contract DE-AC04-76DP00789

Prepared for
U. S. NUCLEAR REGULATORY COMMISSION

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GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

and

National Technical Information Service
Springfield, Virginia 22161

NUREG/CR-2883/1 of 3
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STUDY OF THE VALUE AND IMPACT OF ALTERNATIVE DECAY HEAT REMOVAL CONCEPTS FOR LIGHT WATER REACTORS

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operated by
Sandia Corporation
for the
U. S. Department of Energy

Prepared for
Division of Risk Analysis
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, DC 20555
Under Memorandum of Understanding DOE 40-550-75
NRC FIN No. A-1226

ABSTRACT

Reliability assessments were made of systems used to remove decay heat from pressurized and boiling water reactors. Current design practices in both U. S. and foreign plants were reviewed in order to identify the types of systems commonly used for decay heat removal and to determine the regulatory criteria that control decay heat removal system design in various countries. Typical existing decay heat removal system designs were identified for a number of different plant configurations, and the reliability of each of these systems was assessed. Alternative decay heat removal systems were postulated and improvements in decay heat removal reliability were assessed for various combinations of plant configurations and alternative decay heat removal concepts. Alternative concepts that could be implemented by retrofitting existing plants and ones that would be feasible only for new construction were considered. Cost estimates were made for those alternatives that provided significant improvements in decay heat removal reliability.

Based on an evaluation of nine alternative decay heat removal concepts, it was found that the best alternatives were those which apply proven, small components with few interfacing systems and few containment tie-ins. Closed loop concepts which rely on sensible heating of water were found to lack retrofit feasibility because of the need for large components, piping, and containment penetrations.

For add-on decay heat removal concepts which were engineered to be completely independent of existing decay heat removal systems, a pressurized water reactor auxiliary feedwater train, a pressurized water reactor high pressure injection train, and a boiling water reactor suppression pool cooling and low pressure injection train were found more feasible for retrofit than six other candidate concepts. The engineering, design, interest, and construction cost of these three concepts range from $20 to $30 million, and their estimated plant outage time for retrofit ranges from two to fourteen weeks, depending on the extent of add-on system interface tie-ins to be made inside the containment building.

It was found that for some power plants, typically those with three or more separate trains of decay heat removal capability, further improvements in decay heat removal reliability may not result in significant reduction in core melt probability. In pressurized water reactors having two installed decay heat removal trains, estimated reductions in core melt probability of at least a factor of ten were attained with the addition of an auxiliary feedwater or high pressure injection train. For boiling water reactors, an add-on single train suppression pool cooling/low pressure injection system gains a factor of about six reduction in core melt probability.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 27, 1983

FOREWORD

The report which follows is the first of three to be published in the first half of 1983 which deal with the costs and benefits of specific severe accident prevention and mitigation features. In this report, an accident prevention concept, an alternative decay heat removal (ADHR) system, is investigated. The remaining two reports relate to filtered-vent containment systems and their costs and benefits when hypothetically installed on BWR Mark I and Mark III design types. All of these studies were begun in 1979 as part of NRC's "Improved Reactor Safety" Program, and were intended to be highly detailed analyses of those specific concepts identified as most promising in NRC's 1978 report to Congress on improved safety (NUREG-0438).

In this report, several types of ADHR systems were assumed to be installed in certain PWR and BWR designs. For the systems which appeared most feasible, quantitative analyses were made of their cost and their benefit, that is, their potential for reducing the frequency of severe accidents. In addition, a qualitative evaluation was made of the potential benefit of these systems in coping with "special emergencies," such as earthquakes, fires, etc. In order to perform the quantitative benefit analysis, certain plant-specific PRAs were used. Sensitivity studies were then performed to provide some indication of the benefit of ADHR systems for designs other than those specifically associated with the PRAs.

This report is related to two ongoing NRC regulatory activities. First, this report provides an important contribution to the resolution of one aspect of NRC's Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements." In the USI, detailed analyses are being performed to assess the adequacy of decay heat removal reliability in achieving both hot and cold shutdown conditions. One part of this is the evaluation of alternative means for improving DHR system reliability. The work described in this report provides a foundation for these additional, more detailed analyses.

This ADHR report and the other Improved Reactor Safety Program reports are also clearly related to the NRC's ongoing evaluation of the need to explicitly consider severe accidents in its regulatory process (c.f. NRC's Proposed Policy Statement on Severe Accidents, SECY-82-1B, 48 FR 16014). More specifically, these studies provide important input to the integrating technical program supporting NRC's severe accident deliberations, the Severe Accident Risk Reduction Program. During the remainder of 1983, this program will be using such studies along with reevaluations of accident likelihoods, accident phenomenology experimental results, and improved analytical models (e.g., for "source term" calculations) to provide state-of-technology estimates of the level of risk associated with severe accidents and the cost-effectiveness of means for reducing this risk. Thus, the results of this ADHR report will be reevaluated using more recent accident likelihood information and made more applicable to a broader set of LWR design types, in close coordination with the USI A-45 work described above. (Since the benefit measure used in this report is severe accident likelihood reduction, its results are not influenced by accident phenomenological issues such as the "source term.") In support of this, we solicit any comments on this work which readers feel are appropriate.

We believe it important that the reader of this ADHR report keep clearly in mind the context of these results as one input to the ongoing evaluations of risk and risk-reduction in the Severe Accident Risk Reduction Program. In this program, the ADHR report results will be updated and its merit weighed relative to a spectrum of other severe accident prevention and mitigation schemes. Thus the results of this ADHR report will not be used, as such, in the NRC's severe accident deliberations. Only upon its updating and placement in proper context will this work be used in support of these deliberations.

DFRoss

D. F. Ross, Deputy Director
Office of Nuclear Regulatory Research

ACKNOWLEDGMENT

The authors thank Burns and Roe, Inc., for their support in providing the technical information and design diagrams used in the impact assessment of the alternative decay heat removal systems proposed in this study. A significant portion of their information was used directly to prepare the impact evaluations presented in Chapter 4. In addition, Appendices A, B, and C are exact duplications of the detailed system analyses performed by Burns and Roe in applying the most promising alternative decay heat removal concepts to a range of actual nuclear power plants.

Study of the Value and Impact of Alternative Decay Heat Removal Concepts for Light Water Reactors

1.0 Introduction

As part of the U.S. Nuclear Regulatory Commission (NRC) light water reactor (LWR) safety program, Sandia National Laboratories undertook a study to assess the value and impact of alternative decay heat removal concepts for LWR nuclear power plants. In accordance with the NRC research plan issued to Congress in April 1978,[1] alternative concepts considered by Sandia have focused on improving decay heat removal system reliability by reducing system vulnerability to hazards which can challenge or jeopardize system operation. The Sandia program to assess alternative concepts was divided into the following tasks:

1. Identify events requiring or jeopardizing decay heat removal operations.

2. Evaluate the ability of current U.S. and foreign systems to cope with hazardous events.

3. Identify potential inadequacies of current designs.

4. Propose a group of design criteria for alternative concepts which appear to address current inadequacies.

5. Select several candidate alternative concepts for evaluation.

6. Develop the candidate concepts in sufficient detail to permit an assessment of their value and impact.

7. Formulate a technique for assessing the value and impact of the candidate concepts.

8. Perform a value and impact assessment of the candidate alternative concepts.

In an earlier report, Sandia presented the preliminary results of Tasks 1 through 5.[2] Those preliminary results, along with Tasks 6, 7, and 8, have received further study, and together they serve as the basis for this final report.

Section 2 of this final report summarizes the findings presented in Reference 2 for Tasks 1 through 5, including a description of several alternative decay heat removal concepts chosen for evaluation. Section 3 describes the results of a feasibility assessment of the candidate concepts to select those concepts most worthy of further evaluation. Section 4 presents the results of an impact assessment of the selected candidate concepts. Sections 5 and 6 describe the analysis technique and results used to assess the value of the selected candidate concepts. Finally, Section 7 summarizes the major conclusions reached in this study.

2.0 Current Systems and Proposed Options

This section summarizes the results of an investigation of U.S. and foreign decay heat removal systems, and it presents proposed design criteria and conceptual descriptions for several alternative decay heat removal system concepts. These results, criteria, and descriptions, which are discussed at length in Reference 2, served as the starting point for the more detailed analysis of alternative concepts presented in later sections of this report.

2.1 U.S. and Foreign Decay Heat Removal Systems

Under normal operating conditions, power generated within a reactor is removed as hot water or steam to produce electricity via a turbine generator. Following a reactor shutdown (SCRAM), however, a reactor produces insufficient power to operate the turbine. When this occurs, other measures must be available to remove decay heat from the reactor to ensure that high temperatures and pressures do not develop which could jeopardize the reactor or the reactor coolant system. Thus all light water reactors (LWRs) share the common decay heat removal functional requirements of (1) providing a means of transferring decay heat from the reactor coolant system to an ultimate heat sink and (2) maintaining sufficient water inventory inside the reactor to ensure that the reactor coolant adequately cools the reactor fuel.

Pressurized water reactors (PWRs) generally remove decay heat by using one of three systems: the turbine bypass system, the auxiliary feedwater (AFW) system, or the residual heat removal (RHR) system. To maintain reactor water inventory, PWRs use the chemical and volume control and the high pressure safety injection systems. Unlike PWRs, boiling water reactors (BWRs) combine the functions of removing decay heat and maintaining coolant inventory during a reactor shutdown. Most often, the BWR systems which do this are the turbine bypass system, the reactor core isolation cooling (RCIC) system, and the residual heat removal system. The RCIC system operates as a high pressure system, and it is backed up by a separate high pressure core spray system. The BWR RHR system can operate in a number of different modes for both low pressure injection and decay heat removal.

Reference 2 describes many of the design variations that have evolved in the U.S. and abroad for PWR and BWR decay heat removal systems to accommodate different equipment vendors, utility preferences, technological advances, engineering philosophies, and regulatory developments. In general, these variations involved the numbers of components or equipment trains employed to make up the decay heat removal systems. Very little evidence was found of more innovative design approaches. However, even though the same basic methods of decay heat removal have been adopted throughout the world, a number of examples of non-U.S. power plants revealed the use of more system redundancy and separation than is common in U.S. plants.

A philosophy has evolved in several European countries that requires safety systems to be able to sustain both a random failure of one safety train and a simultaneous maintenance outage of another train and still retain 100 percent operational capacity. This philosophy, known as "N+2 redundancy," has led to the use of three 100 percent trains or four 50 percent trains of decay heat removal equipment, and it goes beyond the U.S. practice of meeting single failure, N+1 redundancy. However, a review of power plants in Switzerland, West Germany, Belgium, Sweden, Italy, France, and Great Britain revealed that the "N+2 redundancy" philosophy has never been adopted with the specific objectives of improving decay heat removal reliability or reducing the probability of a core meltdown. Instead, a concern over less frequently occurring special emergencies (e.g., airplane crash and explosive pressure waves) has prompted the installation of extra trains of decay heat removal equipment. By doing this, some countries have emphasized systems whose failure is of little consequence to preventing core meltdown, while at other times they have improved the reliability of important decay heat removal systems well beyond that needed to reduce core meltdown probability and even beyond the N+2 criteria.[2] The basis for this statement lies in a comparison of those decay heat removal systems found to be important in the Reactor Safety Study[3] with those systems which have received the greatest European emphasis.

As indicated in Reference 2, the following observations can be made for the Surry PWR analyzed in the Reactor Safety Study:

1. Transients and certain small loss of coolant accidents (LOCAs) pose the highest probability for core meltdown.

2. Of all transients and small LOCAs, those involving the high pressure injection and auxiliary feedwater systems pose the highest probability for core meltdown.

3. For the three-train Surry high pressure system and auxiliary feedwater systems, system reliability improvements of as much as a factor of ten will result in less than a factor of two decrease in overall core meltdown probability.

Similarly, the following observations can be made for the Peach Bottom BWR analyzed in the Reactor Safety Study:

1. Transients, together with failure of the residual heat removal system, or the reactor protection system, pose the highest probability for core meltdown.

2. For the two-train Peach Bottom residual heat removal system, improvements of as much as a factor of ten in the reliability of either the residual heat removal and high pressure service water systems or the reactor protection system will result in only about a factor of two decrease in overall core meltdown probability.

In contrast to these observations regarding decay heat removal systems found to be important to safety, other observations were reported in Reference 2 concerning five PWR and five BWR European plants where decay heat removal systems have been designed or upgraded for special emergencies or N+2 redundancy.

PWR Observations:

- Much effort has been placed on providing three, four, and even six trains of residual heat removal system cooling, even though no dominant core meltdown scenarios have been attributed to the typical two-train U.S. system.

- None of the special emergency high pressure injection systems provided on the five non-U.S. PWRs reviewed address small break LOCAs. Instead, the design basis of these systems is to provide water makeup and location to handle leakage and shrinkage which can occur during decay heat removal operations.

- Three- and four-train auxiliary feedwater systems are backed up in some countries by two- and three-train special emergency feedwater systems, even though further improvements in safety rapidly diminish for systems using more than three independent auxiliary feedwater trains.

BWR Observations:

- Some BWRs have provided as many as three, four, or six separate trains of service water and three or four trains of residual heat removal cooling versus the typical two-train U.S. practice, even though the safety benefit of providing greater than three trains appears to lack a technical basis.

From these observations it seems that the steps taken in some countries to improve decay heat removal systems have emphasized systems whose failure is of little consequence to preventing core meltdown and other times have improved the reliability of important systems well beyond that needed to reduce core meltdown probability. Although this conclusion strictly applies to plants which are somewhat similar to Surry and Peach Bottom, the findings of safety analyses for other power plants support the conclusions reached for Surry and Peach Bottom regarding which decay heat removal systems are most important to safety and how safety benefits decrease as the number of safety trains increases beyond three.[4-7]

2.2 Current U.S. and Foreign Design Criteria

Within the U.S., design criteria for decay heat removal systems range from the general guidance presented in Reference 8 and various NRC Regulatory Guides to more specific requirements found in NRC Standard Review Plans and Branch Technical Positions. In addition, more detailed system information, such as flow rates and decay heat levels, may be found in power plant safety analysis reports; however, much of this information presents a design description, rather than criteria for a design. Outside the U.S., documents stating decay heat removal system design criteria often are inaccessible to the public, and many regulatory decisions are made on a case-by-case basis. Therefore, information about non-U.S. design criteria was gathered with the assistance of a foreign nuclear power plant design organization who is familiar with design practices outside the U.S.

Table 1 summarizes the design criteria that have evolved in the U.S. and abroad for decay heat removal systems. This summary is based on more detailed criteria descriptions presented in Reference 2, and it divides the criteria into two categories:

I. Criteria specified worldwide
II. Additional criteria specified in non-U.S. countries

A review of these criteria reveals that conflicts and inconsistencies exist because of the separate evolution of criteria in different countries. Some examples of these differences include the following:

1. Criteria issued by the International Atomic Energy Agency and by some foreign countries require that decay heat removal be accomplished automatically, whereas U.S. criteria only suggest automatic operation (Criterion 1, Category II).

2. The time period specified abroad for automatic operation varies from 10 minutes to 4 hours to 10 hours without a documented design basis (Criterion 1, Category II).

3. Criteria issued by the International Atomic Energy Agency and by some foreign countries require decay heat removal despite a component failure in combination with a maintenance outage, whereas U.S. criteria require sustaining only a single active failure (Criterion 2, Category II).

4. Component failures assumed to occur vary among countries from only active components to either active or passive components (Criterion 6, Catgegory I; Criterion 2, Category II).

Table 1. Summary of Existing Design Criteria for Decay Heat Removal Systems

	United States	Switzerland	Germany	Belgium	Italy	European Export*	France	Sweden
Category I: Criteria Specified Worldwide								
1. Ensure that fuel integrity and pressure boundaries are maintained	X	X	X	X	X	X	X	X
2. Withstand fire, sabotage, natural phenomena, and other extreme conditions	X	X	X	X	X	X	X	X
3. Operate under normal and emergency power conditions	X	X	X	X	X	X	X	X
4. Provide manual backup control capability for automatic systems	X	X	X	X	X	X	X	X
5. Monitor and maintain reactor coolant pressure boundary through inspection, leak detection, and isolation valving	X	X	X	X	X	X	X	X
6. Function despite the single failure of an active component or the occurrence of small pipe breaks	X	X	X	X	X	X	X	X
7. Prevent shared normal or emergency equipment from jeopardizing reliable safety operations	X	X	X	X	X	X	X	X
8. Provide uninterrupted cooling for thirty days	X	X	X	X	X	X	X	X
Category II: Additional Criteria Specified in Non-U.S. Countries								
1. Initiate and operate automatically for a period ranging from 10 minutes to 10 hours		X	X	X	X			
2. Function despite the single failure of an active or passive component in combination with a maintenance outage involving a redundant system		X		X	X			
3. Prevent, through the use of bunkers or separation, extreme emergency conditions, including explosions, from jeopardizing reliable safety operations				X				
4. Provide an independent scram system as a backup to the normal scram system				X				
5. Prevent the use of shared equipment between redundant safety trains		X						
6. Design system operation for less than a 10^{-7} per year probability of two hour outage						X		
7. Permit normal, emergency, and backup emergency systems to serve complimentary safety functions						X		
8. Cooldown to 100°C within six hours								X
9. Operate without on-site repair action for at least 12 hours and without offsite repair action for at least 48 hours		X	X	X	X	X	X	X
10. Accomplish manual actions only if greater than thirty minutes duration and written procedures are available		X						

*"European Export" refers to a power plant design being sold to other countries by a European country. proprietary reasons, the specific countries involved can not be identified.

5. Some countries permit the use of shared equipment between redundant safety trains, whereas other countries prohibit the sharing of equipment (Criterion 7, Category I; Criterion 5, Category II).

6. One country establishes a numerical reliability goal for decay heat removal systems, while other countries do not (Criterion 6, Category II).

7. One foreign country requires a six-hour cooldown capability, while other countries do not (Criterion 8, Category II).

8. Some foreign countries specify the conditions under which repairs or manual operations may be taken to support decay heat removal (Criteria 9 and 10, Category II).

9. One foreign country permits emergency and backup systems to serve complimentary safety functions without being redundant to each other or to normal systems (Criterion 7, Category II).

10. One foreign country specifically endorses the use of independent SCRAM circuits and bunkers or separation to protect against severe special emergencies, while other countries do not (Criterion 3, Category II).

Two other observations can be made from Table 1 that are worth noting. First, none of the countries reviewed have invoked criteria for sustaining a loss of coolant accident while operating an alternative decay heat removal system. Second, except for Criterion 6 in Category II of Table 1, no requirement exists for specifying decay heat removal system reliability in terms of probabilities. The absence of these criteria may be significant. A loss of coolant situation during the Three Mile Island accident hampered decay heat removal efforts,[9] and an evaluation subsequent to the Three Mile Island accident of other power plants demonstrated the importance of probabilistically comparing the reliability of decay heat removal systems.[10]

2.3 Design Criteria for Choosing Candidate Alternative Concepts

In order to choose several candidate alternative decay heat removal systems, design criteria were selected from those criteria currently used or advocated for use throughout the world. By doing this, candidate systems could be selected to meet the criteria with reasonable assurance that a more detailed analysis would prove the candidates to be of value to plant safety.

It was thought, in developing criteria, that a distinction should be made between two different types of initiating events which can require or jeopardize decay heat removal operations.

Some initiating events occurring at nuclear power plants have been seen to occur frequently (e.g., partial loss of electrical power or loss of feedwater),[11-14] while other events, classified as special emergencies, have occurred less frequently or not at all (e.g., seismic disturbances or airplane crash). For those events which have occurred frequently, the probability that a system operates can be estimated from experience in many cases, while for events which have occurred infrequently or not at all, a lack of experience hampers efforts to predict the probability that a system operates.

The distinction between those events which have occurred and those events which have only been postulated implies that design criteria for both existing and alternative decay heat removal systems should be grouped into two categories. The first category should address initiating events and system failures which have been observed to occur and for which industry data allows a quantitative assessment of reliability improvements. The second category should address special emergencies which have been postulated as potential threats to decay heat removal operations but for which a lack of experience permits only a qualitative assessment of reliability improvements. Examples of events in the first category include loss of feedwater, loss of offsite power, loss of onsite electrical power, human errors, random failures, test and maintenance outages, and common-mode failures. While for the second category, examples include hurricane, flood, sabotage, airplane crash, fire, external explosion, and earthquake. This implies the need for two different sets of criteria--one for improving decay heat removal reliability for events which can be assessed probabilistically and one for ensuring that special emergency conditions are met.

2.3.1 Criteria for Probabilistically Evaluated Events

For both PWRs and BWRs, operating experience and reliability estimates indicate that an overwhelming majority of transients which interrupt normal heat removal via the power conversion system and which require the operation of decay heat removal systems can be classified into one of two groups as shown below:[3,9,11-15]

Transients	Approximate Rate of Occurrence
Loss of main feedwater	3 per reactor year
Loss of main feedwater in conjunction with a loss of offsite power	0.2 per reactor year

In addition to these transients, attention recently has been given to a postulated loss of main feedwater in conjunction with a loss of both offsite and onsite alternating current (AC) power, because operating experience has revealed precursors to this scenario (i.e., momentary loss of all AC power).[11,12] The performance of PWR auxiliary feedwater and high pressure injection systems, which may be required soon after reactor shutdown, can be severely degraded by a loss of all AC power. Also, an extended loss of AC power can degrade DC instrumentation and control systems which are necessary to operator systems having no direct AC dependency (e.g., BWR reactor core isolation cooling systems).

For purposes of choosing candidate alternative decay heat removal concepts for subsequent evaluation, probabilistically oriented design criteria intended to cope with the above transients were selected from criteria being used in the U.S. and abroad and from the findings of a number of reliability studies.[3,5,11,12,16,17] In selecting the criteria, it was assumed that alternative decay heat removal concepts would most likely prove beneficial to those power plants whose existing systems cannot be dramatically improved through simple design or operational changes. This assumption was based on experience that existing systems which are already well designed and operated cannot be easily upgraded, whereas an additional single-train system can be designed with relative ease to have an availability of at least 9 successful starts for every 10 operational demands. The value analysis presented in Section 4.0 supports this contention.

On the basis of the previous discussion, the following design criteria for probabilistically evaluated events were established to choose candidate alternative decay heat removal concepts of the one-train variety for PWRs and BWRs.

1. Alternatives shall be able to function without both offsite and existing onsite electricity for power and control; alternative electrical sources should be self-contained.

2. Alternatives shall be independent and separate from all existing plant systems whose functions they duplicate, except that the capability may be provided to manually crossconnect the alternatives to the existing onsite emergency electrical system as a backup.

3. Alternatives shall be designed to prevent exposure to pressures and temperatures which exceed their design limitations.

4. Alternatives shall be designed to permit inspection and testing on a periodic basis under conditions as close to design requirements as practical.

5. Alternatives and their supporting subsystems shall be designed to the same criteria and shall be designed to not interfere with or jeopardize other safety systems during normal or abnormal conditions.

6. Alternatives need not be designed to perform their functions during loss of coolant accidents which are sufficiently large to ensure adequate decay heat removal via operation of the emergency core cooling system.

7. Alternatives shall initiate automatically if system operation is required within 30 minutes, however automatic actuation should not cause or exacerbate accident conditions.

8. Alternatives shall operate automaticaly for 10 hours once initiated. Manual termination and control should be possible to override system malfunctions or to reactivate other decay heat removal systems.

Primarily, these criteria ensure that candidate alternative concepts are independent of existing systems to minimize the possibility of common-mode failure. It should be recognized, however, that for Criteria 7 and 8 the selection of the 30-minute and 10-hour time requirements are somewhat arbitrary. The purpose of these two criteria is to minimize the chance that operator error or unavailability could cause an alternative decay heat removal system to fail at a period when insufficient time is available for taking recovery actions. The 30-minute criterion attempts to account for the fact that, during the initial stages of an accident which require the use of an alternative decay heat removal system, operators will be forced to make numerous high-stress decisions. Under those circumstances, the chance for operator error increases and automatic actuation is preferred.[9,18] The 10-hour criterion attempts to account for a concern that the conditions (e.g., special emergencies, normal system failures, or offsite power outages) which called on the alternative decay heat removal system to operate could continue, unremedied, for an extended period. Although a technical basis for the 10-hour objective appears to be lacking, a number of countries already have adopted the requirement with the justification that over the 10-hour period, the extent and significance of an accident condition can be assessed and equipment repair and recovery actions can be undertaken. In addition to this, it is estimated that after 10 hours decay heat levels are low enough to provide from approximately 1 to 4-1/2 additional hours to restore some form of decay heat removal, assuming that the reactor coolant system is full after the 10-hour period, and assuming that core damage does not occur until 2/3 of the reactor vessel inventory is lost.* Of course,

*These estimates were performed for a 3250 MW_{th} four loop Westinghouse PWR where 2/3 of the reactor inventory is lost before significant core temperature increases occur.[19] The 1-hour estimate was calculated for the scenario where the steam generators are not available as a heat sink after the 10-hour period. If the steam generators are considered full and available, the 4-1/2 hour estimate is reached.

one may argue that for a particular power plant the 30-minute and 10-hour criteria should be modified to other values. However, for purposes of assessing the value and impact of alternative decay heat removal concepts, the exact time periods for automatic actuation and operation were found to have little influence on the acceptability of a concept, provided that the alternative system startup time did not follow scram too closely and provided that automatic operating periods of greater than about 15 hours were not considered. Beyond these limits, high decay heat levels and long term operating requirements (e.g., heat sinks and diesel fuel) prove unfavorable to alternative concepts which are capable of meeting the 30-minute and 10-hour criteria. This point is discussed more thoroughly in Section 3.0.

2.3.2 Criteria for Special Emergency Events

Design criteria for special emergency situations take on a different meaning than those criteria which address probabilistically evaluated events. The occurrence frequency and significance of special emergencies often are not easily predicted, and as a result, arbitrary decisions often are made regarding which special emergencies are most important. In addition, the installation of an add-on independent system to increase reliability, such as discussed in the previous section, may not always be needed to cope with special emergencies. Various levels of protection against special emergencies can sometimes be provided for otherwise reliable existing systems without adding another train of equipment.

From the review of international design criteria presented in Reference 2, it was found that two major categories of special emergency conditions have been considered--those which are site specific and those which are common to all power plants. Finally, it was found that many criteria for special emergency conditions already are part of U.S. regulatory requirements. Table 2 illustrates these findings.

From Table 2 there appears to be an almost worldwide consensus about which special emergencies warrant attention, and in general, all regulatory authorities currently require new licensing applicants to demonstrate that each condition in Table 2 either is inapplicable to a particular reactor site or has been addressed by the plant design. However, because the list of special emergencies shown in Table 2 has evolved over the years, some older power plants cannot handle some emergency conditions.[15,20] Thus, there may be a need to upgrade the decay heat removal systems in these plants for certain special emergencies.

To better define special emergency design criteria for alternative system concepts, the following assumptions were made:

Table 2

Special Emergencies Considered Worldwide

Special Emergency	Current U.S. Regulatory Requirement	International Regulatory Requirement[a]	Site Specific	Non-Site Specific
1. Hurricane	X	X	X	
2. External Flood	X	X	X	
3. Tornado	X	X	X	
4. Tsunami[b]	X	X	X	
5. Seiches[c]	X	X	X	
6. Earthquake	X	X	X	
7. Fire	X	X		X
8. Internal Flood	X	X		X
9. Pipe Whip	X	X		X
10. Internal Missiles	X	X		X
11. Industrial Sabotage	X	X		X
12. Civil Disturbances	I	X		X
13. Ship Collisions	O	X	X	
14. Frazil[d]	I	X	X	
15. Ice Jams	I	X	X	
16. Dust Storms	I	X	X	
17. Drought	I	X	X	
18. Airplane Crash	O	X	X	
19. Lightning	I	X	X	
20. Airborne Insects	O	X	X	
21. Aquatic Organisms	O	X	X	
22. External Explosion	O	X	X	
23. Volcanic Eruption	O	X	X	

X - Indicates a definite requirement.
I - Indicates an implicit requirement.
O - Indicates a requirement lacking specific guidance.
a - "International Regulatory Requirement" refers to requirements that have been suggested or imposed by various countries outside the U.S. or by the International Atomic Energy Agency.
b - Large sea wave caused by movement of the ocean bottom.
c - Periodic oscillations in the surface level of a lake or land-locked sea.
d - High concentration of suspended ice particles in moving water.

1. It was assumed that the use of an alternative decay heat removal concept to cope with special emergencies will be limited to those power plants whose existing systems cannot easily or adequately be protected; relatively minor modifications to existing systems, such as the addition of fire barriers or intrusion alarms, will not be considered as alternative system concepts.

2. It was assumed that several of the most severe emergency conditions in Table 2 can be selected as a basis for establishing bounding design criteria, and that these conditions can be applied individually or in combination to each alternative concept.

3. It was assumed that existing regulatory guidelines adequately define the design requirements (e.g., flood depth or earthquake force) for each special emergency selected from Table 2.

On this basis, the following design criteria for special emergencies were established to choose candidate alternative decay heat removal concepts of the one-train variety for PWRs and BWRs.

1. Alternatives shall be able to withstand industrial sabotage as defined in 10 CFR 73.55, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Industrial Sabotage;" Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Sabotage;" ANSI N18.17-1973, "Industrial Security for Nuclear Power Plants."

2. Alternatives shall be able to withstand earthquake as defined in Regulatory Guide 1.29, "Seismic Design Classification."

3. Alternatives shall be located or protected so that simultaneous loss of existing systems and the alternatives cannot occur as a result of fire, missile, flood, sabotage, and pipe whip, as defined in applicable regulatory documents (see Reference 2 and #1 above).

4. Alternatives shall be able to withstand extreme pressure loading from hurricane, tornado, and external explosions as defined in applicable regulatory documents (see Reference 2).

5. Alternatives shall be located or protected so that simultaneous loss of existing systems and the alternatives cannot occur as a result of airplane crash or ship collision.

6. Alternatives need not be designed to withstand the simultaneous occurrence of more than one special emergency, unless more than one condition can credibly occur simultaneously or in sequence (e.g., aircraft crash followed by an explosion and fire).

The above criteria were applied in choosing several candidate alternative decay heat removal concepts for evaluation.

2.4 Candidate Alternative Concepts

For purposes of establishing appropriate equipment sizes and functional requirements for the candidates, a hypothetical 1000 MWe power plant was configured with the following characteristics:

Reactor Coolant Loops	2
Thermal Rating (MWt)	3400
Primary Coolant Volume PWR/BWR (Ft^3)	10,500/22,000
Operating Pressure PWR/BWR (PSIA)	2200/1020
Operating Temperature PWR/BWR (°F)	600/550
Steam Generator Pressure (PSIA)	1100
Steam Generator Temperature (°F)	560
Steam Generator Coolant Volume/Steam Generator (Ft^3)	2000
Decay Heat Load, Peak (MWt)	150
Decay Heat Load, Average for 10 Hr. (MWt)	40
Reactor Coolant Leak Rate (GPM)	20
Service Water Temperature (°F)	80
Storage Water Temperature (°F)	90

This section briefly describes the candidates and lists some of the potential advantages and disadvantages foreseen for each. Section 3 presents a more detailed engineering feasibility assessment of the candidates.

2.4.1 PWR Candidates

Six PWR candidate alternative decay heat removal concepts were considered. They were:

- an add-on high pressure injection (feed and bleed) train
- an add-on auxiliary feedwater train
- a closed loop auxiliary feedwater train
- a high pressure residual heat removal train
- a passive feedwater train
- a passive makeup or circulation pump

These concepts were selected from a group of foreign and domestic system options and from suggestions brought to Sandia's attention by two private U.S. firms. They reflect a range of alternatives from replication of a conventional existing system to provision of an innovative system that is functionally similar to an existing system.

2.4.1.1 Add-on High Pressure Injection (Feed and Bleed) Train (Figure 1)

This concept is a single-train, 100-percent capacity system, without redundancy or single failure capability. The system would be sized for small LOCAs and would remove decay heat by a process of direct injection of cool water (i.e., feed) into the reactor coolant system and removal of hot water and steam (i.e., bleed) from the reactor coolant system via pressurizer relief valves.

Potential Advantages:

1. May be backfittable to existing plants.

2. May use existing components and technology.

3. May be protected to various levels of special emergencies, as deemed necessary.

4. May be designed to a general format applicable to several plants.

5. May be used under both accident and induced LOCA conditions (i.e., feed and bleed operations) to directly remove heat from the reactor coolant system.

Potential Disadvantages:

1. Relies on active components.

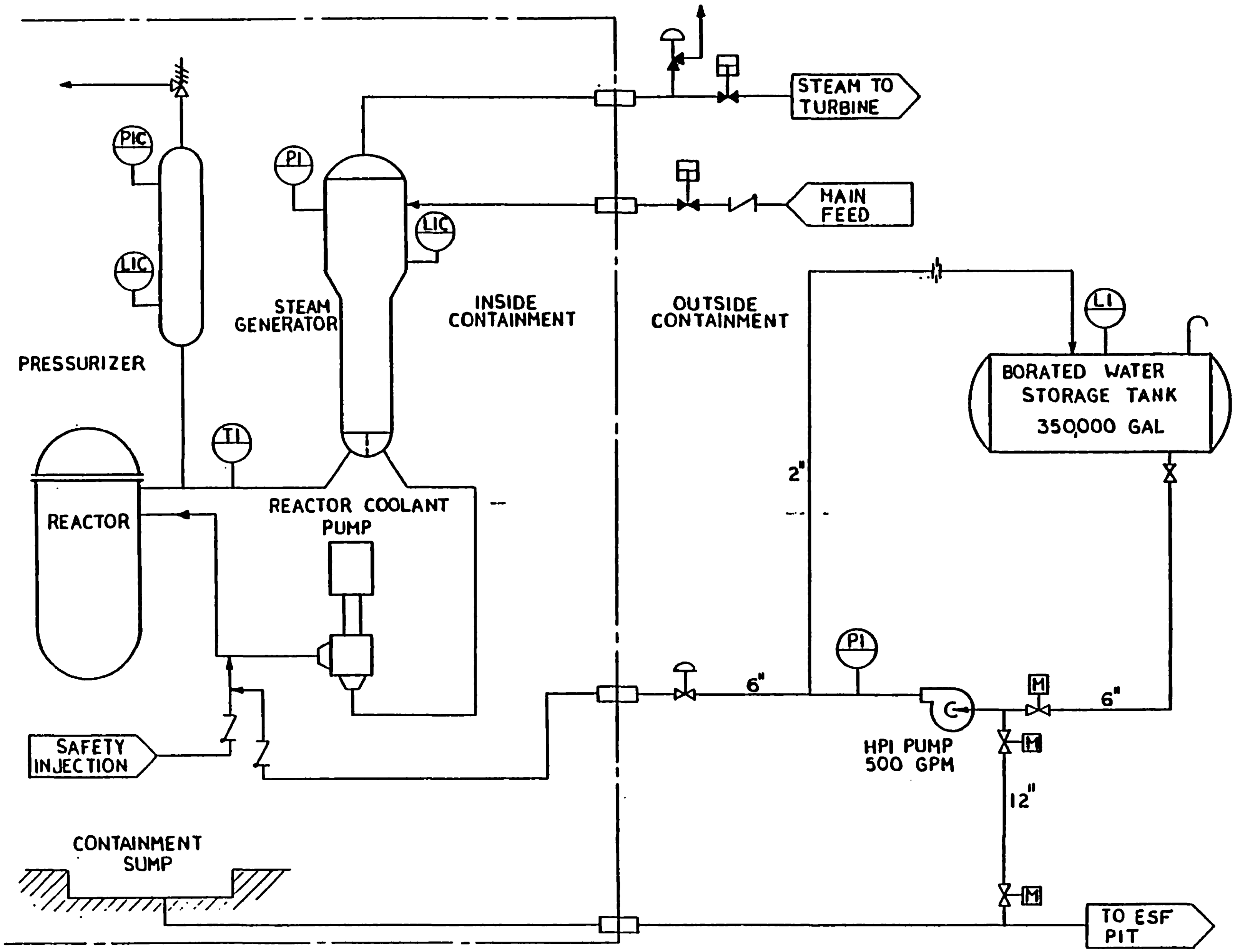
2. Cannot replace auxiliary feedwater decay heat removal function without inducing a LOCA.

3. Recirculation of water lost from the reactor coolant system depends upon the use of existing pumps or requires backfit of another suction system.

4. Long term operation dependent upon containment cooling systems and recirculation or disposal of water released from the reactor coolant system.

2.4.1.2 Add-on Auxiliary Feedwater Train (Figure 2)

This concept is a single-train, 100-percent capacity system without redundancy or single failure capability. The system would provide feedwater makeup to the steam generators for purposes of



NOTE: ONE PRIMARY LOOP SHOWN. PIPING CONNECTIONS TYPICAL FOR FOUR LOOPS.

Figure 1. Flow Diagram for an Add-on High Pressure Injector Train

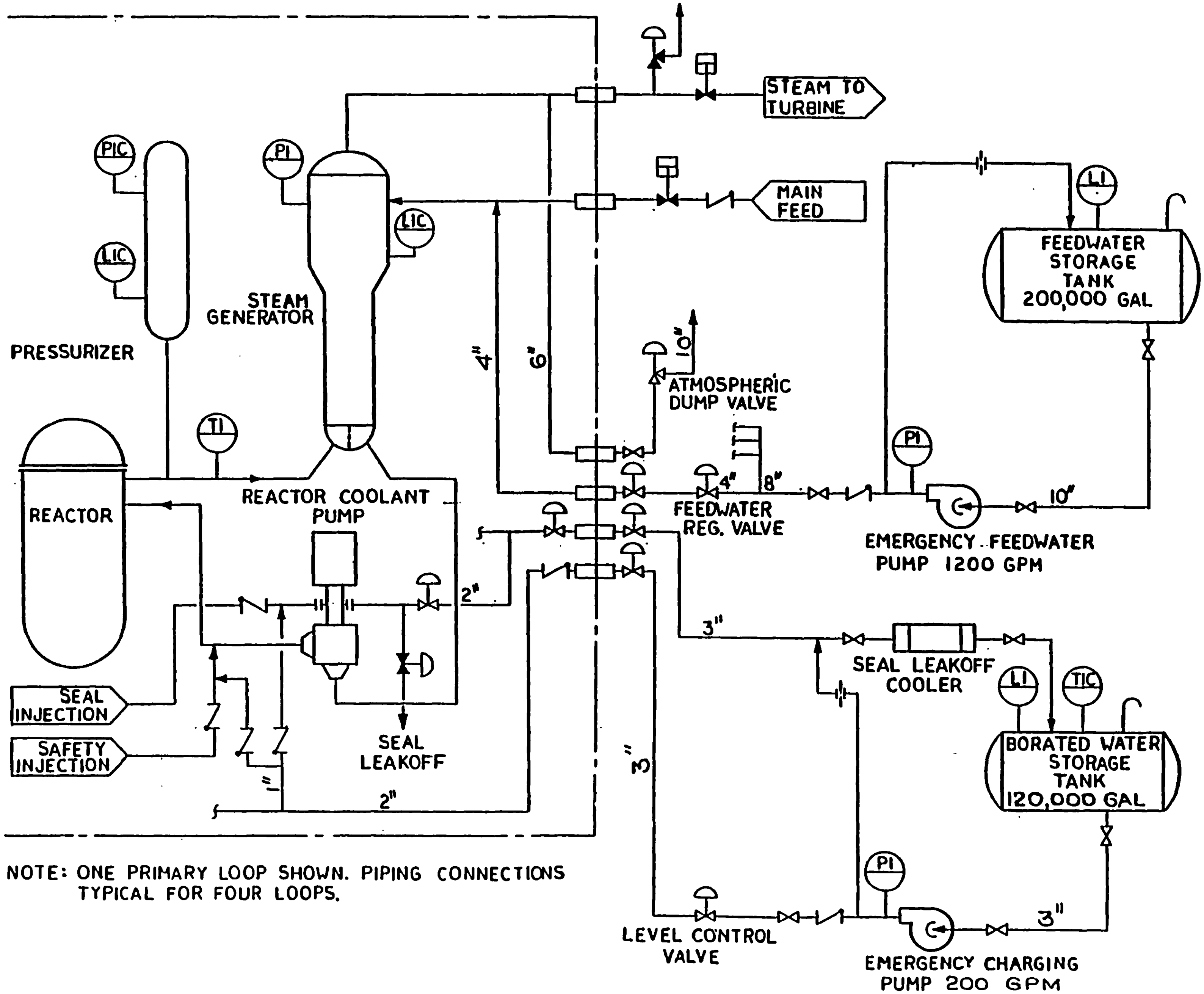


Figure 2. Flow Diagram for an Add-on Auxiliary Feedwater Train

removing decay heat and would provide high pressure borated water makeup to the reactor for small LOCAs, cooldown shrinkage, and "tech spec" leakage.

Potential Advantages:

1. May be backfittable to existing plants.

2. May use existing components and technology.

3. May be protected to various levels of special emergencies, as deemed necessary.

4. May be designed to a generic format applicable to several plants.

Potential Disadvantages:

1. Relies on active components.

2. Susceptible to steam generator failures.

2.4.1.3 Closed Loop Auxiliary Feedwater Train (Figure 3)

This concept is a single-train, 100-percent capacity system, without redundancy or single failure capability. Components in the system would be sized to handle decay heat loads which are present at the time steam generators boil dry following a reactor scram with no feedwater makeup. In practice, of course, the system would be brought on line prior to steam generator dryout to assure the availability of steam for operation. The closed loop auxiliary feedwater train would accept steam from the steam generators, condense it, and return it to the steam generators as feedwater. Heat removed while condensing the steam would be transferred to an ultimate heat sink by a service water system. The system may be visualized as comparable to the high pressure residual heat removal system shown in Figure 4, except that the connections would be to a steam generator instead of the reactor.

Potential Advantages:

1. May be backfittable to existing plants.

2. May use existing components and technology.

3. May be protected to various levels of special emergencies, as deemed necessary.

4. May be operated without a large makeup requirement.

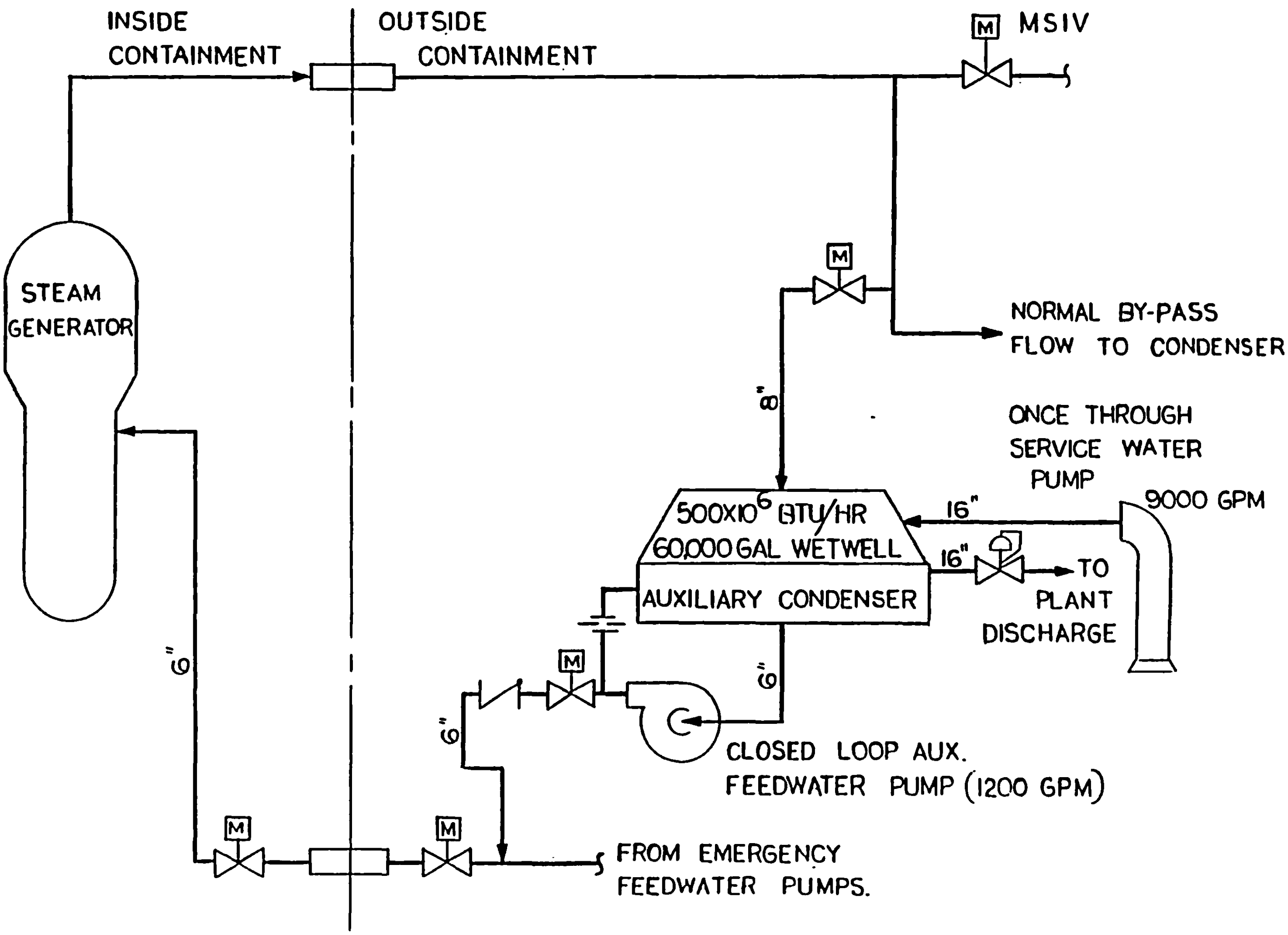


Figure 3. Flow Diagram for Closed Loop Auxiliary Feedwater Train

Potential Disadvantages:

1. Relies on active components.

2. May require a large condenser to handle initial decay heat loads.

3. Requires heat transfer to an ultimate heat sink by a service water system or an air heat exchanger.

4. Susceptible to steam generator failures.

5. Cannot handle small LOCAs.

6. Susceptible to loss of secondary inventory (e.g., stuck open relief valves).

2.4.1.4 High Pressure Residual Heat Removal Train (Figure 4)

This concept is a single-train, 100-percent capacity system, without redundancy or single failure capability. Components in the system would be sized to handle decay heat loads which are present about the time steam generators boil dry following a reactor scram with no feedwater makeup. Unlike currently existing residual heat removal systems, the high pressure system would be capable of withstanding pressures up to the design pressure of the reactor coolant system.

Potential Advantages:

1. May be backfittable to existing plants.

2. May use existing components and technology.

3. May be protected to various levels of special emergencies, as deemed necessary.

4. May be designed to a generic format applicable to several plants.

5. May be operated without a large makeup requirement.

6. May function despite steam generator tube failure, provided reactor coolant system depressurization can be accomplished independently using relief valves, pressurizer sprays or other similar means.

Potential Disadvantages:

1. Relies on active components.

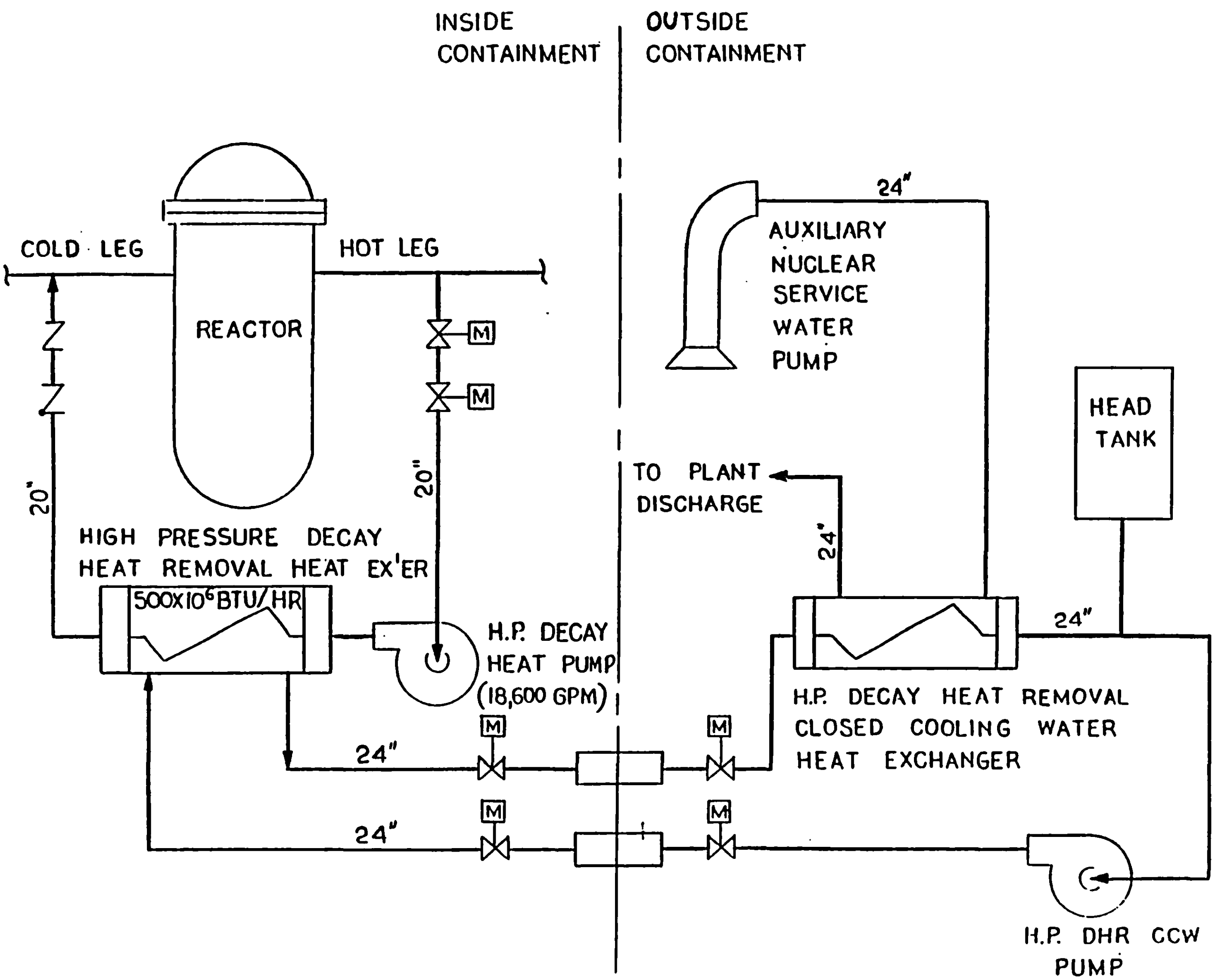


Figure 4. Flow Diagram for a High Pressure Residual Heat Removal Train

2. May require large components (i.e., heat exchangers) to handle initial decay heat loads.

3. May require special containment provisions if located outside the containment building (i.e., suitable protection for high pressure reactor coolant brought outside of containment).

4. Requires heat transfer to an ultimate heat sink by a service water system.

5. May be inoperable during small LOCA conditions.

2.5.1.5 Passive Feedwater Train (Figure 5)

This concept is a single-train, 100-percent capacity system, without redundancy or single failure capability. The system would use gravity feed from a tank located horizontally even with the steam generators. In this orientation, the tank may be located or shaped in whatever manner best suits the arrangements of a particular power plant, and therefore, the torus configuration shown in Figure 5 only represents an example. Decay heat would be dissipated by relieving steam from the steam generators.

<u>Potential Advantages</u>:

1. May be backfittable to existing plants.

2. May be operated initially without operation of an active makeup system.

3. May be operated without active components.

<u>Potential Disadvantages</u>:

1. Requires major structural or foundation support.

2. Susceptible to steam generator failures.

3. Cannot handle small LOCAs.

4. May cause uncontrolled overcooling.

2.4.1.6 Passive Makeup or Circulation Pump (Figure 6)

This concept would use the same flow paths as described for the closed loop high pressure residual heat removal train (Section 2.4.1.3) and the closed loop auxiliary feedwater train (Section 2.4.1.4), except that the pump in these trains would be modified to operate as a passive steam ejector pump. Since the only difference between this concept and those discussed earlier

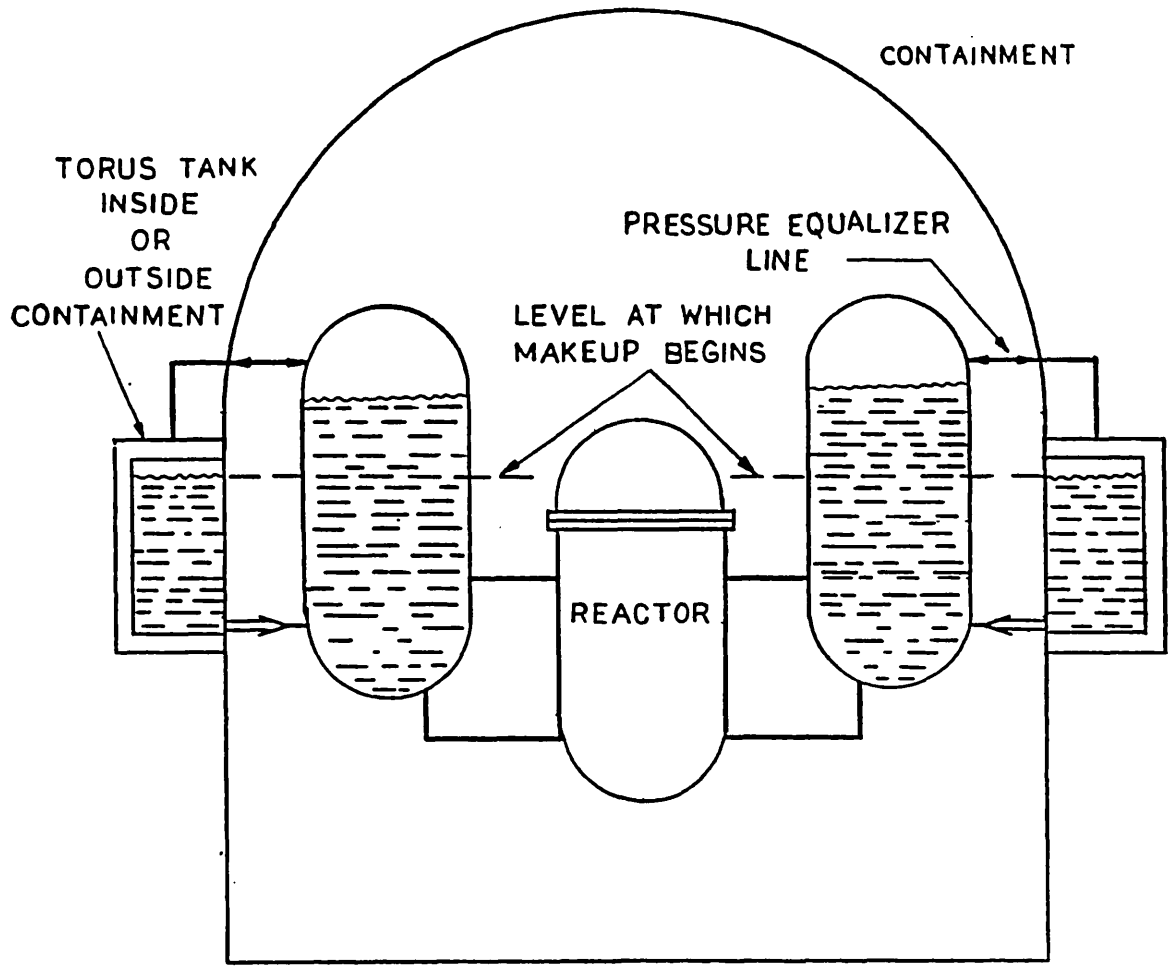


Figure 5. Flow Diagram for a Passive Feedwater Train

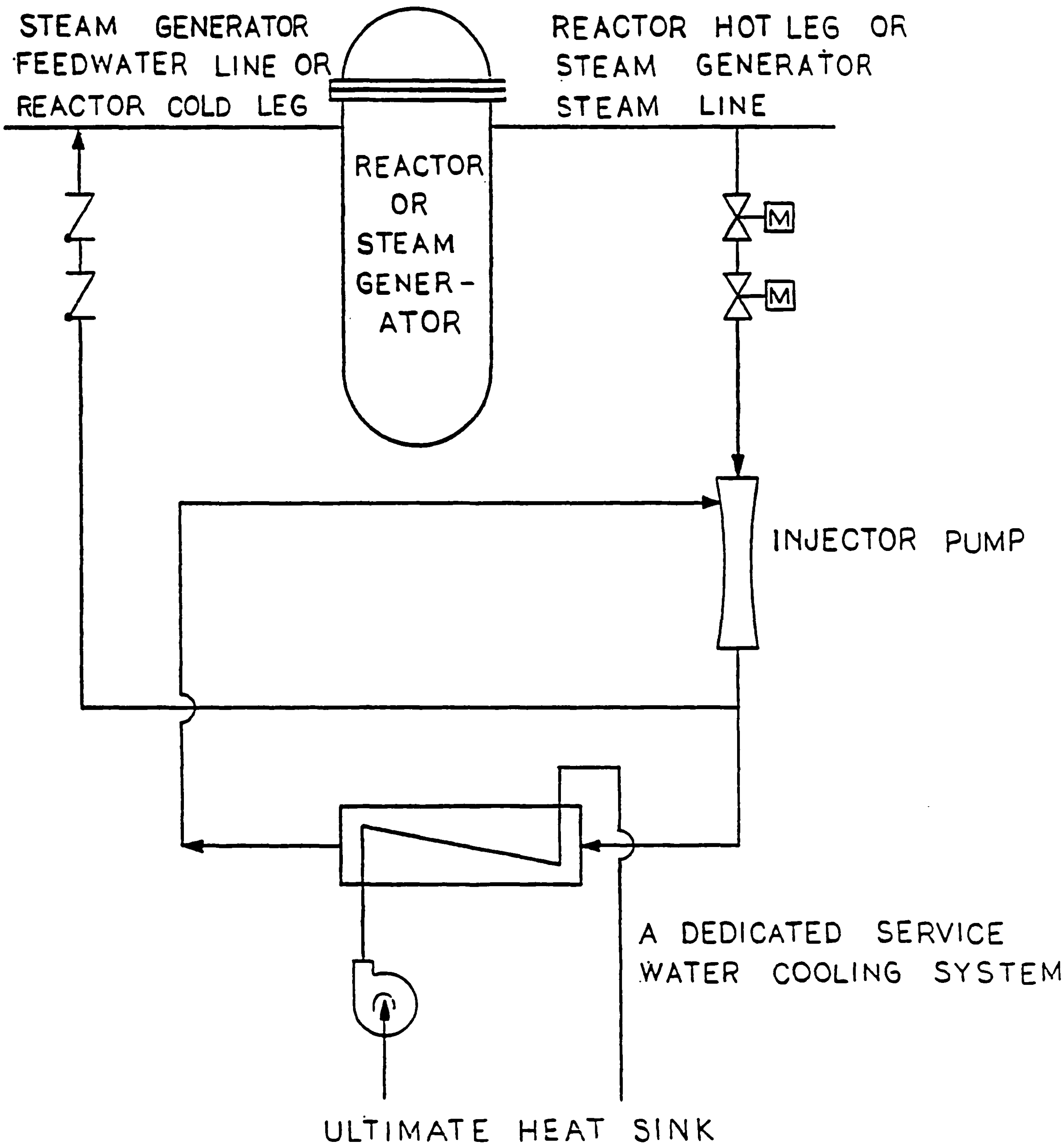


Figure 6. Passive Makeup or Circulation Pump

in Sections 2.4.1.3 and 2.4.1.4 lies in the use of a passive pumping scheme, most of the comments already made regarding the applicability and the potential advantages and disadvantages of the concept still apply. Two exceptions to this involve the potential added advantage of passive operation and the potential added disadvantage of using components and technology not previously used for this application in nuclear power plants.

2.4.2 BWR Candidates (Figures 7-9)

Three BWR candidate alternative decay heat removal concepts were considered. They were:

- a low pressure makeup and suppression pool cooling train

- a high pressure makeup and suppression pool cooling train

- a controlled/variable pressure makeup and suppression pool cooling train

These BWR concepts were selected from a group of foreign and domestic options. However, unlike PWRs for which there are options for removing decay heat via either the reactor coolant system or the steam generators, BWRs appear to have only a few logical options.

For the low pressure cooling option, the add-on system would rely upon the automatic depressurization relief valves or some add-on dedicated relief valves to depressurize the reactor vessel (Figure 7). At this point, the method of maintaining inventory would resemble the low pressure coolant injection mode of the residual heat removal system. For this option, the add-on pump would be modeled after the low pressure coolant injection pumps. For the high pressure cooling option, the add-on system would function as a reactor core isolation cooling (RCIC) or high pressure coolant injection (HPCI) system, using a motor-driven pump at flow rates and pressures comparable to the RCIC or HPCI pumps (Figure 8). The third option would involve the use of special depressurization valves to reduce pressure in a controlled fashion (Figure 9). This method would involve injecting water over a range of pressures between those of the RCIC and HPCI systems and the residual heat removal system. The add-on pump would be designed correspondingly.

The BWR candidate concepts are single-train, 100-percent capacity systems, without redundancy or single failure capability. They would include their own fluid systems (including high pressure service water to an ultimate heat sink), power supplies, control systems, and instrumentation (e.g., reactor vessel level and suppression pool temperature sensors). Valving to regulate the reactor coolant makeup and suppression pool cooling functions

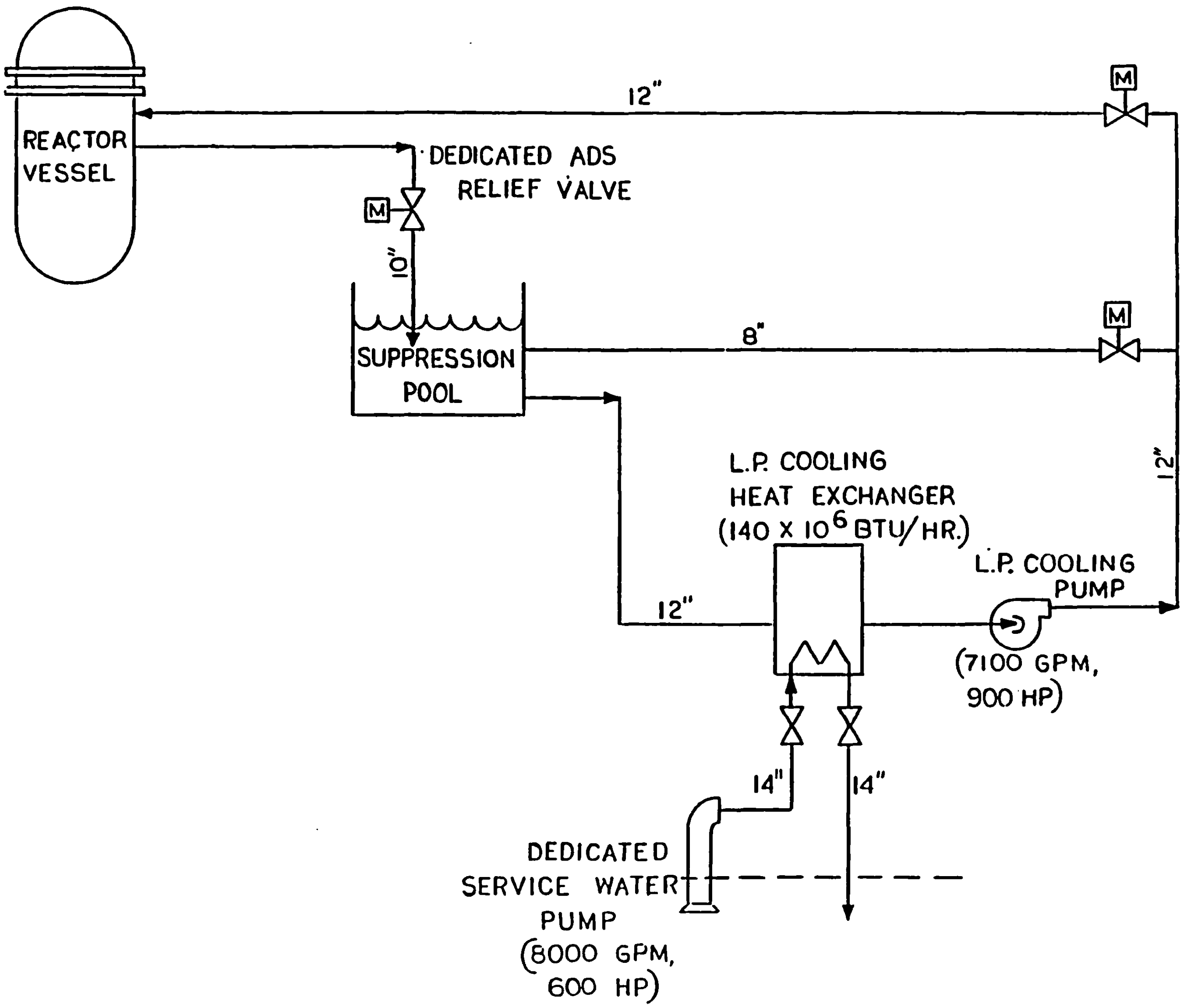


Figure 7. Flow Diagram for Low Pressure Makeup and Suppression Pool Cooling Train

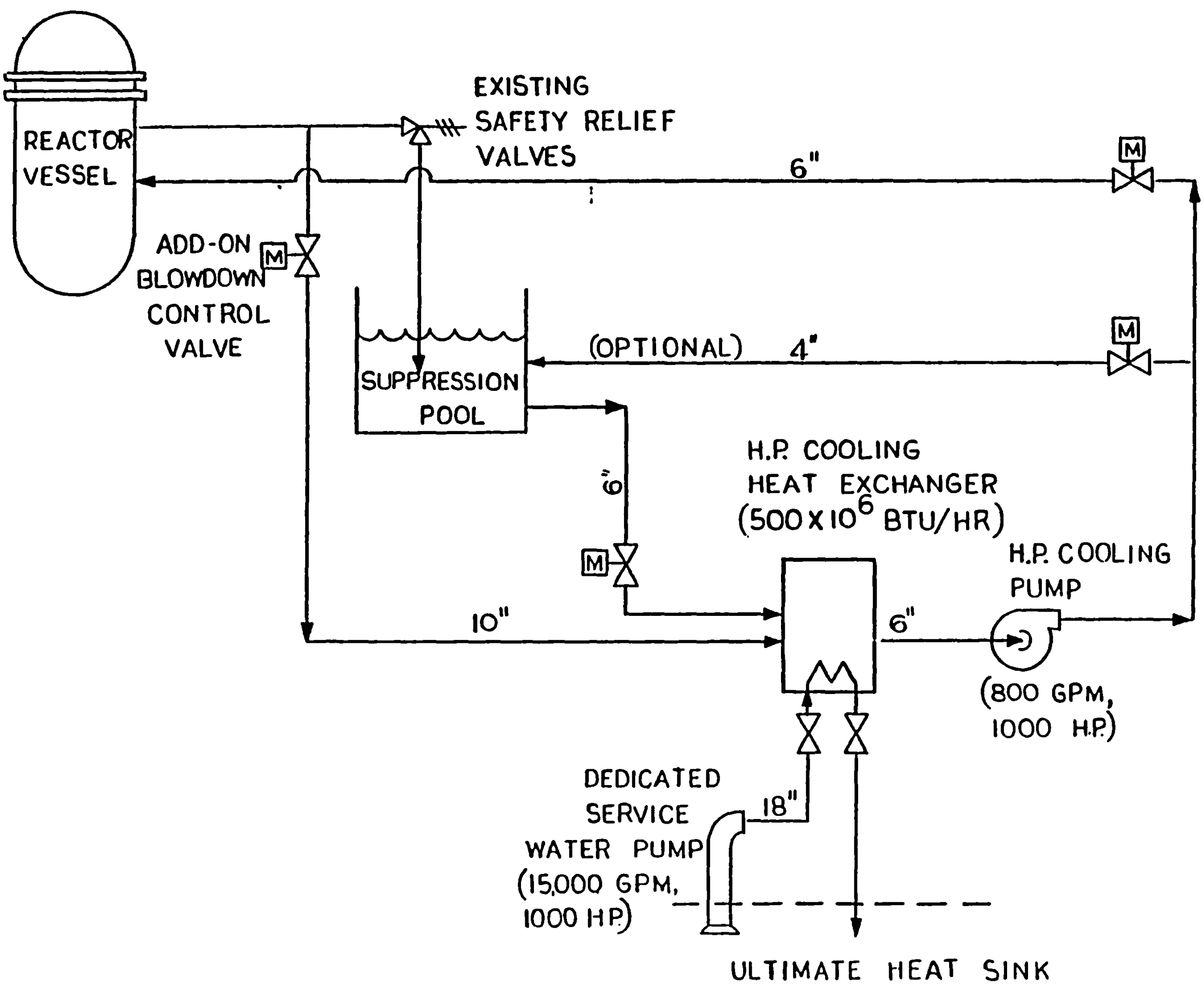


Figure 8. Flow Diagram for High Pressure Makeup and Suppression Pool Cooling Train

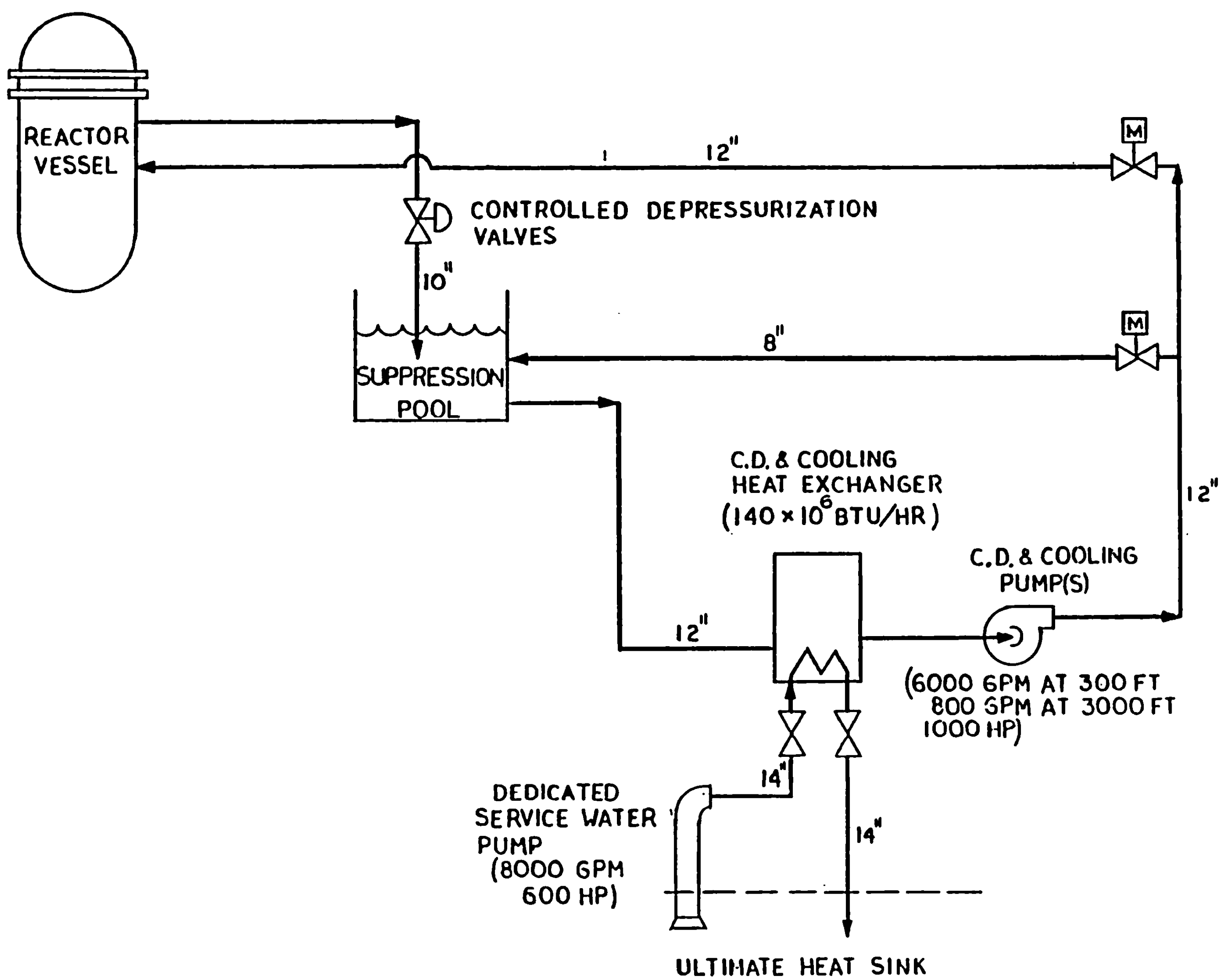


Figure 9. Flow Diagram for Controlled/Variable Pressure Makeup and Suppression Pool Cooling Train

of the system would be provided. Cooling water to the add-on heat exchanger, pump seals, motor bearing coolers, and room coolers would be provided by a dedicated service water system which connects to an ultimate heat sink.

Potential Advantages:

1. May be backfittable to existing plants.

2. May use existing components and technology.

3. May be protected to various levels of special emergencies, as deemed necessary.

4. May be designed to a generic format applicable to several plants.

Potential Disadvantages:

1. Relies on active components.

2. Requires heat transfer to an ultimate heat sink by a service water system.

In addition to these observations, it should be noted that the passive ejector pump discussed in Section 2.4.1.6 for PWRs may be able to take the place of the electric pump suggested for BWRs. Of course, this would only be true during the high pressure operating modes, when reactor coolant temperatures are high enough to operate an ejector.

3.0 Feasibility Assessment of the Candidate Concepts

Before performing a detailed study of the value and impact of any alternative decay heat removal concept, it was decided that the six PWR and three BWR candidate concepts should be evaluated and screened on the basis of their feasibility. To perform this screening process, Sandia subcontracted with an architect-engineering firm experienced in designing nuclear power plants--Burns and Roe, Inc. With an emphasis on the backfitting capability of each concept, Burns and Roe applied the Sandia design criteria discussed in Section 2.3 for probabilistically evaluated events and special emergencies, and then they evaluated the feasibility and impact of each concept using their best engineering judgment.

The screening process considered five major factors:

1. functional capability
2. compliance with the design criteria
3. feasibility of construction
4. potential costs
5. operational and maintenance difficulties.

The ability of each candidate concept to meet these factors satisfactorily was rated subjectively on a scale from 1 (lowest) to 10 (best). The ratings were then multiplied by a weighting factor intended to reflect their relative importance. The cost and operational screening factors were weighted by 5 and 2, respectively, while the functional capability, criteria compliance, and construction feasibility factors were weighted by 10. Table 3 summarizes the results of the concept screening.

From Table 3, it can be seen that for PWRs the add-on high pressure injection (i.e., feed and bleed) and the add-on auxiliary feedwater concepts scored best, and for BWRs the low pressure injection/suppression pool cooling concept ranked highest. In general, these three concepts were judged to be simpler, less expensive, and more easily retrofitted than the other candidate concepts which scored lower. The add-on high pressure residual heat removal system and closed loop auxiliary feedwater system were estimated to require large components and a service water dependency which could prove to be a major retrofit liability, although a new-design plant could accommodate these concepts much more easily. The passive feed-water system ranked poorly because it could increase the pressure inside containment beyond original design limits during a postulated steam line rupture and because it requires a pressure vessel that is currently unavailable. The "passive" makeup pump was judged to be unbuildable as a truly passive device, and even if it could be built as envisioned, it would require a developmental program since it is currently unavailable. The BWR high pressure cooling and controlled depressurization concepts were assessed to provide no significant improvement over the low pressure option, even though they appear to entail greater complexity and cost than the low pressure system.

Table 3

Concept Screening Summary

	Functional Capability	Criteria Compliance	Construction Feasibility	Construction Cost	Operational Expense	Total[1]
PWR Concepts						
Add-on HPSI	100	100	100	45	20	365
Add-on Aux. Feedwater	90	100	80	40	18	328
Closed Loop Aux. Feedwater	80	90	60	15	10	255
Add-on RHR	80	80	30	5	4	199
Passive Feedwater	50	50	10	5	10	125
Passive Makeup Pump	10	50	10	50	2	122
BWR Concepts						
Low Pressure Cooling	100	100	100	50	20	370
High Pressure Cooling	90	100	90	40	20	340
Controlled Depressurization Cooling	80	100	80	35	10	305

NOTE: 1) Maximum Total 370

The purpose of ranking the alternative decay heat removal systems in Table 3 was to identify a few of the more promising concepts for a more detailed value/impact evaluation. It seemed inappropriate to develop any concept further, if serious questions could be raised regarding its feasibility. In this study the ability of a concept to be retrofitted was judged to be of major importance, because of a belief that existing older power plants will most likely derive greater safety benefit from an alternative decay heat removal system than will newer, state-of-the-art, power plants. The value assessment results presented in Section 5 support this position.

The following subsections discuss the underlying reasons for the concept rankings presented in Table 3. On the basis of these rankings, the PWR add-on high pressure injection and auxiliary feedwater concepts and the BWR low pressure injection and cooling concept were selected for the detailed value and impact evaluations presented in Sections 5 and 7. It should be recognized that for each of the concepts considered, the discussion of pros and cons presented here places emphasis on identifying the drawbacks of each concept. This emphasis on the drawbacks should not be interpreted to mean that the detriments of the concepts always outweigh their benefits. Instead, it should be recognized that the feasibility of a concept, most likely, will be determined by the extent to which engineering problems can be solved. In general, the concepts that ranked best not only satisfy the design criteria for the alternative systems, but they also pose drawbacks which were judged to be less significant than those associated with the poorer ranking concepts.

3.1 Add-on High Pressure Injection Train (See Section 2.4.1.1)

Pros

This concept would use commonly available components with conventional sizes and capacities. The pump used could be of similar type and manufacture as the existing plant high pressure safety injection system pumps, thus enhancing operation and maintenance. The rest of the components are all standard items readily obtainable. The system does not rely on heat exchangers to function and, hence, does not need a source of service water (either existing or new). This makes the construction both easier and less costly than for heat exchanger based concepts, and also increases the system's resistance to sabotage.

Cons

This system has a large number of active components, all of which must be maintained and inspected routinely to assure reliability. The addition of another 100 percent capacity high pressure injection system could exceed the relief capacity of the safety relief valves. This raises the potential for over-pressurization in the event of a simultaneous actuation of all high pressure makeup/injection systems, and therefore, the addition of another ASME III Class 1 relief valve might be required. This system will dump hot reactor coolant into the containment building sump causing LOCA conditions even in the absence of a LOCA. In view of the flow and temperatures involved, the conditions inside of the containment would probably be acceptable for the 10 hours specified by the design criteria. However, in the absence of containment cooling systems, the temperature and pressure reached in containment at the end of 10 hours might be substantial and could lead to serious degradation of containment systems and components. Depending on the volume of water to ultimately be injected into the containment building, portions of the reactor vessel may become submerged. If the temperature of the metal has not been lowered sufficiently before this happens, serious damage could result.

The proposed tie-in to the containment sump seems impractical. A traditional problem of high pressure injection (HPI) system pumps is a sensitivity to net positive suction head (NPSH) requirements. Existing HPI pumps are always located at a lower elevation than the containment sump elevation for this reason. The NPSH margin for these pumps is quite small, and the additional pressure losses associated with placing an add-on pump, similar to existing HPI pumps, several hundred feet away would degrade its performance. It would require large line sizes to reduce the piping losses and would need to be at a lower elevation than the normal HPI pumps. This would require an extensive excavation effort and involve retrofitting large diameter pipe (12"-20").

New tie-ins to existing systems inside containment could pose a serious construction problem if no suitable spare penetrations can be found. The concept would require from two to four (depending on the number of reactor coolant loops) spare penetrations. Locating enough spare penetrations in the correct area may be a problem, and it may be necessary to tie in to the safety injection lines outside of containment. If the tie-ins are made in the containment building, at least a portion of the line must be considered high energy, with the potential for increasing the chance of a pipe break accident.

3.2 Add-on Auxiliary Feedwater Train (See Section 2.4.1.2)

Pros

This sytem would use commercially available equipment, with conventional sizes and capacities. No new engineering or certification is required. Plant operations and maintenance would not be handicapped because the new components can be identical to existing system components and adequate spares and maintenance experience will be on hand. The system does not rely on heat exchangers to function and hence, does not need to have a source of service water (either existing or new). This makes the construction both easier and less costly than for heat exchanger based concepts, and also increases the system's resistance to sabotage (since intake structures are vulnerable).

Cons

The system has a large number of active components, all of which must be maintained and inspected periodically to assure reliability. It is unlikely that there would be enough spare containment penetrations for 4" and 6" pipes available, and it may be necessary to make the required tie-ins to existing piping systems in the auxiliary or intermediate buildings. While this would increase the system's dependence on existing systems, it would decrease the cost and complexity of the system. The number of tie-ins to existing systems is large. Because there must be a minimum of four (4) breaks postulated for each high energy pipe, a minimum of 24 new pipe breaks to be analyzed (40 for a four-loop plant). These new breaks might pose unacceptable jet impingement loads to surrounding equipment and could force the equipment to be relocated.

3.3 Closed Loop Auxiliary Feedwater Train (See Section 2.4.1.3)

Pros

This system would use commercially available components, thus easing the spare parts and maintenance difficulties. The steam line tie-ins could be outside of containment since the main steam isolation valves are typically outside containment. A comparable system is currently being offered by one of the major PWR manufacturers as an option for new plants.

Cons

The system has a large number of active components which must be maintained and inspected in order to assure reliability. Because the new condenser will require cooling water, a new service water system will be required. This service water system is large and will be difficult to retrofit. Since the existing intake structure cannot be utilized, a new intake will have to be constructed. The system closely resembles the add-on auxiliary feedwater system, except that it replaces a 200,000-gallon storage tank with a more complicated auxiliary condenser and service water system.

3.4 High Pressure Residual Heat Removal Train (See Section 2.4.1.4)

Pros

This system would provide the capability to cool the reactor from SCRAM to hot (or cold) shutdown. It could do this without lifting relief valves or degrading the containment systems. This would allow an orderly recovery from the transient in spite of a large range of potential problems. By forcing coolant circulation through the core, the chance of fuel damage is greatly reduced.

Cons

The system has a large number of active components, all of which must be maintained and inspected to assure reliability. Many of these components would be inside containment, thus making these activities even more difficult. The components would be very large and locating them in an existing containment building may present an insurmountable problem. The heat exchanger, for instance, would be around 60 feet long, 7 feet in diameter, and would weigh nearly 300,000 pounds flooded. This presents not only a space problem but a significant seismic problem. In addition, such a heat exchanger would require an additional 45 feet of length for tube pull space. The pump would be around 25 feet long, 10 feet in diameter, and would weigh around 150,000 pounds. These components would be virtually impossible to retrofit into an existing containment building. The flow rate of nearly 20,000 gpm would dictate using a 16"-20" nominal diameter piping. Most current large tie-ins to the reactor coolant system piping are in the range of 10"-12". This would mean the addition of a new tie-in of the same order of size as the reactor coolant system piping itself. Routing and supporting such piping runs would be extremely difficult and costly. Also, the number of new postulated pipe break locations would increase substantially.

The preceding difficulties indicate that an outside containment variant might be preferable. However, this option also has many problems. Finding two spare penetrations in the 20"-30" range is unlikely. Even assuming the availability of such penetrations, there is then the problem of taking reactor coolant out of the containment building during an accident. If appreciable fuel damage has occurred, the dose rates around these lines will be prohibitive. Extraordinary steps would be needed to ensure the shielding and leak integrity of the system's components. The piping would also present a tempting target for sabotage since it raises the potential for a LOCA outside of containment.

The cooling water system for the high pressure residual heat removal heat exchanger would need to be supplied via piping of approximately 20" diameter. Hence, even if the high pressure residual heat removal system is located inside containment, two spare penetrations for piping in the range of 20" to 30" must be found. This is not the typical size of spare containment building penetrations and this fact would likely disqualify this option for retrofit considerations. In addition to this, the large cooling water flow rates would require a new large diesel generator, comparable in size to that of those typically used at nuclear power plants for all emergency power loads.

3.5 Passive Feedwater Train (See Section 2.4.1.5)

Pros

This concept would rely wholly on static head differences to operate. Thus, it would be a passive system requiring very little maintenance and inspection.

Cons

A torus concept would be impossible to retrofit into existing plants. The elevation required for the torus would be at or near grade. The reactor building is attached to several other buildings at this elevation, and to retrofit a torus into an existing design would require a massive construction effort. Placing the torus inside of the containment building would be even worse because of interferences, difficulty of supporting it seismically, and construction outage time. The concept can only proceed on the basis of a tank, or tanks, in the yard around a plant. However, even in this instance, the following special criteria would need to be met by the tank(s):

a. design pressure equal to main steam system 1100 psig
b. volume of 200,000 gallons
c. fully vented to steam generator with separate relief valves
d. top of tank not higher than "low" level elevation of steam generator
e. static head flow capability of approximately 1200 GPM when tank is full
f. static head flow capability of 130 GPM when tank is nearly empty
g. static head flow capability consistent with decay heat load for the first 10 hours after shutdown

Criteria a and b above indicate that the storage tank(s) would actually need to be large ASME pressure vessels. Criterion c follows from this fact. Criteria d, e, f, and g place very severe limitations on the size and shape of these "tanks". The difference in elevations being described is not very large. Although this difference in elevation may vary from plant to plant, it is usually no more than 10 or 15 feet.

The density difference must also be considered. The water in the storage tank will be much cooler than the water in the steam generator. As a result, the density of water in the steam generator may be only 50 to 75 percent of that in the storage tank. Thus, a foot of "water" column in the steam generator may only be 6 to 9 inches of water in the tank in the yard. Thus, the tanks must be very low for the sort of pressure and capacity required.

The pipe run from the storage tank to the steam generator will need to be of nearly constant elevation for this option to function. This places several constraints on the system piping design. The smallest pipe size that will allow a flow rate of 1200 GPM at less than 15' of head is 8" nominal (assuming 500 feet of effective pipe length using schedule XS pipe). This means that at least one spare 8" penetration must be found in containment at an elevation less than the bottom of the storage tank. Once inside containment, the piping system would have to be routed below the elevation of the bottom of the storage tanks. Since the existing emergency feedwater nozzles on the steam generators are typically located too high on the vessel for this concept, new nozzles will need to be added specifically for this system. Also, new penetrations through the "Biological Shield" D-ring will probably be necessary due to the location of the new nozzles. This piping layout would be difficult to construct in an existing plant and the probable need for new steam generator nozzles and D-ring penetrations would force a long plant outage during construction.

Another serious shortcoming of this design is the lack of containment isolation valves. It is difficult to rationalize how such a system could ever be licensable without isolation valves. If automatic isolation valves are added, much of the advantage of the system is destroyed since it would no longer be a passive system. Check valves could be used in the feedwater line for isolation, but that would leave no method of controlling the flow path in the event of a secondary pipe failure in containment. If a single tank feeds all steam generators, any secondary side pipe failure would automatically blow down all steam generators, thus over-pressurizing the containment building. This fact demonstrates the need for a separate storage tank for each steam generator. However, each tank would need to be full-sized to allow the cool down of the plant using any combination of steam generators. Even in this case, the rupture of any steam generator system piping (e.g., main steam lines or feedwater lines) could automatically inject an extra 200,000 gallons of hot water into the containment building from the passive feedwater tank. Much of this water would flash into steam, and it could seriously impact containment pressurization.

In addition to these factors, the passive feedwater system could represent a serious parasitic heat loss for a power plant, because it is maintained on line with the isolation valves open. The pressurization line between the steam generators and the passive feedwater tank would constantly draw steam from the steam generators and condense it in the tank(s). This hot condensate will displace the cooler water in the feedwater tank, thereby injecting water at a temperature less than normal feedwater temperature. Efforts made to reduce the heat loss from the tank would elevate the tank's water temperature and worsen the effects of a steam or feedwater line rupture. Raising the water temperature will also increase the cost of fabricating an already expensive tank.

3.6 Passive Makeup or Circulation Pump (See Section 2.4.1.6)

This concept is not a system but a proposal to use a "passive" pump for any of the preceding systems requiring pumps. The proposed pump, called a "Shock Condenser," is an injector which uses steam to deliver more discharge pressure than either driving pressure or suction pressure. The use of steam injectors for feedwater pumps is an established, although generally discarded, technology. Such pumps have been demonstrated to operate in small boilers and in old steam-driven locomotives.

Pros

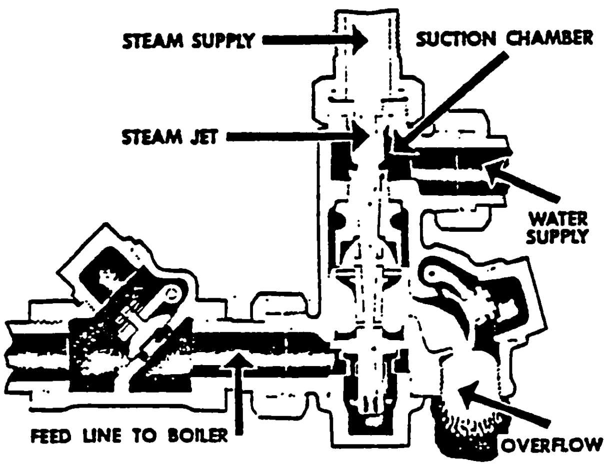
The shock condenser pump would not rely on an auxiliary power supply and would thereby eliminate much of the complexity and cost of a conventional pumping system. Since these pumps have fewer moving parts than conventional pumps, their maintenance requirements could be less, and if estimates of the pump dimensions are correct (i.e., approximately 10-foot length by 6-inch diameter),[21] installation of the pump would present no serious space problems in an existing power plant. Because the pump can lower both the temperature and pressure of the reactor coolant system as it removes decay heat, it can possibly reduce the effects of small LOCAs and the problems of pressurized thermal shock.[22]

Cons

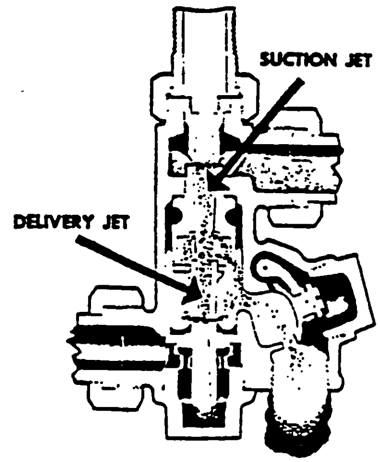
It may be a misnomer to call a steam injector a "passive" device. While injectors are less complicated than conventional pumps, their current configuration has several moving parts, and they should be classified as active components (Figure 10). Also, even if the injection pump itself were passive, the system in which the pump is used would require a large number of active components (Figure 6). Current state-of-the-art injectors are "single point" devices. This means that they will only function over an extremely narrow range of steam and suction water conditions. The proposed use of an injector pump would control neither of these parameters.

The proposed use on the primary side of a PWR would seem to present insurmountable problems. First, starting a steam injector requires a period of blowdown until the dynamic head inside the injector is enough to lift the discharge check valve. Hence, starting an injector always causes at least a temporary LOCA and a

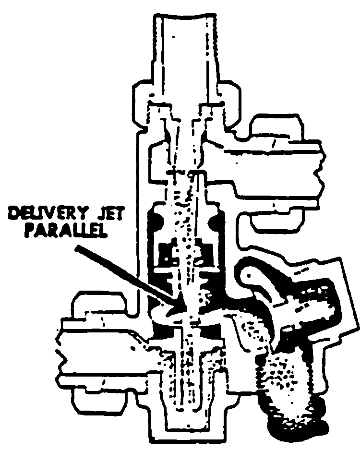
Operating Principle of Penberthy* Automatic Injectors



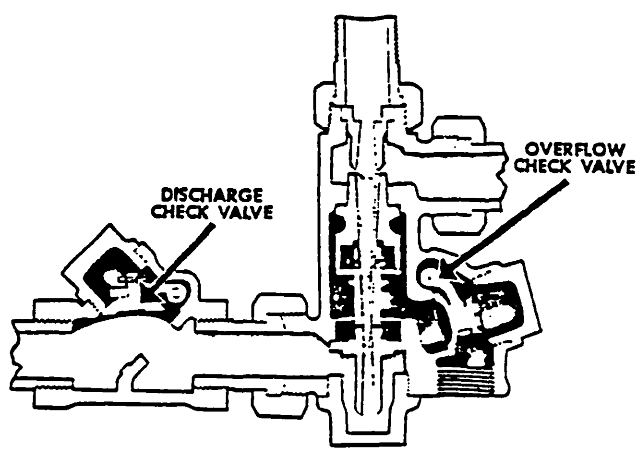
When steam supply valve is opened, steam passes through steam jet into suction chamber, proceeds through suction jet and out of the overflow. Steam, which attains a velocity of approximately 2500 ft per second as it leaves the steam jet, entrains the air in suction chamber and creates a vacuum.



The vacuum created in suction chamber begins to draw in water from supply line. The water is now entrained by the steam and a high velocity mixture of water and steam passes through the suction jet and out of the overflow.



When the amount of steam and water reach the proper proportion, the steam gradually condenses as the mixture advances through injector. Upon reaching delivery jet parallel the mixture is fully condensed.



The energy contained in the water passing through delivery jet is sufficient to build up a pressure, greater than the boiler pressure, causing water to flow through the discharge check valve into the boiler. When flow into the boiler is established the overflow valve closes automatically and prevents the entrance of air which would disrupt operation of injector. Total operating cycle requires only a few seconds.

* Formerly Penberthy Injector Co.

Figure 10. Operating Principle of Penberthy Automatic Injectors

failure to start results in a LOCA. Starting any steam ejector/ injector requires both the rapid arrival of high quality steam and the timely arrival of the suction water. Without either of these, the ejector/injector can easily overheat and "vapor lock". At the time of actuation on the primary side of a PWR, isolation valves open to the inlet of the injector and "blow down" the hot leg. Bulk boiling in the hot leg forces slugs of water and low quality steam into the steam injector. Assuming that the destructive force of this does not destroy the injector, it is unlikely that it would start before over-heating.

Another difficulty with startup of this type of pump is that it requires a reservoir of feedwater for the blowdown phase. The proposed system has a closed suction water system with a fixed volume of water. Hence, the system startup could reduce the water inventory available for system operation. The addition of some sort of passive makeup capability only reduces the severity of this problem because injectors normally stop and restart intermittently during operation, and the makeup capacity would have to be quite large.

The heat exchanger required to operate the injector (Figure 6) is physically large and probably could not be retrofitted inside of an existing containment building. Hence, the heat exchanger would have to be located outside of containment and the connecting piping would have to be quite large to minimize pressure drop. Finding suitable penetrations for such an arrangement is highly doubtful. Also, this would mean bringing reactor coolant out of containment during an accident and would raise the possibility of a LOCA outside of containment.

For these reasons, the injector pump was considered more appropriate for use on the secondary side of the plant. Still, even here, there are serious doubts about its suitability. First, it appears that conventional turbine driven auxiliary feedwater pumps are inherently more reliable and more flexible than injector pumps, because injectors historically stop operating whenever inlet conditions change or whenever debris clogs the steam jets. Second, once outside containment, a number of other less exotic expedients can be taken to inject feedwater into the steam generator. Third, inservice inspection of a steam injector presents a practical problem for the maintenance personnel even if the injector is on the secondary side. Since each "test" will cause severe perturbations in the main steam and feedwater systems, it will be necessary to shut down the plant for each inspection or to build a dedicated high pressure auxiliary steam supply to test it.

3.7 Low Pressure Makeup and Suppression Pool Cooling Train (See Section 2.4.2)

Pros

This system is of conventional size and design. All the components are readily available and could be identical to existing components, thus easing the maintenance difficulties. By allowing the suppression pool to take the initial decay heat load rather than the heat exchanger, the low pressure cooling heat exchanger can be sized the same as the existing residual heat removal heat exchangers. Since this system depressurizes the reactor, it would serve to minimize the blowdown from small LOCAs. Since this is a low pressure system, it is compatible with the normal residual heat removal conditions. The reactor would be ready to line up to normal long term cooling systems at the end of the 10-hour design basis period. This would allow smooth transition from alternative system to plant system.

Cons

The system has a large number of components which must operate in order for it to function. This would require a routine maintenance and inspection program in order to assure the availability of this system.

Three (3) fairly large containment and dry well penetrations are required for this option. These may not be available as spares, especially in the older plants. Also, the components and piping of the system are rather large for retrofit applications, especially inside an existing containment building. The system would quickly depressurize the reactor vessel causing a rapid lowering of the water level, with makeup relying on the ability of a diesel generator to start, come up to speed and voltage, and to assume full load. If the diesel, or any other component, fails to function, the system could aggravate, rather than mitigate, the consequences of a loss of normal decay heat removal capability accident. Clearly, this could be minimized if the instrumentation interlocks are carefully designed to avoid unexpected or spurious actuations.

3.8 High Pressure Makeup and Suppression Pool Cooling Train (See Section 2.4.2)

Pros

This system is conventional in scope and design. Since it would be similar to the reactor core isolation cooling system, it represents, in principle, a proven decay heat removal concept. All of the components are currently available and can be designed similar to existing plant components, thus easing maintenance difficulties. It provides a method of cooling the reactor at or near the design pressure. This avoids pressure transients and hence level fluctuations typical of the automatic depressurization system.

Cons

This system would require routine maintenance and inspection to assure operability. If the system operates as a closed cycle and does not take credit for the suppression pool heat capacity, the required heat exchanger would be larger than the normal residual heat removal heat exchangers. This, in turn, would force the dedicated service water system for the heat exchanger to be quite large and necessitate housing the equipment in a larger and more expensive building than any of the other BWR concepts. However, even if credit for the suppression pool heat capacity was allowed, large equipment sizes would make retrofitting it into most containments quite difficult, if not impossible. Since it seems that the heat exchanger must be located in the new building, the routing of the steam line(s) will be a major undertaking. As such lines would be subject to numerous dynamic loading conditions, it would be extremely difficult to route them out of an existing building with all of the interferences.

The system would attempt to maintain the reactor coolant system pressure at or near its normal level, and therefore, the high pressure nature of this system would serve to maximize the blowdown rate from a break and thus will aggravate the effects of a LOCA. Because this system would be capable of developing pressure equal to the design pressure of the reactor vessel, it could raise the possibility of over-pressurizing the reactor vessel in the event of a simultaneous actuation of all high pressure makeup systems. If the existing safety relief valves cannot handle the extra capacity of this additional high pressure injection system, a new Class 1 safety/relief valve which complies with the ASME boiler pressure vessel code may be required.

3.9 Controlled/Variable Pressure Makeup and Suppression Pool Cooling Train (See Section 2.4.2)

Pros

This system is of conventional size and design. All components are readily available and could be identical to existing components to ease maintenance difficulties.

By allowing the suppression pool to take the initial decay heat load, the system heat exchanger can be of low pressure rating, and it can be sized the same as the existing residual heat removal heat exchangers. Since this system depressurizes the reactor, it would serve to minimize the blowdown from small LOCAs. Also, because the operating pressures span the range of pressures, the system pump(s) would be well suited to handle a spectrum of LOCA accidents.

The system would depressurize slowly, thus preventing the rapid lowering of reactor coolant level. The system would allow a gradual transition to the low pressure range and would ensure a steady rate of decay heat removal. When the system has completed its 10-hour design operation period, the reactor would be at a pressure compatible with normal residual heat removal conditions to allow a smooth transition from the alternative system to normal plant safety systems.

Cons

This system has a large number of components which must operate in order for the system to function. This would require a routine maintenance and inspection program in order to assure the availability of this system.

Three (3) fairly large containment and dry well penetrations are required for this option. These may be extremely difficult to locate, especially in older plants. Also, the pipe sizes are rather large for retrofit applications. The components for this option are quite large physically and it is unlikely that any of them could be retrofitted into an existing plant. All new components would have to be located in the new building. This option is more complicated than the other two BWR concepts. Two pumps may be necessary to cover the range of pressures required. Also, it is possible that in the startup of this, or any other auxiliary system, the normal plant decay heat removal systems may have already actuated the automatic depressurization system (ADS) relief valve(s). Hence, the reactor vessel may already be at the lower pressure range when this system begins operation. Even if ADS is terminated, ADS valves have a rather high "fail open" rate. Thus, the extra complexity and expense of this system may not be necessary.

4.0 Impact Assessment

Based on the feasibility assessment in Section 3.0 of nine candidate alternative decay heat removal systems, one BWR concept and two PWR concepts were chosen for a more detailed evaluation of their value and impact. These concepts are:

- an Add-on High Pressure Injection Train (PWR)
- an Add-on Auxiliary Feedwater Train (PWR)
- an Add-on Low Pressure Makeup and Suppression Pool Cooling Train (BWR)

This section discusses the results of an evaluation of the potential impacts of these concepts.

4.1 Plant Variations Assessed

In order to perform a realistic impact assessment, it was recognized that major design features unique to each concept and to each power plant to which a concept is applied must be defined. Without this, a credible evaluation cannot be made of costs, system interface requirements, retrofit difficulties, and construction schedules. The impact assessment technique adopted to account for these factors involved performing a preliminary design of each concept for several different nuclear power plants. Burns and Roe, Inc., an architect-engineering firm, was subcontracted to perform this impact evaluation effort, and Appendices A, B, and C present the conceptual design details which Burns and Roe developed for the alternatives.

Appendices A and B describe the PWR add-on high pressure injection train and add-on auxiliary feedwater train concepts, while Appendix C describes the BWR add-on low pressure makeup and suppression pool cooling concept. All three of these appendices are subdivided into three subsections. In Appendices A and B, the subsections present flow diagrams, equipment arrangement drawings, electrical one-line diagrams, control and actuation logic drawings, equipment descriptions and sizing information, and system operational and testing requirements for the PWR concepts as applied to three different PWRs. Appendix C provides similar information for the BWR concept, as applied to three different BWRs.

The power plants to which the alternative concepts were applied may not be identified by name, but they do represent all U.S. LWR reactor manufacturers and several different design vintages. In particular, the power plants used for the impact evaluation were:

Combustion Engineering PWR	-	3410 MW (thermal rating)
Babcock and Wilcox PWR	-	2772 MW (thermal rating)
Westinghouse PWR	-	1876 MW (thermal rating)

General Electric BWR - 3462 MW (thermal rating)
General Electric BWR - 2381 MW (thermal rating)
General Electric BWR - 1930 MW (thermal rating)

Each of these power plants corresponds to a particular site location and plant layout, and although the six plants should not be construed to represent all sites, Table 4 indicates that the plants encompass a wide range of site variations.

4.2 Important Design Characteristics and Engineering and Operational Difficulties of the Alternative Concepts

As a result of the preliminary design work reported in Appendices A, B, and C for the add-on concepts, a number of engineering and operational difficulties were found to influence the practicality of the concepts. In order to address these problems, the concepts were developed to incorporate as many design characteristics which were dictated by the original design criteria as appeared to be reasonably feasible. This section presents some of the more important design characteristics that were influenced by the alternative DHR system design criteria. Section 4.2.1 presents those characteristics common to all three alternative concepts, while sections 4.2.2 through 4.2.4 discuss features unique to each alternative. Where appropriate, engineering and operational difficulties identified during the preliminary design effort are discussed.

4.2.1 Design Characteristics Common to Each Alternative Concept

4.2.1.1 Electrical System

All three alternative decay heat removal concepts developed for this study were configured to have a similar electrical system. The electrical power requirements for the add-on trains are normally provided by an existing Class 1E feeder breaker, in a manner similar to that used to feed the Class 1E buses from the off-site power sources in many existing plants. Upon a loss of the Non-Class 1E power, the circuit breaker interconnecting the add-on electrical system with the existing plant's system will open, the add-on diesel will start, and when up to speed and voltage, its output breaker will close to reenergize the add-on 1E bus. Upon receipt of a system actuation signal, all non-essential electrical loads are shed from the add-on bus. Electrical loads essential to the operation of the system are then sequenced onto the bus on the basis of size and importance. The add-on diesel will only operate in the event of the loss of offsite power. This allows the add-on system to actuate without relying on the add-on diesel's ability to start if off-site power is still available.

Table 4

Comparison of Plant Sites Used for Impact Evaluation

Plant	Location	Site EL[1]	Flood EL[2]	SSE	OBE	Tornado	Nearest Water Body	Special Notes
3410MW - CE	East Coast	29'	22'	.22g	.11g	300 mph	Tidal Estuary	-
2772MW - B&W	Northeast	304'	301'	.12g	.06g	320 mph	River	Near Airport
1876MW - W	West Coast	59'	60.4'	.40g	.20g	160 mph	Ocean	Near Active Volcano
3462MW - GE	Northwest	441'	373'	.25g	.125g	214 mph	River	Near Active Volcano
2381MW - GE	Midwest	903'	903'	.20g	.10g	380 mph	River	Near Navigable River
1930MW - GE	East Coast	22'	22'	.22g	.11g	300 mph	Tidal Estuary	-

1. Elevation with respect to Mean Sea Level (MSL).
2. Maximum flood elevation with respect to MSL.

If off-site power is recovered after the add-on diesel has been started, the alternative decay heat removal system can be restored to its normal lineup by paralleling with the plant electrical supply, shifting loads, and opening the diesel generator circuit breaker.

Add-on diesel generator testing would be accomplished by paralleling the add-on system electrical loads with the existing plant bus. Operation would continue in parallel at the required load for the specified time, after which the load would be returned to the normal source, the diesel breaker opened, and the diesel returned to its standby mode.

As configured, the add-on systems can be manually loaded onto the existing emergency diesels or a portion of the existing 1E power systems can be powered from the add-on diesel. However, these procedures would involve serious technical specification violations and would be reserved for desperate emergencies only. With regard to powering an existing power plant 1E bus with the add-on diesel generator, an investigation showed that the sizes of the add-on diesel generators are too small to replace an existing emergency diesel generator (see Table 5). No capability to handle existing plant Class 1E loads with the add-on diesel generators was incorporated into the alternative concepts.

4.2.1.2 System Interfaces

Each conceptual system was designed to be able to carry out its required function without existing plant service systems, electrical systems, or operator action. The designs are all in accordance with existing regulatory practices and should have no adverse impact on either plant operations or existing safety systems. However, there is the need to interface the add-on concepts with some existing plant systems, particularly the fluid systems which serve the reactor (i.e., reactor coolant system) or the steam generators (i.e., feedwater system).

In an effort to minimize the vulnerability of the add-on concepts to common mode failures or special emergencies, a design objective initially was established to route piping for the add-on concepts as far away from other existing pipe runs as possible and to interface the add-on piping with existing plant systems inside containment buildings. However, it was often found that making tie-ins inside of existing containment buildings is extremely difficult. It is impossible to make new penetrations into an existing containment building safely. Because of prestressing, the entire containment structure is at a high level of compressive stress. This makes the installation of a new penetration extremely dangerous since the removal of material from the containment wall will form a local stress concentration which could lead to a progressive failure of the structure and jeopardize the integrity of the entire containment building. Therefore, tie-ins inside containment can only be made through existing, spare penetrations.

Table 5

Diesel Generator Comparison for Alternative Concepts
(Ratings in KW)

	PWR			BWR		
Plant	3410MW CE	2772MW B&W	1876MW W	3462MW GE	2381MW GE	1930MW GE
Existing Diesel Generator	7200	3000	4400	4400	4000	2500
Add-on HPI Train Diesel Generator	1700	2000	1400	-	-	-
Add-on AFW Train Diesel Generator	2100	2300	1900	-	-	-
Add-on Low Pressure Makeup and Suppression Pool Cooling Train Diesel Generator	-	-	-	1400	1200	1200

Spare penetrations in existing containment buildings are often inadequately sized, insufficient in number, or inappropriately located to accommodate the add-on systems. The penetration constraint forces the layout of the retrofit piping in and around the containment to be complicated and cumbersome. Supporting this new piping seismically would be an elaborate and costly effort. All of the piping involved inside containment is either Class 1 or Class 2, which complicates the design and construction effort further and adds significantly to the cost. All retrofit work inside containment would force a significant amount of plant downtime and involve some increase in man-REM exposure to the work crews, both of which have significant economic impacts.

Another liability of making tie-ins inside containment is that most of the lines involved are high energy pipes. This will make at least a portion of the add-on piping high energy and increase the number of postulated pipe rupture locations inside containment. If the tie-ins were made outside of containment, these same lines are usually considered moderate energy and, hence, have much less of an impact on the plant.

As for the concern about sabotage or other special emergencies if the tie-in is made outside of containment, this study has found that the add-on piping often had to be routed very near existing ESF piping in order to get to spare containment penetrations. In these cases, real separation was found to be an unachievable goal, because a saboteur-induced failure or other type of common mode failure would need only to occur in one area in order to disable both the existing and add-on systems, regardless of the location of the actual tie-in. This was especially true of the newer plants that had already employed extensive separation criteria.

Tie-ins outside of the primary containment were found to be permissible only when existing valves inside containment and downstream of the tie-in point were check valves. Because of prohibitions against powering one electrical device from two different Class 1E power systems, any system that had a motor operated isolation valve downstream of the tie-in point could not be used. Pneumatically operated valves could be controlled by the add-on system, but typically these were not used as isolation valves in the plants used for this study.

Pneumatic valves were used in this study for control or letdown purposes whenever this capability was needed and existing pneumatic valves were available. This was accomplished by add-on pneumatic systems that blocked the existing pneumatic control signals when actuated. This resulted in a significant cost and leadtime reduction, and also decreased the complexity of the retrofit work which served to reduce the length of the forced outage. Plants which did not utilize pneumatic valves required the installation of a dedicated valve that duplicated the function of the existing valves. This proved to be a serious liability.

Tie-ins for the BWR option were all made in the reactor building outside of the primary containments. No attempt was made to add new penetrations to the dry well or the suppression chamber. Tie-ins for the PWR options were made either in the reactor building or in the auxiliary building, depending on the power plant being considered. For the Babcock and Wilcox plant, tie-ins inside containment were used; while for the Combustion Engineering plant, tie-ins were made in the auxiliary building. For the Westinghouse plant, high pressure injection and makeup tie-ins were done inside containment, and add-on feedwater tie-ins were established in the auxiliary building.

4.2.1.3 Add-on Buildings

Each concept system was housed in two separate buildings located adjacent to existing power plant buildings. For the PWR concepts, one building contained the pump(s), diesel generator, switchgear, and control panels, while the second building housed the water supply tanks. This arrangement minimized concerns over internal flooding. For the BWR concept, one building contained the pump, heat exchanger, diesel generator, switchgear, and control panels, while the second building housed the service water pump for the residual heat removal heat exchanger. This arrangement allowed the service water portion of the BWR system to be located near the site ultimate heat sinks (i.e., rivers or tidal estuaries) and, simultaneously, minimized length of the RHR piping runs.

The add-on buildings were located in a manner which:

(1) minimized pipe runs from add-on buildings to the existing plant,
(2) provided separation from the existing diesel generators, safety related storage tanks, electrical switchyard, and ultimate heat sinks,
(3) was not subject to turbine generated missiles,
(4) was inside the existing security area and distant from the site periphery, and
(5) avoided existing safety related piping in the yard.

These often conflicting requirements were resolved using engineering judgment, and a more thorough treatment might reduce the cost somewhat since there was insufficient time in this study to optimize the layout of the alternative concepts.

All alternative decay heat removal system buildings are seismic Category I structures capable of protecting and maintaining the add-on systems in accordance with all the specified design criteria for separation and special emergency conditions. A sump drain system, heating and ventilation system, fuel oil system, diesel cooling system, and diesel startup system are installed in the add-on buildings to support operation of the alternative decay heat removal systems.

4.2.2 Characteristics of the Add-on High Pressure Injection Train (PWR)

The add-on high pressure safety injection concept was chosen as the most desirable of the PWR concepts because of its simplicity and because it acted directly on the reactor coolant system, thus performing both decay heat removal and reactor coolant inventory control simultaneously. The detailed design study did not change the basic advantages of the system. However, if one considers operation of this add-on system beyond the 10-hour design basis, operator action to mitigate the effects of both containment heating and flooding will be required.

Because long term cooling of the containment building was judged to be impractical with any "reasonable' retrofit system, it would be necessary to actuate an existing containment cooling system in order to operate the high pressure injection concept much beyond 10 hours. The easiest method identified to remove heat from containment involves operation of the containment ventilation/purge system, because the approach does not rely on a cooling water system. However, if fuel damage has occurred, direct venting to the environment becomes unacceptable, because it would release large quantities of radioactive fission products. Under these circumstances, other existing containment cooling systems would be needed, along with electric power. Unfortunately, the add-on diesel generator for the alternative DHR concept would be too small to operate most of these containment cooling systems, and therefore, plant operators would have to depend on starting an existing diesel generator. With regard to containment flooding after the 10-hour design period for the add-on concept, provisions would have to be made to energize a containment sump pump manually with the add-on diesel generator or to depend on starting an existing diesel generator.

One feature originally proposed for the add-on high pressure injecter concept was found to be impractical for a retrofit system and was deleted from the final design. This feature involved a recirculation connection from the containment sump to the add-on high pressure injection pump suction to facilitate extended system operation (see Section 2.4.1.1). A design study of the recirculation line verified that the required net positive suction head (NPSH) for the add-on HPSI pump could not be achieved on any of the three PWRs in this study without a significant impact on the injection concept. The addition of the recirculation line would add more than an estimated 30 percent to the cost of the entire system in order to install larger piping and to locate the add-on pump at a low elevation relative to the containment sump.

As an alternative to the recirculation line, tie-ins are provided to existing sources of water such as the existing ESF storage tanks (borated water storage tank, refueling water storage tank, etc.) or the demineralized water storage tank. This is a much more cost effective way of ensuring additional injection capability while retaining potentially contaminated water inside the containment building. However, for purposes of assessing the value of this concept in reducing core melt frequency, no credit was given for recirculation capability or for extended operation beyond 10 hours.

The add-on system borated water storage tank is maintained with a boron concentration consistent with normal ESF storage tank concentrations (2200-4400 ppm). Circulation pumps are provided to prevent corrosion that takes place when stagnant borated water is in contact with stainless steel, and immersion heaters are also used to ensure that the water temperature remains above 40°F. Because of this boron injection capability, credit was taken in the value assessment for this alternative system mitigating accident sequences involving failure to scram.

Originally, this concept was configured to rely upon the safety relief valves for blowdown; however, this was judged to be unacceptable because the safety relief valves could not be relied on to consistently reseat over the 10-hour design period. The concept now relies on either a dedicated temperature control valve (TCV) which is sized to relieve only the flow required for decay heat removal or, if possible, independent control over one of the existing power operated relief valves. The pump for this concept has been sized so that its shutoff head is just adequate to lift the PORV should the TCV or the power operated relief valve fail to operate. The rated flow at shutoff head is sufficient to remove the decay heat a short time after the steam generators boil dry.

The add-on high pressure injection system is actuated by dedicated instrumentation added to the reactor coolant system (RCS), utilizing a two-out-of-four logic for high RCS temperature or a two-out-of-three logic for low RCS pressure. Two-out-of-four logic is used on the temperature because the thermal wells are on the hot legs and four instruments are required to get a reliable indication in the event of stagnant or insufficient flow through one of the reactor coolant loops. The set points are chosen to actuate the alternative decay heat removal system only if other safety systems have failed to mitigate the results of the transient. The TCV is controlled by RCS hot leg temperature indication and will prevent the cooldown rate from exceeding 100°F/hr.

4.2.3 Characteristics of the Add-on Auxiliary Feedwater Train (PWR)

The add-on auxiliary feedwater train was chosen for evaluation because of its relative simplicity and its similarity to existing systems. The detailed preliminary design study did not change the basic advantages of this system. However, it was determined that this concept requires the addition of at least some pressurizer heater capability in order to maintain pressure control of the reactor for the specified 10-hour design period. The addition of new heaters was not attractive because of space limitations in the pressurizers and therefore, all of the designs in this study have one group of steady state heaters that may be energized from the add-on auxiliary feedwater Class 1E bus. The control of this heater group has been modified so that either the existing controls or the add-on controls can operate the heater group.

This arrangement for controling pressure presents a difficulty in that pressurizer heaters are not usually Class 1E or safety related components. This not only forces the add-on system to rely on a non-safety component but is a point at which a non-Class 1E electrical failure could jeopardize the add-on Class 1E bus. Because of the potential unreliability of the heater, the alternate makeup pump, which is part of the auxiliary feedwater concept, has been sized and can be controlled to provide some degree of pressure control over the entire range of system conditions. Both the makeup pump and pressurizer heater group are controlled by an add-on saturation processor.

The capacity of the add-on makeup pump has been sized on the following basis: small break LOCA, shrink, and leakage. Small break LOCAs were estimated to require 150 gpm. The leakage amount was taken from the final safety analysis report technical data and the maximum shrink rate was calculated based on the maximum cooldown rate of 100°F/hr. This resulted in a pump that was similar to, or bracketed by, the capacities of the existing charging pumps and high pressure safety injection pumps for the plants reviewed (see Table 6). The removal of the small LOCA requirement would not result in a significant cost reduction for this option.

The add-on system borated water storage tank for the makeup pump in this concept is maintained with the same boron concentration as the normal ESF storage tank concentration (2200-4400 ppm). Circulation pumps are provided to prevent corrosion that takes place when stagnant borated water is in contact with stainless steel, and immersion heaters ensure that the water temperature remains above 40°F. Because of this boron injection capability, credit was taken in the value assessment for this alternative system mitigating accident sequences involving failure to scram.

The add-on system removes decay heat by feeding water into the steam generators where the water boils and forms steam. The steam is released from the steam generators via an add-on atmospheric dump valve (ADV) or redundant control of an existing ADV. This dump valve is controlled by the reactor coolant hot leg temperature for the loop feeding the appropriate steam generator. The valve is controlled to avoid a cooldown rate in excess of 100°F/hr.

Table 6

Comparison of Add-on and Existing Makeup Systems

	3410 MW(t) CE	2772 MW(t) B&W	1876 MW(t) W
Add-on Pump	300 gpm @ 4625' 600 HP	310 gpm @ 4625' 600 HP	100 gpm @ 4625' 200 HP
Number	1	1	1
Existing Makeup Pump	40 gpm @ 6300' 100 HP	300 gpm @ 5545' 700 HP	160 gpm @ 5100' 600 HP
Number	4	3	2
Existing HPSI Pump	460 gpm @ 3200' 900 HP		700 gpm @ 2750' 800 HP
Number	2	*	2

* B&W utilizes charging pumps for HPSI.

The add-on system actuates on a two-out-of-three logic from dedicated instrumentation added onto the steam generators to sense low steam generator level. This system has logic to prevent pumping to a failed steam generator. The add-on borated water makeup system to the RCS is actuated by low RCS pressure or by the startup of the add-on emergency feedwater train.

4.2.4 Characteristics of the Add-on Low Pressure Makeup and Suppression Pool Cooling Train (BWR)

This concept was chosen over the other BWR alternatives because it was simpler than the controlled depressurization concept and less expensive than the high pressure cooling concept. However, the detailed design effort has demonstrated that although this concept appears to be better than the other two alternatives, it still has serious shortcomings in meeting all of the criteria. As originally envisioned, the low pressure system would consist of a combination of automatic depressurization, low pressure core spray, and residual heat removal functions. However, in order to perform the low pressure core spray function immediately after a scram, the add-on low pressure concept proved to be too large to be retrofitted into an existing plant. Accordingly, the concept was revised to take credit for reactor core isolation cooling (RCIC) system operation during the first two hours following a scram.

By designing the low pressure add-on system to begin functioning two hours following a scram, the size of its components and piping could be substantially reduced as shown in Table 7. Unfortunately, an implicit assumption of this design approach is that the RCIC system can be relied upon to operate for two hours with only DC power available (i.e., two-hour battery capacity). However, there appears to be evidence that key components of some RCIC support systems rely upon AC power. Also, some BWRs do not even have a RCIC system. Despite these problems, however, the two-hour delayed operation of the add-on system was selected as a design basis for purposes of evaluating the value and impact of the low pressure option.

The flow rate of the add-on low pressure pump was based on the flow rate required to remove the decay heat load from the core two hours after reactor trip and to cool the suppression pool from 170°F to 140°F in 10 hours. This flow rate is delivered at a pressure consistent with reactor pressure conditions present at actuation. All coolant flow is directed to the reactor vessel until the suppression pool temperature exceeds 140°F. At this point, water is diverted to the suppression pool spray header and the flow is controlled by a temperature control valve.

The add-on heat exchanger has been placed downstream of the pump to minimize pressure drop to the suction of the pump. This forces the tube side to be rated for the maximum suppression pool

Table 7

Comparison of Component and Piping Sizes for BWR Low Pressure Cooling Concept As a Function of Time of Operation Following SCRAM For a 3400 MW(th) Reactor

Component	Sizes to Handle Flow Requirements at SCRAM	Sizes to Handle Flow Requirements 2 Hours After SCRAM
Suction Line Size	24"	8"
Discharge Line Size:		
From Pump	18"	8"
To Reactor	16"	6"
To Suppression Pool	8"	8"
ARHR Pump	7100 gpm @ 350'	2100 gpm @ 350'
Power	1700 HP	350 HP

pressure added to the shutoff head of the add-on pump. The heat exchanger has been sized to remove the residual heat load of the core two hours after reactor trip.

The service water system for the low pressure add-on system is based on a split-case horizontal pump rather than the more typical vertical turbine pump. This was done to eliminate the need for a separate intake structure and to make the system less susceptible to sabotage. By using a horizontal pump, the pump and its support systems are located nearer the plant and away from the ultimate heat sink (i.e., river, lake, oceans). An intake pipe is run from the pump to the ultimate heat sink via an existing intake structure. A priming system is provided to ensure that the suction line is filled with water before the service water pump starts. The service water pump is sized by the cooling water requirements of the reactor two hours after shutdown, the diesel engine heat load, and the ambient cooling water temperature. The system actuates on a two-out-of-three logic from dedicated instrumentation added onto the reactor vessel to sense low reactor coolant level. A time delay has been incorporated to assure that the system does not actuate needlessly.

4.3 Cost Estimates

For the two PWR concepts and one BWR concept discussed in Section 4.2 and described in detail in Appendices A, B, and C, cost estimates were developed for electrical and mechanical equipment, piping, instrumentation, civil/structural work, cabling, engineering services, and replacement power costs. The results of these cost estimates are summarized in Table 8 and are based on fourth quarter 1981 dollars. Table 9 gives a more detailed comparison of the construction costs, while Table 10 lists some of the major equipment and characteristics of the three alternative concepts.

The cost estimates shown in Tables 8 through 10 are based on the design information contained in Appendices A, B, and C. Piping layouts used for the estimates were approximated in accordance with the flow diagrams and the site layouts. Material costs were based on R.S. Means, Richardson Estimating Manual, and Burns and Roe, Inc., "in-house" cost data. Special items of material (such as diesel generators, pumps, stainless steel pipes and fittings, and major valves) were solicited from vendor sources having familiarity with the product involved. Labor manhours were developed for each task based on the estimators' experience and judgment. Unit rates in R.S. Means and the Richardson Manual were also used to establish labor manhours. Craft labor rates were taken from U.S. Department of Labor Handbook of Wages and Benefits cost index dated January 1981, adjusted to reflect rates effective fourth quarter 1981. Composite crew rates were developed utilizing these

Table 8

Comparison of Selected Concepts
Total Costs

COST	Add-on HPI (Feed & Bleed) with Small LOCA Makeup Capability			Add-on AFW Train With Small LOCA Makeup Capability			Add-on Low Pressure Makeup and Suppression Pool Cooling Train		
	3410MW-CE 1150 MW(e)	2772MW-B&W 880 MW(e)	1876MW-W 620 MW(e)	3410MW-CE 1150 MW(e)	2772MW-B&W 880 MW(e)	1876MW-W 620 MW(e)	3462MW-GE 1100 MW(e)	2381MW-GE 778 MW(e)	1930MW-GE 620 MW(e)
Construction	18,200,000	17,300,000	18,500,000	18,600,000	17,700,000	20,100,000	14,000,000	14,800,000	17,000,000
Other, Including Escalation, Interest During Construction, Engineering Services, Construction Management and Owner's Costs - Note 1	8,200,000	7,800,000	8,300,000	8,400,000	8,000,000	9,100,000	6,300,000	6,600,000	7,600,000
Subtotal	26,400,000	25,100,000	26,800,000	27,000,000	25,700,000	29,200,000	20,300,000	21,400,000	24,600,000
Outage Replacement Energy - Note 2	34,800,000	53,200,000	12,500,000	11,600,000	62,100,000	12,500,000	22,200,000	15,700,000	12,500,000
Total	61,200,000	78,300,000	39,300,000	38,600,000	87,800,000	41,700,000	42,500,000	37,100,000	37,100,000

NOTES:

1. These costs are taken as 45 percent of construction costs. This is only a rough estimate used for purposes of comparison and will depend to a large extent on the construction schedule and other factors involved.

2. Replacement power cost is taken as $720,000/day-1000 MW(e) and can vary widely with the utility involved, plant location and other factors. The possibility of doing all or part of the ADHRS tie-in during a normal scheduled outage (i.e., annual refueling) could significantly reduce or eliminate these costs.

Table 9

Comparison of Detailed Cost of the Alternative Concepts
(Installed Contractor Cost in Fourth Quarter 1981 Dollars)

	Add-on HPI (Feed & Bleed) with Small LOCA Makeup Capability			Add-on AFW Train With Small LOCA Makeup Capability			Add-on Low Pressure Makeup and Suppression Pool Cooling Train		
Item	3410MW-CE	2772MW-B&W	1876MW-W	3410MW-CE	2772MW-B&W	1876MW-W	3463MW-GE	2381MW-GE	1930MW-GE
Site Civil Work	7,677,000	7,188,000	8,753,000	8,462,000	7,922,000	9,649,000	4,328,000	4,518,000	5,170,000
Piping & Mech.	5,775,000	5,658,000	4,925,000	5,172,000	5,027,000	5,113,000	5,887,000	6,230,000	7,300,000
Equipment									
Electrical	308,000	308,000	308,000	326,000	326,000	326,000	274,000	274,000	274,000
Instrumentation	834,000	730,000	808,000	961,000	895,000	1,012,000	753,000	789,000	834,000
Subtotal	14,594,000	13,884,000	14,794,000	14,921,000	14,170,000	16,100,000	11,242,000	11,811,000	13,578,000
Contingency	3,649,000	3,471,000	3,699,000	3,730,000	3,543,000	4,025,000	2,811,000	2,953,000	3,395,000
Total Cost	18,243,000	17,355,000	18,493,000	18,651,000	17,713,000	20,125,000	14,053,000	14,764,000	16,973,000
Manhours	233,760	233,980	232,150	258,400	257,764	257,220	150,620	150,110	152,660

Table 10

Comparison of Costs, Plant Outage Times, Equipment, and Interfaces of the Alternative Concepts

	Add-on HPI (Feed & Bleed) with Small LOCA Makeup Capability			Add-on AFW Train With Small LOCA Makeup Capability			Add-on Low Pressure Makeup and Suppression Pool Cooling Train		
	3410MW-CE	2772MW-B&W	1876MW-W	3410MW-CE	2772MW-B&W	1876MW-W	3462MW-GE	2381MW-GE	1930MW-GE
Construction:									
Cost ($)	18,243,000	17,355,000	18,493,000	18,651,000	17,713,000	20,125,000	14,053,000	14,764,000	16,973,000
Duration	24 Mos	27 Mos	23 Mos	26 Mos	29 Mos	26 Mos	16 Mos	16 Mos	16 Mos
Outage	6 Wks	12 Wks	4 Wks	2 Wks	14 Wks	4 Wks	4 Wks	4 Wks	4 Wks
Equipment:									
Decay Heat Removal Pump	560 gpm @ 4150 ft 900 HP	485 gpm @ 4625 ft 1250 HP	300 gpm @ 4625 ft 700 HP	700 gpm @ 2641 ft 800 HP	940 gpm @ 2600 ft 1000 HP	800 gpm @ 2600 ft 900 HP	2100 gpm @ 450 ft 300 HP	1600 gpm @ 450 ft 250 HP	1450 gpm @ 810 ft 450 HP
Makeup/Service Water Pump	N/A	N/A	N/A	310 gpm @ 4625 ft 600 HP	310 gpm @ 4625 ft 600 HP	100 gpm @ 4625 ft 250 HP	10500 gpm@ 350 ft 800 HP	7500 gpm @ 350 ft 600 HP	6100 gpm @ 350 ft 450 HP
DHR Tank or Heat X/er. (gal.)	350,000	350,000	250,000	200,000	200,000	120,000	140×10^6 Btu/hr	100×10^6 Btu/hr	80×10^6 Btu/hr
Makeup/Service Water Source	N/A	N/A	N/A	120,000 gal Tank	130,000 gal Tank	120,000 gal Tank	Existing Spray Pond	River	Tidal Estuary Canal
Boron Conc.	≥4400 ppm	≥2270 ppm	≥4000 ppm	≥4400 ppm	≥2270 ppm	≥4000 ppm	N/A	N/A	N/A
Diesel Generator	1700 KW	2000 KW	1400 KW	2100 KW	2300 KW	1900 KW	1400 KW	1200 KW	1200 KW
Tie-ins Inside Cont:									
Major	1	3	2	-	3	1	-	-	-
Instrumentation	8	8	8	16	16	16	-	-	-

rates. Equipment rental/operational rates were taken from R.S. Means-1981 edition. Indirect costs reflect standard general contractor markups. A contingency of 25 percent has been included to cover uncertainties.

Although no detailed construction schedule was developed for this study, outage times for construction were estimated on the basis of the scope of major system interface work involved, using an average work force of 60 people working 40-hour weeks. As can be seen from Tables 8 and 10, the estimated cost of replacement power during a construction outage can represent the major portion of the cost of a concept. In principle, this economic impact could be minimized by performing the required construction activities during a normal outage. However, scheduled outages are normally peak maintenance activity periods in which most regulatory mandated modifications and high radiation area repairs are performed. It is unrealistic to expect that all of this work, plus refueling activities and installation of an add-on system, could be performed within the duration of a typical scheduled outage.

By comparing the required outage time for the Combustion Engineering (CE) plant to the forced outage time of the Babcock and Wilcox (B&W) plant, it can be seen that the outage time and hence the cost of making all tie-ins inside containment can be quite large. Because of the existing piping layouts, it was decided to make some of the tie-ins for the CE plant in the auxiliary building and all the tie-ins for the B&W plant inside the containment building.

Contrary to what was expected, the site-to-site differences, for the power plants considered, were found to have a minimal effect on the estimated costs of the alternative concepts. Once the requirement for seismic design was established, only minor design changes were needed to accommodate the different safe shutdown earthquake, ground accelerations, windloadings, and flood levels of the six sites. Piping runs and excavation work were minimized for each site, and the add-on buildings for the alternative decay heat removal trains were separated from existing plant systems in accordance with the special emergency design criteria discussed in Section 2.3.2.

4.4 Potential Impact of Design Criteria Changes

During development of the two PWR concepts and one BWR concept, additional design criteria were identified which could influence the feasibility, impact, and value of the concepts. These additional criteria, which go beyond those established in Section 2.3, call for the alternative decay heat removal systems to operate beyond 10 hours, to achieve cold shutdown conditions, and to provide reactor shutdown capability through boration. Accident sequences which are related to these criteria include seismically-induced loss of steam generator water supplies, PWR steam generator tube failure, and boron dilution accidents. However, because safety importance of these and similar accident sequences has not been quantitatively analyzed as yet, an investigation of the criteria was limited to a qualitative assessment of their potential impact.

Table 11 summarizes the potential impact that each new criterion could have on the two PWR and one BWR alternative decay heat removal concepts. For extending operation beyond 10 hours, the add-on HPI train is expected to have the greatest difficulty, primarily because of a buildup of water and heat inside containment. If this criterion was adopted, many of the benefits of the add-on HPI concept would be lost. For achieving cold shutdown, the add-on AFW train proves to be deficient. As presently visualized, the add-on AFW concept cannot reach cold shutdown without the addition of a vacuum boiling capability on the steam generators or resorting to a feed-and-bleed cooling mode similar to the add-on HPI train. For providing reactor shutdown capability through boration, all three concepts seem to be comparable, although a quantitative design effort would be needed to size the boron addition system and establish boron concentration requirements.

Table 11

Effects of Criteria Changes

Criteria Change	Add-on HPI (Feed & Bleed) with Small LOCA Makeup Capability	Add-on AFW Train With Small LOCA Makeup Capability	Add-on Low Pressure Makeup and Suppression Pool Cooling Train
Extended Operation > 10 hrs.	1. Requires larger tankage or alternate sources of water and fuel oil. 2. Requires containment cooling and flood control. 3. Seriously reduces attractiveness of concept.	1. Requires larger tankage or alternate sources of water and fuel oil. 2. Since thermal driving head will reduce, reactor coolant natural circulation flow will become intermittent. 3. Minor impact on concept.	1. Requires larger fuel oil tank. (For plant #4, 3462MW(t) may require operator action to refill spray pond.) 2. Minor impact on concept.
Achieve Cold Shutdown	1. No modification to components, need new instrumentation logic only. 2. Minor impact on concept.	1. Requires a vacuum system or must revert to feed-and-bleed cooling using HPI system. 2. Seriously reduces the attractiveness of concept.	1. No modification to components, need new instrumentation logic only. 2. Minor impact on concept.
Borated Shutdown	1. Needs Boron injection system. 2. Minor impact on concept.	1. Needs Boron injection system. 2. Minor impact on concept.	1. Needs Boron injection system. 2. Minor impact on concept.

5.0 Value Assessment

In order to evaluate the potential safety value of the alternative PWR and BWR decay heat removal concepts, two existing probabilistic risk assessments (PRAs) were selected as base cases for analysis. For PWRs the Oconee #3 PRA[4] was used, and for BWRs the Grand Gulf #1 PRA[5] was adopted. To permit an extension of the analysis to a variety of plant configurations other than Oconee and Grand Gulf, the base case PRAs were simplified to express safety system unavailabilities in terms of major component and subsystem failures, rather than in terms of minor component unavailabilities. For a safety function, such as providing feedwater to steam generators for decay heat removal, the simplified PRA analysis involved regrouping component failures to reflect only the unavailability of the flow paths, pumps, and actuation systems, without explicitly listing system details, such as valves, instrumentation, and minor component maintenance activities. Appendices D and E present the Boolean equations for Oconee and Grand Gulf which resulted from this regrouping effort.

By using the regrouping analysis technique, it was assumed that differences between the original and simplified PRA logic models did not significantly alter the Boolean analysis results of the original PRA. Also, it was assumed that the modified logic models contained sufficient detail to permit further logic modifications to approximate other similar power plant configurations. Both of these assumptions proved to be valid. A favorable comparison was made of the core melt accident sequence frequencies published for Oconee and Grand Gulf,[4,5] with the frequencies calculated using the regrouped expressions from Appendices D and E. In addition, it was found that the regrouped logic models developed for Oconee could be modified slightly and reanalyzed to agree with the detailed PRA analysis conducted for the Crystal River #3 power plant.[7] This ability to modify one power plant PRA to correspond to that of another power plant permits the value of the alternative decay heat removal systems to be assessed relative to several power plants having differing degrees of in-place decay heat removal reliability.

5.1 Value Assessment Technique

Postulated core melt accident sequences reflect combinations of system, subsystem, or component failures that occur as a result of various initiating events, such as loss of electrical power, loss of feedwater, or loss of coolant accidents. PRA analyses attempt to identify those accident sequences which have the highest expected frequency of occurrence and which, when added together, constitute substantially the total expected core melt frequency for a particular power plant. By improving decay heat removal system reliability, the expected occurrence frequency of core melt accident sequences involving decay heat removal failure can be reduced. However, depending upon the extent of the improvement and the magnitude of the affected accident sequences, it is possible for an alternative system to have either a significant or an insignificant effect on plant safety. Recognizing this, it was decided to assess the value of the candidate alternative decay heat removal concepts in the following four steps:

1. The Oconee #3 and Grand Gulf #1 probabilistic risk assessments[4,5] were reperformed, using the regrouping technique discussed above, to represent five different base-case power plants (see Appendices F and G).

2. For the five base-case power plants, a sensitivity analysis was carried out for a variety of limiting decay heat removal and nondecay heat removal system improvements to establish the maximum extent which core melt frequency can be reduced using various add-on systems having hypothetical failure probabilities of zero (see Appendices H and I).

3. For the base-case power plants, core melt frequency reductions were estimated for the two PWR concepts and one BWR concept whose impacts were reported in Section 4.0 (see Appendices H, I, and J).

4. For the base-case power plants, reductions in the estimated frequency of radioactive material releases were estimated for the two PWR concepts and one BWR concept whose impacts were reported in Section 4.0 (see Appendices K, L, and M).

5.2 Base-Case Plant Configurations

5.2.1 PWR Configurations

Three different PWR power plant configurations were selected for analysis, starting with the Oconee #3 plant. For each successive configuration, certain design features of the Oconee plant were modified to reflect lower decay heat removal reliabilities typical of other power plants. The three PWR cases considered were:

1. The Oconee #3 power plant having three AC-dependent auxiliary feedwater trains, one additional independent auxiliary feedwater train, on-site AC hydro power generation, and the capability to remove decay heat via feed-and-bleed operation of the HPI system.

2. A modified version of the Oconee #3 power plant having two auxiliary feedwater trains instead of four trains, two diesel generation electrical trains instead of hydro power generation, and the capability to remove decay heat via feed-and-bleed.

3. A further modified version of the Oconee #3 power plant having two auxiliary feedwater trains, two diesel generation electrical trains, and no feed-and-bleed capability.

For these three plant configurations, the base-case frequencies of core melt were estimated using regrouped and modified versions of the Oconee #3 PRA,[4] as described in Section 5.0 and Appendices D and F. For the three plant configurations outlined above, the estimated core melt probabilities per reactor year were estimated to be 7.9×10^{-5}, 1.3×10^{-4}, and 2.6×10^{-3}, respectively.

5.2.2 BWR Configurations

Two different BWR power plant configurations were selected for analysis, starting with the Grand Gulf #1 plant. The two cases considered were:

1. The Grand Gulf #1 power plant having an estimated reactor protection logic system unavailability of about 10^{-5}/demand.

2. A modified version of the Grand Gulf #3 power plant having an estimated reactor protection logic system unavailability of about 10^{-7}/demand.*

For these two plant configurations, the base-case frequencies of core melt were estimated using regrouped and modified versions of the Grand Gulf #1 PRA,[5] as described in Section 5.0 and Appendices E and G. For the two plant configurations outlined above, the estimated core melt probabilities per reactor year were estimated to be 2.8×10^{-5} and 2.3×10^{-5}, respectively.

5.3 Value Sensitivity Analysis of Limiting Cases

In order to estimate the maximum possible improvements in overall safety that can be derived from various types of alternative decay heat removal and nondecay heat removal concepts, a sensitivity study was performed for hypothetically "perfect" add-on concepts being applied to each of the base-case power plant configurations. For this analysis, a "perfect" add-on concept was visualized as a concept which can perform its function with zero probability of failure. Of course, a fail-safe concept of this type is unattainable in practice; however, by evaluating the

*General Electric has stated an ability to achieve a reactor protection logic system unavailability of about 10^{-6}/year. This corresponds to an unavailability of 10^{-7}/demand given that there are about seven transients/year to challenge the system. (Reference 23) safety value of a perfect system, bounds can be set for real system improvements. If a theoretically fail-safe add-on concept reduces a power plant's overall estimated core melt frequency by only a small increment, then there would be little incentive to consider what safety improvements a less reliable, real add-on system could provide.

For the PWR and BWR base-case power plants discussed in Section 5.2, Appendices H and I describe those core melt accident sequences whose probability is reduced by the installation of various hypothetically "perfect" add-on concepts, including fail-safe emergency power, LOCA injection, and auxiliary feedwater trains for PWRs and residual heat removal and low pressure injection trains for BWRs. Tables 12 and 13 summarize these findings.

Four major observations can be made from Tables 12 and 13. First, it appears that for both the original Oconee #3 base case and the reconfigured Oconee plant having feed-and-bleed capability, only small reductions in overall estimated core melt frequency can be achieved through numerous different improvements in safety systems. Second, for a PWR power plant without feed-and-bleed capability, the importance of having a reliable feedwater system is accentuated. Third, for BWRs similar to Grand Gulf, significant improvements in safety can be achieved only after RPS reliability has been improved. Fourth, BWRs similar to Grand Gulf already have an estimated core melt frequency below that estimated for PWRs similar to Oconee.

These findings indicate that if the core melt frequency of a power plant is fairly small (i.e., less than 10^{-4} per reactor year), no one safety system vulnerability can be identified as an overwhelming contributor to core melt. Because of this, no one safety system improvement can significantly reduce the overall estimated core melt frequency. The PWR fail-safe options considered in Table 12 essentially eliminate many of the transient and small LOCA accident sequences which can lead to core melt, but these options do little to reduce the estimated frequency of large LOCA accident sequences or, except for the feed-and-bleed concept, failure to scram (i.e., ATWS) accident sequences. Similarly, the BWR fail-safe options considered in Table 13 essentially eliminate many of the transient, small LOCA, and ATWS accident sequences which can lead to core melt, but those options do little to reduce the frequency of large LOCA accident sequences.

Of course, it may be possible for an add-on concept to significantly reduce the public risk of a power plant, without greatly reducing the estimated core melt frequency. This could occur by the elimination of a select group of core melt accident sequences whose radiological consequences are large. Then the factors of safety improvement based on core melt could be lower than factors of improvement based on radioactive material release

Table 12
Changes in Estimated Core Melt Probability Per Reactor Year for Various PWR Fail-Safe Add-on Safety Systems

		With Zero Unavailability for the:				
	Initial	Feedwater System	LOCA High Pressure Injection System	Feedwater and LOCA High Pressure Injection System	Emergency Power System	Feed and Bleed System
Oconee #3[b]	7.9×10^{-5}	5.6×10^{-5}	7.2×10^{-5}	5.1×10^{-5}	7.6×10^{-5}	4.3×10^{-5}
Factor of Improvement[a]		1.4	1.1	1.5	No Improvement	1.8
Reconfigured Oconee[c] (with Feed-and-Bleed Capability)	1.3×10^{-4}	5.3×10^{-5}	1.2×10^{-4}	5.1×10^{-5}	1.1×10^{-4}	4.4×10^{-5}
Factor of Improvement		2.4	1.1	2.5	1.2	2.9
Reconfigured Oconee[d] (without Feed-and-Bleed Capability)	2.6×10^{-3}	1.2×10^{-4}	2.6×10^{-3}	1.2×10^{-4}	2.6×10^{-3}	4.5×10^{-5}
Factor of Improvement		21.7	No Improvement	21.7	No Improvement	58.0

[a]Factor of Improvement = (Initial Estimated Core Melt Frequency)/(Estimated Core Melt Frequency With Fail-Safe Option)

[b]Oconee #3, PRA (Reference 4) expressed as system and subsystem failures.

[c]Oconee #3, Modified Oconee-3 plant to reflect: two trains of emergency feedwater vs. four trains for Oconee; two trains of diesel electrical power vs. hydro power generators for Oconee;

[d]Oconee #3, Modified Oconee #3 as in Footnote b but without HPI feed-and-bleed capability.

Table 13

Changes in Estimated Core Melt Probability Per Reactor Year for Various
BWR Fail-Safe Add-on Safety Systems

		With Zero Unavailability for the:		
	Initial	RPS	RHR Suppression Pool Cooling	RHR Suppression Pool Cooling and Low Pressure Injection
Grand Gulf #1 (RPS failure probability of 10^{-5} per demand	2.8×10^{-5}	2.3×10^{-5}	9.0×10^{-6}	7.5×10^{-6}
Factor of Improvement[a]		1.2	3.1	3.7
Reconfigured Grand Gulf (RPS failure probability of 10^{-7} per demand)	2.3×10^{-5}	2.3×10^{-5}	3.4×10^{-6}	2.0×10^{-6}
Factor of Improvement		No Improvement	6.8	11.5

[a]Factor of Improvement = (Initial Estimated Core Melt Frequency)/(Estimated Core Melt Frequency With Fail-Safe Option)

frequency estimates. However, as indicated in Section 5.5 for the power plants and add-on concepts considered in this study, the factors of improvement in release are typically lower than the factors of improvement in core melt frequency.

5.4 Value Sensitivity Analysis for Real Concepts

For the PWR add-on HPI and add-on AFW concepts and the BWR add-on low pressure makeup and suppression pool cooling concept, a sensitivity analysis was performed to estimate the potential value of these concepts for reducing estimated core melt frequency. This was done for the five base-case power plants treated in Section 5.3.

The approach taken to evaluate estimated core melt frequency reductions involved identifying those accident sequences where probability of occurrence is reduced by the installation of the add-on concepts. However, unlike the limiting value analysis discussed in Section 5.3 which considered only fail-safe add-on concepts, the value analysis in this section treats the three alternative decay heat removal concepts as real systems having finite failure probabilities.

5.4.1 Failure Probability Estimates for Alternative DHR Concepts

The concept development and impact evaluation effort reported in Section 4.0 defined two PWR concepts and one BWR concept which employed state-of-the-art equipment and design techniques to ensure reliable operation. On the basis of this design approach, reliability estimates can be made which treat the add-on concepts as independent, single-train systems, having reliabilities comparable to those estimated for similar systems in existing power plants. By using the component, maintenance, and human error failure probabilities provided in the Oconee #3 and Grand Gulf #1 PRA reports,[4,5] conservative estimates were made for the reliability of the add-on concepts. For the PWR concepts, the Oconee #3 numbers yielded an estimated failure probability per demand of 1.1×10^{-1} for the add-on HPI concept and 9.7×10^{-2} for the add-on AFW concept. Table 14 shows how well these numbers agree with system unavailability estimates reported in other PRAs. Based on this agreement, an unavailability per demand of 0.1 was selected for the add-on HPI and AFW concepts. This number includes an electric support system unavailability of 0.07 per demand; and therefore, for accident sequences having normal offsite electric power available, an add-on system unavailability of 0.03 per demand was used. Appendix H describes in more detail the basis for PWR add-on system unavailability estimates.

For the BWR concept, a similar single-train analysis of the Grand Gulf PRA produced three different unavailability estimates to reflect the different operating modes of the add-on BWR concept. For suppression pool cooling operations, an unavailability

Table 14

A Comparison of Estimated Unavailabilities for Single-Train Systems

AFW	Unavailability/Demand (Includes Electric Power Train)
Add-on Concept	9.7×10^{-2}
Sequoyah AFW Train[6]	8.3×10^{-2}
Zion AFW Train[24]	8.3×10^{-2}
Crystal River #3 AFW Train[7]	7.6×10^{-2}

HPI	Unavailability/Demand (Includes Electric Power Train)
Add-on Concept	1.1×10^{-1}
Sequoyah HPI Train	9.4×10^{-2}
Zion HPI Train	7.7×10^{-2}
Crystal River #3 HPI Train	1.2×10^{-1}

of 0.08 per demand was assessed for the add-on train; and for low pressure makeup, an unavailability of either 0.03 or 0.04 per demand was estimated, respectively, for accident sequences in which offsite electrical power is either available or unavailable.

5.4.2 Estimates of Core Melt Reductions for Alternative DHR Systems

In Appendices H and I, 256 PWR and 34 BWR accident sequences have been postulated to lead to core melt. A sensitivity analysis of these sequences identified the core melt accident sequences whose frequency can be reduced by the alternative DHR concepts discussed in the previous section. Tables 15 and 16 summarize the effects of the alternative concepts on overall core melt frequency estimates for the base-case power plants.

It can be seen from Table 15 that relatively small factors of safety improvement (i.e., 1.5 to 2.6) can be gained for PWRs which already have reliable auxiliary feedwater and feed-and-bleed decay heat removal capability. However, for a PWR without feed-and-bleed capability and only two installed trains of auxiliary feedwater, either an add-on feed-and-bleed train or an add-on auxiliary feedwater train can significantly reduce the estimated frequency of core melt. It is interesting to note that the difference between the estimated core melt frequencies for the second and third base case PWRs correspond to comparing plants with and without three trains of HPI feed-and-bleed capability. Physically, this could represent differences in relief valve set points or capacity or differences in HPI pump discharge head or capacity. Therefore the differing factor of improvement between the second and third base case plants can be viewed as the effective addition of three trains of feed-and-bleed. In contrast to the PWR findings, Table 16 shows that for the BWR configurations analyzed, larger reductions in core melt frequency occur only after improvements in reliability have been made to the reactor protection system.

5.4.3 Value of Alternatives for Special Emergencies

In the previous section, the safety value of three alternative decay heat removal concepts was estimated using PRA techniques to quantify the estimated frequency of core melt for several different base-case power plant configurations. For the power plants considered, it was assumed that the existing decay heat removal systems have been properly designed to withstand all types of special emergency situations which may threaten the plants. As a result, the core melt frequency estimates in Section 5.4.2 reflect only accident sequences which may be expected to occur, such as: loss of offsite power; loss of feedwater; LOCAs; and loss of onsite and offsite power. The contribution of special emergencies to overall estimated core melt frequency was neglected in the Oconee #3 and

Table 15

Estimated Reductions in Core Melt Probability per Reactor Year Associated With Add-on Alternative PWR Decay Heat Removal Concepts

PWR	Before Adding Alternative	With Add-on Feed-and-Bleed Alternative	With Add-on Auxiliary Feedwater and Makeup Alternative
Oconee[a]	7.9×10^{-5}	4.6×10^{-5}	5.3×10^{-5}
Factor of Improvement		1.7	1.5
Reconfigured[b] Oconee (with feed-and-bleed capability)	1.3×10^{-4}	5.0×10^{-5}	5.6×10^{-5}
Factor of Improvement		2.6	2.3
Reconfigured[c] Oconee (without feed-and-bleed capability)	2.6×10^{-3}	1.9×10^{-4}	2.6×10^{-4}
Factor of Improvement		13.7	10.0

[a]Oconee #3 - As expressed in Reference 4.

[b]Oconee #3 - Reconfigured to be similar to Crystal River #3:
-- two trains of emergency feedwater vs. four trains for Oconee;
-- two trains of diesel emergency power vs. hydrogenerators for Oconee.

[c]Oconee #3 - Reconfigured to be similar to Crystal River #3 but without feed-and-bleed capability.

Table 16

Estimated Reductions in Core Melt Probability per Reactor Year Associated With an Add-on Alternative BWR Decay Heat Removal Concept

BWR	Before Adding Alternative	With Add-on Low Pressure Makeup and Suppression Pool Cooling Alternative
Grand Gulf[a] (with RPS failure probability of 10^{-5} per demand)	2.8×10^{-5}	9.6×10^{-6}
Factor of Improvement		2.9
Reconfigured Grand Gulf[b] (with RPS failure probability of 10^{-7} per demand)	2.3×10^{-5}	4.2×10^{-6}
Factor of Improvement		5.5

[a]Grand Gulf #1 - As expressed in Reference 4.

[b]Grand Gulf #1 - Reconfigured to reflect a higher reliability RPS.

Grand Gulf #1 PRAs. However, recent evidence for other power plants indicates some special emergencies (e.g., fire and earthquake) can dominate the estimated core melt frequency of a power plant.[24,25] Unfortunately, the extent to which a power plant's decay heat removal systems can be jeopardized by a particular special emergency depends upon numerous site specific and plant specific conditions which are not easily quantified in magnitude or frequency of occurrence. Because of this uncertainty, the alternative add-on concepts developed in the present study were designed in combination with existing plant systems to handle all the special emergencies dictated by current design guidelines, including fire, flood, earthquake, sabotage, and airplane crash. Of course, the safety value of the add-on concepts meeting all special emergency conditions will vary from one power plant to another.

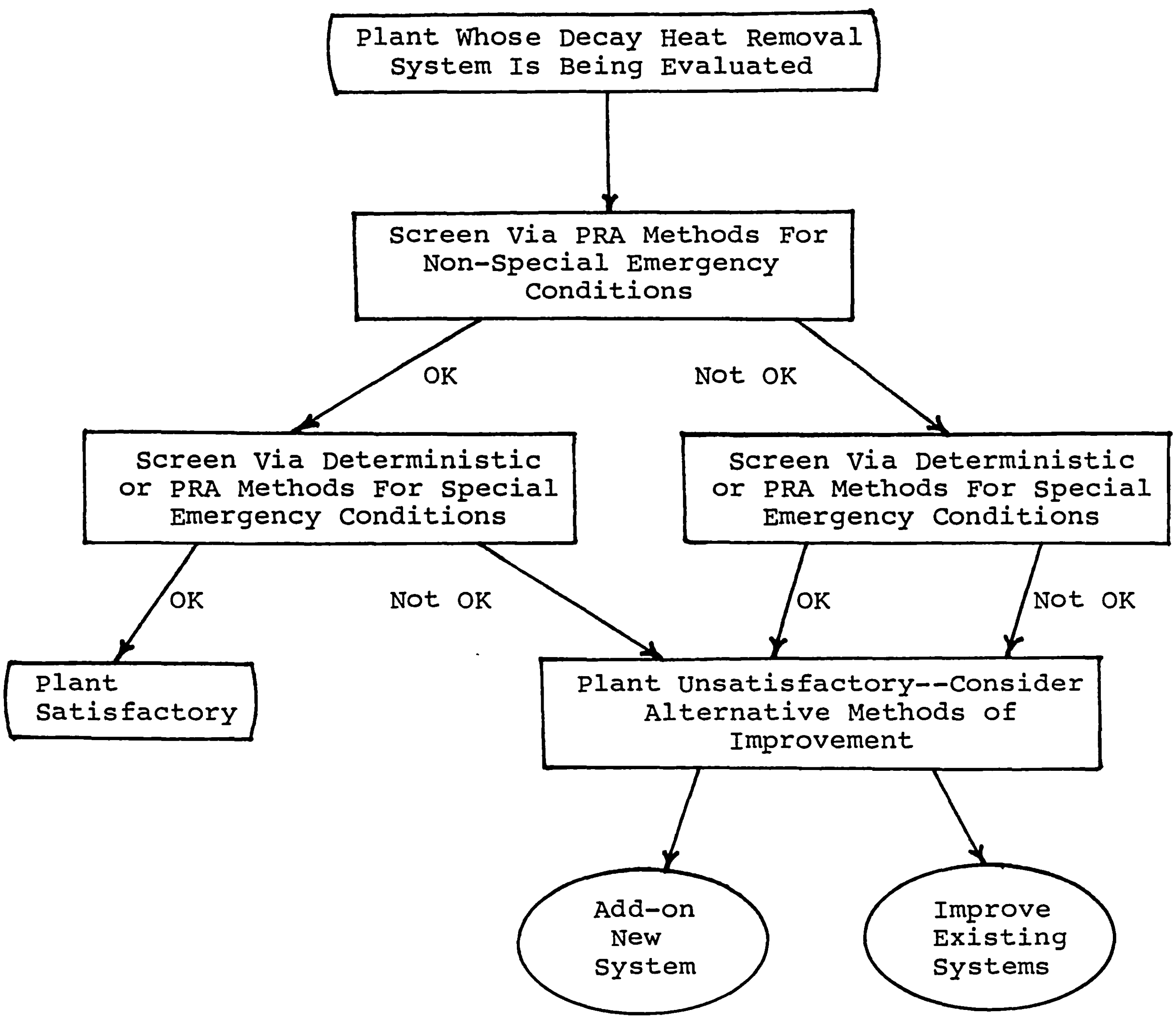
If a power plant can be shown capable of withstanding most special emergency conditions, it would be illogical to consider an add-on system which either duplicates the plant's existing capabilities or does not address those special emergencies to which the plant is vulnerable. Similarly, it would be illogical to install a completely new add-on decay heat removal train to cope with a particular special emergency if the existing plant systems could be made invulnerable to the emergency condition (e.g., adding fire barriers or adding missile shields).

One way to evaluate whether a power plant needs an add-on system or simply an improvement to an existing system is shown in Figure 11. Following the logic in Figure 11, a power plant would undergo an initial evaluation of its core melt frequency by applying PRA techniques similar to those used in Section 5.4.2. Following this evaluation, a second evaluation would be done for special emergency conditions using either deterministic or probabilistic techniques. If the results of both evaluations proved satisfactory, then no improvements in decay heat removal capability would be warranted. However, if the PRA or the special emergency evaluations or both revealed decay heat removal system weaknesses, then alternative methods of decay heat removal improvements would be considered. These improvements can range from relatively minor fixes of existing systems to the installation of an add-on system.

To illustrate how the value of an add-on decay heat removal system compares with the value of improving an existing system for special emergencies, an analysis was performed for three different fire scenarios. The fire scenarios considered involved the auxiliary feedwater system, the high pressure injection system, and the emergency power system of the second base case PWR (Crystal River #3) described in Section 5.2.1. Fire was chosen for analysis primarily because of an available statistical base and recent information which shows fire to be a possible large contributor to overall core melt frequency.[25] Because an assessment of special

Figure 11

Decision Logic Diagram for Evaluating and Improving a Power Plant's Decay Heat Removal Systems



emergency vulnerabilities depends on many plant specific design features, the fire analysis results to be presented here are not strictly applicable to Crystal River, nor are they generic or comprehensive. Other fire scenarios or other special emergencies (e.g., earthquake and sabotage) would need to be analyzed before the special emergency screening would be complete.

Appendix J presents the fire analysis technique used to estimate the frequency with which fires may be expected to occur in certain power plant areas and the probability that these fires will go undetected, unsuppressed, and uncontained, eventually spreading to other plant areas. For the three fire scenarios considered, Appendix J presents the following estimates for the core melt frequency associated with each fire for the Crystal River #3 simulated power plant:

	Associated Core Melt Frequency
Auxiliary Building Fire Which Damages Both Auxiliary Feedwater Pumps	1.8×10^{-8}/reactor-year
Auxiliary Building Fire Which Damages All Three High Pressure Injection Pumps	5.8×10^{-9}/reactor-year
Control Building Fire Which Damages Both Trains of Emergency Power	1.0×10^{-4}/reactor-year

It can be seen from these estimates that, in order for fire to be a significant contributor to core melt frequency, the fire must damage more than one safety system. The electrical fire scenario effectively does this, considering that the nonfire accident scenarios estimated for the Crystal River #3 plant add up to only 1.3×10^{-4}/reactor-year (Table 15). Of course, it should be remembered that all three fire scenarios evaluated here are hypothetical and do not necessarily reflect specific design features of Crystal River #3.

For the most severe fire scenario involving redundant electrical trains, an analysis was done to compare the relative safety value of the two PWR alternative decay heat removal concepts. In addition, consideration was given to the effectiveness of eliminating the fire vulnerability by the use of an independent electrical power train or additional fire protection measures (i.e., fire barriers). Table 17 summarizes the results of this analysis.

Table 17

Comparison of the Safety Value of Add-on Decay Heat Removal Systems With Improvements to Existing Systems for a Fire Special Emergency

	Estimated Core Melt Frequency per Reactor-Year				
	Nonfire Accident Sequences	+	Fire Accident Sequences	=	Total
Simulated Crystal River #3 Base Case	1.3×10^{-4}	+	--	=	1.3×10^{-4}
Base Case With Fire Disabling Redundant Electrical Power Trains	1.3×10^{-4}	+	1.0×10^{-4}	=	2.3×10^{-4}
Base Case With Fire Scenario and the Add-on Feed-and-Bleed Alternative System	0.5×10^{-4}	+	0.1×10^{-4}	=	0.6×10^{-4}
Base Case With Fire Scenario and the Add-on Auxiliary Feedwater System	0.6×10^{-4}	+	0.1×10^{-4}	=	0.7×10^{-4}
Base Case With Fire Scenario and an Add-on Electrical Power Train	1.1×10^{-4}	+	0.7×10^{-5}	=	1.2×10^{-4}
Base Case With Fire Scenario and Added Fire Barriers	1.3×10^{-4}	+	0.1×10^{-4}	=	1.4×10^{-4}

It can be seen from Table 17 that all four of the alternative methods of coping with the fire problem offer some degree of safety value. In fact, the installation of fire barriers alone can be expected to essentially eliminate the fire problems. However, the installation of either an add-on feed-and-bleed system or an add-on auxiliary feedwater system results in a larger predicted reduction in core melt frequency than the use of fire barriers, because the add-on decay heat removal systems cope with both fire and nonfire accident sequences. However, although these add-on decay heat removal systems provide larger reductions in core melt frequency than the installation of a fire barrier, it is doubtful whether the factor of two difference between the improvement schemes would warrant the greater cost impact of an add-on decay heat removal train. Unless an alternative add-on decay heat removal system could be justified on the basis of other special emergency vulnerabilities, it would be logical to address this particular fire problem by "improving the existing systems" through the use of fire barriers, not by installing a completely new add-on system. Therefore, a decision to add on a completely new system or to improve an existing system must be made on a case-by-case basis for any power plant or special emergency being evaluated.

5.5 Reductions in Estimated Radioactive Material Release Frequencies

In addition to the core melt reduction value sensitivity analysis reported in Section 5.4, an analysis was conducted of the extent to which the two PWR and one BWR alternative decay heat removal concepts reduce the frequency of radioactive material releases for the release categories defined in the Reactor Safety Study[3]. This was done to compare risk reduction to core melt reduction for the alternative DHR concepts.

For the PWR and BWR base-case power plants which were discussed in Section 5.2 and used for the core melt sensitivity analysis, Appendices K and L describe the radioactive material release accident sequences which have been identified for Oconee #3 and Grand Gulf #1. Using the estimated frequency of radioactive material release for each of these base cases as a starting point, a sensitivity analysis was performed for hypothetical fail-safe and real add-on decay heat removal and nondecay heat removal concepts. Appendix M presents the results of this analysis.

For the fail-safe alternative concepts considered, it was found that little reduction in release frequency could be achieved for the Oconee #3 and Crystal River #3 simulated base cases (see Tables M-1 and M-2)--a finding consistent with the core melt sensitivity results presented in Section 5.3. Because of this, consideration of the release reduction value of real add-on PWR concepts was limited to the third PWR base-case plant which has no feed-and-bleed capability. In contrast, the release reduction value of real BWR add-on concepts was estimated for both of the base-case BWR plants. Tables 18 and 19 summarize the results of the radioactive

Table 18

Estimated Reductions in Release Probability Per Reactor-Year Associated With Add-on Alternative PWR Decay Heat Removal Concepts for a Crystal River Simulated Plant Without Feed-and-Bleed Capability

Release Category	Before Adding Alternative	With Add-on Feed-and-Bleed Alternative	With Add-on Auxiliary Feedwater Alternative
1	4.4×10^{-6}	4.8×10^{-7} (9.2)*	4.8×10^{-7} (9.2)*
2	1.1×10^{-5}	5.2×10^{-6} (2.1)*	5.2×10^{-6} (2.1)*
3	1.2×10^{-3}	6.9×10^{-5} (17.4)*	1.1×10^{-4} (10.9)*
4	1.7×10^{-7}	7.9×10^{-8} (2.2)*	7.9×10^{-8} (2.2)*
5	1.7×10^{-5}	1.0×10^{-6} (17)*	1.6×10^{-6} (10.6)*
6	1.2×10^{-5}	6.3×10^{-6} (1.9)*	6.3×10^{-6} (1.9)*
7	1.2×10^{-3}	6.9×10^{-5} (17.4)*	1.1×10^{-4} (10.9)*

* Factor of Improvement = (Release Probability Before Adding Alternative)/(Release Probability After Adding Alternative)

Table 19

Estimated Reductions in Release Probability Per Reactor-Year Associated With an Add-on Alternative BWR Decay Heat Removal Concept

Grand Gulf (with RPS failure probability of 10^{-5} per demand) Release Category	Before Adding Alternative	With Add-on Low Pressure Makeup and Suppression Pool Cooling Alternative
1	8.4×10^{-8}	2.5×10^{-8} (3.4)*
2	2.5×10^{-5}	8.1×10^{-6} (3.1)
3	1.4×10^{-6}	6.8×10^{-7} (2.1)
4	1.4×10^{-6}	6.8×10^{-7} (2.1)

Grand Gulf (with RPS failure probability of 10^{-7} per demand) Release Category	Before Adding Alternative	With Add-on Low Pressure Makeup and Suppression Pool Cooling Alternative
1	8.4×10^{-8}	2.5×10^{-8} (3.4)
2	2.0×10^{-5}	2.8×10^{-6} (7.1)
3	1.4×10^{-6}	6.8×10^{-7} (2.1)
4	1.4×10^{-6}	6.8×10^{-7} (2.1)

*Factor of Improvement = (Release Probability Before Adding Alternative)/(Relase Probability After Adding Alternative) material release frequency analysis for the PWR and BWR alternative add-on decay heat removal concepts. Appendix M presents comparable analysis results for fail-safe versions of these concepts and for other fail-safe and real concepts capable of handling decay heat removal and nondecay heat removal system failures.

By comparing the factors of improvement shown in Tables 18 and 19 for release frequency reductions with the factors of improvement shown in Tables 15 and 16 for core melt frequency reductions, two major observations can be made. First, safety improvement factors based on core melt frequencies generally equal or exceed improvement factors based on release frequency estimates. Second, improvement factors in some release categories are much smaller than estimates of overall core melt reduction. Therefore, for the plants and add-on concepts analyzed, the potential risk reduction of an add-on decay heat removal concept can be less than the potential for reducing the estimated core melt frequency. If this is true for other power plants, an estimate of core melt frequency reduction could be used to screen alternative DHR concepts, because a concept which insignificantly reduces the estimated frequency of core melt could not be assumed to significantly reduce radioactive material release frequency. On the other hand, an add-on decay heat removal concept may be attractive if it significantly reduces estimated core melt frequency, even if the frequency of some radioactive material releases is unaffected.

6.0 Findings and Conclusions

This report has documented numerous observations regarding the current use of decay heat removal systems, the feasibility of alternative DHR concepts, the potential impact of several alternative systems, and the potential value of these alternatives. This section restates some of the more important findings and conclusions of the study.

6.1 The Current Use of Decay Heat Removal Systems

Generally, the same design criteria and design approaches for decay heat removal systems have been adopted in the U.S. and abroad. However, a new European design philosophy is now being followed which calls for sustaining both a random failure of one safety train and a simultaneous maintenance outage of another train, while still retaining 100 percent operational capacity. This approach, known as N+2 redundancy, was prompted by a concern over special emergencies (e.g., airplane crash) and goes beyond U.S. practice of safety system single-failure capability.

- Much effort in some countries has been placed on providing three, four, and even six trains of PWR residual heat removal system cooling, even though no dominant core melt accident scenarios have been attributed to failure of the residual heat removal system.
- Add-on PWR DHR systems generally provide inadequate reactor coolant makeup to cope with small break LOCA accident sequences.
- Some non-U.S. PWRs have installed as many as six trains of auxiliary feedwater (i.e., four 50-percent trains plus two 100-percent trains), despite insignificant potential safety improvements of using more than three 100-percent trains.
- Some non-U.S. BWRs have installed as many as four trains of residual heat removal and low pressure injection, despite the fact that more than three 100-percent trains appear to provide insignificant potential safety improvements.

6.2 The Feasibility of Alternative DHR Concepts

- DHR alternatives which use relatively small, proven components and require few interfacing system tie-ins appear to be the most feasible alternatives for retrofit installation.

- Alternative DHR concepts which rely on sensible heating of water to remove high-level initial decay heat loads lack feasibility for retrofitting, because of the large sizes required for piping, pumps, heat exchangers, and containment penetrations. Most closed-loop PWR and BWR concepts fall into this category.

- The feasibility of performing relatively small modifications to existing DHR systems must be determined on a plant-by-plant basis, because unique plant features often dictate feasibility. Plant-specific assessments of this type were not treated in this study.

- For add-on DHR concepts which were engineered to be completely independent of existing DHR systems, a PWR auxiliary feedwater train, a PWR high pressure injection train and a BWR suppression pool cooling and low pressure injection train were found to be more feasible for retrofit than six other candidate add-on concepts.

6.3 The Potential Impact of Several Alternative Systems

- The engineering, design, interest, management, and construction costs of an add-on auxiliary feedwater or high pressure injection DHR train for PWRs or a suppression pool cooling/low pressure injection DHR train for BWRs was estimated to range between $20 and $30 million.

- The plant outage time required for establishment of an add-on concept was estimated to range from 2 to 14 weeks, depending on the extent of add-on system interface tie-ins to be made inside the containment building. This outage time corresponds to estimated power replacements costs of $13 to $62 million for the six power plants analyzed.

- The impact of site-to-site differences was found to have a minimal effect on the cost of the alternative concepts. Once a requirement for seismic design was established, only small-impact design changes were needed to accommodate the different safe shutdown earthquake ground accelerations, windloadings, and flood levels of the six sites evaluated.

- Changes in the design criteria for length of operation and plant cooldown established for the alternative DHR concepts can seriously affect the practicality of certain alternatives. For extended operation beyond ten hours, the PWR add-on HPI train (i.e., feed-and-bleed operation) may be questionable because of water and heat buildup inside containment. For achieving cold shutdown, the PWR add-on AFW train proves to be inadequate because of its dependence on boiling water in the steam generators.

6.4 The Potential Value of Alternative Systems

- For the PWR add-on AFW and HPI concepts, an unavailability of 0.10 per demand was established based on component failure estimates derived from the Oconee #3 risk assessment.[4] For the BWR add-on suppression pool cooling/low pressure injection concept, an unavailability of 0.08 per demand was assessed based on component failure estimates provided in the Grand Gulf #1 risk assessment.[5]

- For the PWRs analyzed, an add-on single train auxiliary feed or HPI (i.e., feed-and-bleed) system reduces the core melt frequency of power plants having three or more DHR trains by only a factor of about two, while PWR plants having two installed DHR trains can be improved by a factor of at least ten.

- For BWRs similar to Grand Gulf, an add-on single-train suppression pool cooling/low pressure injection system reduces core melt frequency by a factor of about six, assuming that independent steps are taken to improve the reliability of the reactor protection system.

- A special emergency can simultaneously jeopardize several DHR trains and significantly increase overall core melt frequency (e.g., fire involving redundant electrical power cables). However, the vulnerability caused by the special emergency may be remedied, without the installation of a completely new add-on DHR system, by providing protection for existing plant systems.

- The safety value of an add-on system should be assessed relative to the value of other, lower-impact, improvement schemes for existing systems.

- For the plants and alternative DHR concepts analyzed, safety improvement factors based on core melt frequency reductions generally equal or exceed improvement factors based on radioactive material release frequencies; and therefore, the potential risk reduction of the alternative concepts can be less than their potential for core melt reduction.

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1. REPORT NUMBER *(Assigned by DDC)*
NUREG/CR-2883/1 of 3
SAND82-1796

4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*
STUDY OF THE VALUE AND IMPACT OF ALTERNATIVE DECAY HEAT REMOVAL CONCEPTS FOR LIGHT WATER REACTORS

2. *(Leave blank)*

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
Dennis L. Berry and Gary A. Sanders

5. DATE REPORT COMPLETED

MONTH	YEAR

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*

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P. O. Box 5800
Albuquerque, NM 87185

DATE REPORT ISSUED

MONTH	YEAR
June	1983

6. *(Leave blank)*

8. *(Leave blank)*

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Division of Risk Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. PROJECT/TASK/WORK UNIT NO.

11. FIN NO.
A1226

13. TYPE OF REPORT
Technical

PERIOD COVERED *(Inclusive dates)*

15. SUPPLEMENTARY NOTES

14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

Reliability assessments were made of systems used to remove decay heat from pressurized and boiling water reactors. Current design practices in both U. S. and foreign plants were reviewed in order to identify the types of systems commonly used for decay heat removal and to determine the regulatory criteria that control decay heat removal system design in various countries. Typical existing decay heat removal system designs were identified for a number of different plant configurations, and the reliability of each of these systems was assessed. Alternative decay heat removal systems were postulated and improvements in decay heat removal reliability were assessed for various combinations of plant configurations and alternative decay heat removal concepts. Alternative concepts that could be implemented by retrofitting existing plants and ones that would be feasible only for new construction were considered. Cost estimates were made for those alternatives that provided significant improvements in decay heat removal reliability.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
NTIS

19. SECURITY CLASS *(This report)*
Unclassified

20. SECURITY CLASS *(This page)*
Unclassified

21. NO. OF PAGES
106

22. PRICE
S

NRC FORM 335 (11-81)

☆ U.S. GOVERNMENT PRINTING OFFICE: 1983-0-676-027/551

120555003908 1 1AN1R1
US NRC
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WASHINGTON DC 20555

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