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# An Approach to the Quantification of Seismic Margins in Nuclear Power Plants:
## The Importance of BWR Plant Systems and Functions to Seismic Margins

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Prepared for
U.S. Nuclear Regulatory Commission

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## ABSTRACT

In NUREG/CR-4334 ("An Approach to the Quantification of Seismic Margins in Nuclear Power Plants"), the Expert Panel on Quantification of Seismic Margins presented a technique for studying the issue of quantifying seismic margins. As part of that technique, the panel included methods for simplifying the margins assessment by screening out components and systems using both systems and fragilities screening guidelines. At the time of that report, the panel was able to develop fragilities screening guidelines for all plants, however the systems screening guidelines applied only to PWRs (due to a shortage of BWR seismic PRAs upon which to base BWR systems screening guidelines). This report develops the BWR systems screening guidelines by utilizing the results of a number of BWR PRAs which have become available since the publication of NUREG/CR-4334.

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## LIST OF ACRONYMS

| | |
|---|---|
| AC | Alternating Current |
| ADS | Automatic Depressurization System |
| BWR | Boiling Water Reactor |
| CM | Core Melt (or Core Damage) |
| DC | Direct Current |
| ECC | Emergency Core Cooling |
| HPCI | High Pressure Coolant Injection (System) |
| HPCS | High Pressure Core Spray (System) |
| LOCA | Loss of Coolant Accident |
| LPCI | Low Pressure Coolant Injection (System) |
| MCC | Motor Control Center |
| NC | Normal Cooldown |
| PRA | Probabilistic Risk Assessment |
| PWR | Pressurized Water Reactor |
| RCIC | Reactor Core Isolation Cooling (System) |
| RHR | Residual Heat Removal (Function)<br>Residual Heat Removal (System) |
| RPS | Reactor Protection System |
| RS | Reactor Subcriticality |
| SLCS | Standby Liquid Control System |
| S/RV | Primary Safety/Relief Valve |
| VAC | Volts Alternating Current |
| VDC | Volts Direct Current |
| VS | Vapor Suppression |

# 1. INTRODUCTION

This report is an extension of the work of the Expert Panel on the Quantification of Seismic Margins which was documented in NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants" [Budnitz, R.J., et al, August 1985]. That work can be briefly summarized by quoting from the abstract of the report.

> "The objective of this report is to discuss progress to date in studying the issue of quantification of seismic margins in nuclear power plants. In particular, it deals with progress towards the establishment of review guidelines that would be useful in studying how much seismic margin exists....
>
> The work presented in this report is the result of a detailed study of seismic Probabilistic Risk Assessments, historical earthquake performance of nuclear and non-nuclear facilities, and test data, augmented by the individual experience and expertise of the Panel members. The major development discussed in this report is the HCLPF concept, which demonstrates margin by showing that there is a High Confidence of a Low Probability of Failure for a given earthquake size."

The extension work documented in this report expands upon one of the areas treated in NUREG/CR-4334 for which insufficient data was available at the time that report was written. Since this report is not intended as a stand-alone document, the HCLPF concept and its associated documentation will not be discussed here in detail. The reader is referred to NUREG/CR-4334 for further information, and a familiarity with that report is essential for a thorough understanding of the material presented in this report.

There is, however, one specific part of the overall HCLPF concept which is particularly important to the work reported here, and so it will be discussed in slightly greater detail. A major step in the approach developed in NUREG/CR-4334 is the screening of systems and components in the margins analysis. In order to simplify the margins analysis, it is necessary to limit the application of detailed analytical techniques to as few structures and pieces of equipment as possible. NUREG/CR-4334 developed a screening technique which allows the analyst to eliminate from consideration certain classes of components and some entire systems based on a set of rules. The system screening eliminated systems based on a review of seismic PRA results which indicated that certain plant safety functions dominated the core melt scenarios most likely to occur following a seismic event, while other plant safety functions did not have a dominant effect on seismic core melt. Thus, by screening out those systems (or parts of systems) which were used only in the performance of these "non-dominant" functions and concentrating on the systems (or parts of systems) which performed the "dominant" functions, the margins review could be greatly simplified. Unfortunately, at the time this "systems screening" technique was developed, there were only a sufficient number of PWR seismic PRAs available to come to a consensus on screening insights. Thus, NUREG/CR-4334 only contained a systems screening technique for PWRs. Since that time, a number of BWR seismic PRAs have become available, which makes it possible to expand the systems screening technique from NUREG/CR-4334 to include BWRs. It is this BWR system screening technique which is the subject of this report.

A word of warning is required prior to the presentation of the results. It should be noted that these insights are based on the results of only six BWR seismic PRAs containing forty-two dominant sequences. However, substantial conservatism was injected into the development of the BWR insights in order to compensate in some way for this shortage of data (see Section 5). It is felt that the insights presented here can be used with the same level of confidence as exists for the PWR insights contained in NUREG/CR-4334. Also, it should be noted that core damage frequency, not risk, is used as the figure of merit for developing these insights, consistent with the current seismic margins methodology.

## 2. FUNCTIONAL INSIGHTS ON SEISMIC MARGINS FOR BWRs

This section discusses the functional insights on seismic margins developed for BWRs based on the review of six BWR PRAs. Section 2.1 describes the five plant safety functions which are commonly associated with BWR response to upset in steady state operating conditions. Section 2.2 discusses how these functions appear to relate to each other during seismic events and the implications of those relationships for the seismic margin review process.

### 2.1 BWR Plant Functions

In order to evaluate and compare the results of the seismic PRAs and develop insights for system level screening, it is necessary to look at the results in terms of the plant safety functions. This is required because these functions are the one aspect of plant response to accident conditions which are the same for all BWRs, while the systems which perform these functions may differ from plant to plant. A list of the plant safety functions generally considered in BWR plant PRAs (seismic or otherwise) is as follows:

1) Reactor Subcriticality - shutting down the nuclear reaction such that the only heat being generated is decay heat.

2) Normal Cooldown - providing cooling to the reactor core through the use of the normal power conversion system, normally defined as the main steam, turbine bypass, condenser, condensate, and main feedwater subsystems.

3) Vapor Suppression - controlling the build-up of pressure in the containment due to the evolution of steam by condensing this steam throughout the event sequence. This is accomplished by passing the steam released by the reactor coolant system through a large volume of water in the containment, condensing the steam and heating the water. A secondary effect of this function is to remove some of the radioactive effluents which may be released along with the steam.

4) Emergency Core Cooling - providing cooling to the reactor core during the transient and stabilized phases of an event sequence by the use of one or more of the emergency systems designed for this purpose. The exact timing of this function is somewhat plant and sequence dependent, but it can be deemed to be the time period during which these systems are initially called upon to operate through the time when the reactor coolant system level and pressure are stabilized and the heat being generated in the core drops below the capability of the residual heat removal function (see below).

5) Residual Heat Removal - removing heat from the containment during the stabilized phase of an event sequence by the use of one or more of the emergency systems designed for this purpose.

## 2.2 Presentation of BWR Functional Screening Insights

The BWR functional screening insights have been developed by reviewing and interpreting the results of six seismic PRAs performed on BWRs. This is discussed in detail in Section 4. This section presents the screening insights obtained from that review.

Regarding the functions identified in the previous section, the dominant core melt sequences always involved failure of Normal Cooldown and never involved failure of Vapor Suppression. As far as the other functions, various success and failure combinations were observed in the dominant sequences, and no particular pattern was observed. Thus, the functional interrelationships identified for PWRs, as documented in NUREG/CR-4334, do not appear to exist for BWRs. Rather, the straightforward insight that Normal Cooldown always fails and Vapor Suppression never fails allows us only to eliminate these functions from the margin review. All other functions must be considered.

As with the PWR case, it is also necessary to consider the initiating events which must be considered. With respect to transient initiators, every dominant sequence involves a loss of offsite power. Therefore, loss of offsite power can be assumed, and other transient initiators need not be considered. With respect to loss of coolant (LOCA) initiators, the only LOCAs observed in the dominant sequence list can be categorized as large LOCAs or vessel rupture. Thus, only LOCAs involving the largest RCS piping (recirculation and main steam/feedwater piping) need be included in the margin review. Since smaller LOCAs did not appear in the dominant sequences other than as transient induced S/RV LOCAs (which are evaluated directly on the event trees), they need not be considered. Finally, no seismic induced containment failures appeared in the list of dominant sequences, so they do not need to be considered in the margin review either.

Based on the above discussion, the final BWR functional screening insights are presented in Table 2.1. As previously noted, these insights are based on the end state of core damage. If some other measure were used (such as one of the offsite consequence measures) some of the insights <u>might</u> be different. (Note - see Section 5).

## 3. PLANT SPECIFIC EXAMPLE OF FUNCTION/SYSTEM RELATIONSHIPS

Up to this point the discussion has concentrated on plant functions. Obviously, the heart of these functions is the systems which perform them. However, the precise systems which are required in the performance of these functions can differ greatly from plant to plant. Getting system insights directly is, therefore, extremely unlikely. However, for a specific plant undergoing a margin review it is important that the systems required to perform each function be identified. Since it is not possible to create a generic list of systems, this section presents a specific example of the identification process for the LaSalle nuclear plant, a BWR owned by Commonwealth Edison. The text assumes some understanding of the systems in the plant. For those who are not familiar with the plant design, a description is provided as Appendix A.

Table 3.1 shows the front line systems which are used to perform each function. The leftmost column lists the functions and the systems are listed across the top. An "X" under a system means that the system is utilized in the performance of the function in that row. By reading across a functional row, it is possible to quickly identify all the front line systems involved in the performance of the function. Front line systems are defined as those systems which are directly used in the performance of a function. Therefore, since our screening insights infer that we are interested only in the first, fourth, and fifth functions for a seismic margin review, the front line systems list for a margin review at LaSalle would be:

- Reactor Protection System
- Standby Liquid Control System
- Safety/Relief Valves
- Reactor Core Isolation Cooling
- High Pressure Core Spray System
- Low Pressure Coolant Injection/Residual Heat Removal
- Low Pressure Core Spray System
- Condensate System

The list of front line systems is only part of the problem. Excluded from this designation are those systems which provide support (cooling, power, control, etc.) to the front line systems. These are referred to as support systems. In order to determine which support systems are required in the analysis, another matrix needs to be constructed. This matrix relates the front line systems to their required support systems, and is shown in Table 3.2. The determination of which support systems would have to be included in the analysis is performed by going across the table for each front line system from the above listing and finding the indicated dependencies. An "X" signifies a dependency in all cases, a "Y" signifies an AC power dependency restricted to offsite power only (no emergency AC power is available to operate the system), and a "Z" signifies that the indicated dependency exists only for the utilization of the system for performing the residual heat removal function. The first finding from this table is that the condensate system should be removed from the front line systems list because it requires offsite power and the screening insights state that offsite power should be assumed to fail. Following that, the support system list for a margin review of LaSalle would be as follows:

- DC Power System
- AC Power System
- Diesel Generator Service Water System
- Residual Heat Removal Service Water System

The final step is to determine if the support systems already identified on the list require any additional support systems which are not yet included. That is, are there any support systems which do not directly support a required front line system but which do support a required support system? The matrix for this is shown on Table 3.3. An "X" or a "Y" signify the same thing as on Table 3.2 and an "N" means "not applicable." Following the procedure previously used, it can be determined that no additional support systems need be added to the list.

It is important to note that, in addition to screening out some support systems in their entirety, there are large parts of the support systems retained which do not have to be analyzed. Support systems are very complex and provide support to many plant systems. When the margins assessment is performed, it will only be necessary to include those parts of the support systems which provide support to the required front line systems.

The above exercise has developed a final list of which systems would be included in a margin review of LaSalle. The list is as follows:

- Reactor Protection System
- Standby Liquid Control System
- Safety/Relief Valves
- Reactor Core Isolation Cooling

- High Pressure Core Spray System
- Low Pressure Coolant Injection/Residual Heat Removal
- Low Pressure Core Spray System
- DC Power System
- AC Power System
- Diesel Generator Service Water System
- Residual Heat Removal Service Water System

# 4. TECHNICAL BASIS FOR SCREENING INSIGHTS

This section will present the information used for the development of the functional insights discussed in Section 2 and the reasoning used to develop those insights from that information.

## 4.1 Summary of Seismic PRA Results

In this section, the dominant seismic core melt sequences from the six PRAs used for this report will be discussed in detail. These results are summarized in six tables (4.1 through 4.6). The format for each of the tables is identical. The first column identifies the particular functional failure (from the functions given in Section 2.1) which directly resulted in core melt. That is the functional failure which, given the state of the plant when that function failed, would lead to a core melt sequence regardless of the success or failure of any subsequent functions. The second column is a brief narrative description of the sequence. The third through seventh columns give the state of each plant function for each sequence. An "S" in the column means the function has succeeded, an "F" in the column means that the function has failed, and a "-" in the column means the state of that function is not important (i.e., that function has no effect on the end state of the sequence). The sequences are listed in descending order of their contribution to core melt frequency. Every sequence for all plants involves a loss of offsite power due to failure of low capacity switchyard components, so this will not be mentioned in the discussion of each sequence. It should be noted that the names of the plants are not given. This is because the results of the seismic PRAs for most of the plants are not generally available. Therefore, no names are given in order to prevent association of the plant with the results.

As an aside, it is worth noting that NUREG/CR-4334 discussed some general insights from seismic PRAs. Those insights are supported by the additional PRAs presented here. Of particular note, these additional PRAs support the contention that seismic core damage is usually dominated by events in the range of 0.3 to 0.5g or so.

### 4.1.1 BWR-A

The results for plant BWR-A are presented in Table 4.1. The first sequence is caused by a complete loss of AC power due to seismic induced relay chatter. This fails all systems except the RPS, RCIC, and HPCI. RCIC and HPCI fail due to failure of the condensate storage tank, which provides suction to the RCIC and HPCI pumps, and the inability to switch RCIC or HPCI to their alternate suction source prior to pump failure. The failure to depressurize the RCS is not really an operator error, but rather the conscious decision by the operator not to depressurize because it would do no good due to the unavailability of AC power to the LPCI system.

The second and fifth sequences are caused by the collapse of the reactor building or the control building, either of which will sever numerous pipes and electrical lines. This results in the failure of virtually all plant systems. The difference between the two sequences is that in the second sequence the control rods can still insert and shut down the reactor whereas in the fifth sequence the reactor core shroud support fails, resulting in core movement preventing insertion of the control rods.

The third sequence involves failure of vessel support, in this case resulting in a large LOCA due to vessel rupture. This is assumed to create loss of all RCS inventory and an uncoolable geometry. Failure of RHR occurs because of secondary failures to the system resulting from the vessel failure.

The fourth sequence is caused by failure of the control rods to insert due to core internal failure resulting in core movement. The standby liquid control system (SLCS) fails to shut down the reactor due to random (non-seismic) failures. The other plant systems are not sized to remove heat following shutdown failure, so their states are not important.

The sixth sequence is caused by seismic induced failure of the RHR heat exchangers, which fails long term cooling. In this case, initial core cooling is provided by either RCIC or HPCI.

### 4.1.2 BWR-B

The results for plant BWR-B are presented in Table 4.2. The first sequence is caused by the loss of all AC power due to random (non-seismic) failures of the emergency diesel generators. This fails all systems except the RPS and RCIC. RCIC fails because of seismic induced failure of the condensate storage tank, which provides suction to RCIC, and the inability to switch the RCIC to its alternate suction source prior to pump failure.

The second sequence is also caused by the loss of all AC power due to random failures of the diesel generators. In this case, the RCIC succeeds in providing cooling. Core melt still results because the diesel generators are not recovered prior to the need for the RHR system for long term heat removal, which requires AC power to operate.

### 4.1.3 BWR-C

The results for plant BWR-C are presented in Table 4.3. The first sequence is caused by seismic induced failure of the RHR heat exchangers, which are required for long term cooling. The high pressure injection systems are not affected, thus initial core cooling is available.

The second sequence is caused by the loss of all AC power due to random (non-seismic) failures of the diesel generators. This fails all systems except the RPS and RCIC. Core melt results from the lack of AC power to run the RHR system when long term cooling is eventually required.

The third sequence is caused by the collapse of the auxiliary building, which results in failure of all systems except the RPS due to damage to numerous pieces of equipment and the severing of piping and electrical lines.

The fourth sequence is caused by the failure of the core shroud support, which results in the inability to insert the control rods due to core movement. Random (non-seismic) failure of the SLCS prevents the use of this system to shut down the reactor. The other systems are not sized to remove heat following shutdown failure, so their states are not important.

The fifth sequence is caused by a number of seismic induced failures which fail all the injection systems. One combination is failure of the condensate storage tank, which fails all high pressure injection, along with failure of the RHR heat exchangers, which fails low pressure injection. Other failures include relay chatter and failure of the service water system, either of which will fail both high and low pressure injection. RHR failure also results from these seismic failures.

The sixth sequence is caused by random (non-seismic) failures of high pressure injection and automatic depressurization, which result in a loss of initial core cooling. Failure of the core shroud support results in the inability to insert control rods, but the SLCS is unaffected and successfully shuts down the reactor. The RHR system is also unaffected, and long term cooling can be provided in the containment.

### 4.1.4 BWR-D

The results for plant BWR-D are presented in Table 4.4. The first sequence is caused by loss of all AC power due to random failures of the emergency diesel generators. This results directly in the failure of all systems except the RPS and RCIC. RCIC fails indirectly due to loss of cooling for the RCIC pump room, since the cooling system requires AC power. The failure to depressurize the RCS is not really an operator error, but rather the conscious decision by the operator not to depressurize because it would do no good due to the unavailability of AC power to the LPCI system.

The second sequence is caused by the random (non-seismic) failure of the "C" diesel generator, which results in the failure of HPCS, and seismic induced failure of the emergency chilled water pump supports, which results in failure of RCIC and LPCI due to loss of required cooling. This also causes the RHR system to fail for the same reason. The RPS system in unaffected.

The third sequence is caused by the seismic failure of the vessel top guide, which results in the inability to insert the control rods. HPCS and RCIC fail because the operator fails to transfer pump suction from the normal source, the suppression pool, to the backup source, the condensate storage tank, prior to suppression pool overheat. It should be noted that, even though the PRA evaluated it, the success or failure of HPCS and RCIC does not affect the eventual core melt result due to failure to shut down the reactor.

The fourth sequence is caused by total loss of AC power due to seismic failure of the emergency circulating water sluice gates, which results in a loss of cooling to the emergency diesel generators. This directly fails all systems except the RPS and RCIC, however, RCIC fails due to loss of room cooling.

The fifth sequence is also caused by a total loss of AC power, this time due to seismically induced unrecoverable relay chatter on the 4160V switchgear breakers. This is accompanied by a large LOCA caused by seismic failure of the recirculation pump supports resulting in recirculation line failure due to pump movement.

The sixth sequence is also caused by total loss of AC power, this time due to seismic failure of a lube oil sump tank which fails the "A" and "B" diesel generators and random (non-seismic) failure of the "C" diesel generator.

### 4.1.5 BWR-E

The results for plant BWR-E are presented in Table 4.5. The first sequence results from a loss of service water due to seismic induced failure of the service water pumphouse or screenwell building walls or seismic induced trip of the local 480V MCCs. This fails all cooling to the RHR systems and to the emergency diesel generators. This fails all systems except the RPS, RCIC, and HPCI (the latter two of which utilize turbine driven pumps). Core melt results due to the inability to provide power and cooling to the RHR systems when long term cooling is required.

The second sequence results from a seismic failure of the reactor vessel supports or the recirculation pump supports, which leads to a very large LOCA which is likely to involve shearing of safety system lines and/or leak rates in excess of ECC capability. All fluid systems are assumed to be disabled.

The third sequence results from seismic failure of the control building or of the cable trays in that building. This interrupts power and control signals to all systems and results in failure of all systems which require active power and control to function (all ECC and RHR systems).

The fourth and eighth sequences are similar, and result from seismic failure of the emergency diesel systems (due to various seismic failures of support bolts, switchgear, etc.) resulting in station blackout. Core cooling is still provided by either RCIC or HPCI, but power cannot be recovered in time to establish long term cooling, so core melt results. The difference between the two sequences is that the eighth sequence also involves non-seismic (random) failure of a primary S/RV to reclose, resulting in a small LOCA.

The fifth and tenth sequences are similar, and result from non-seismic (random) failures which disable the ECC and RHR systems (HPCI, RCIC, and LPCI/RHR). The difference between the two sequences is that the tenth sequence also involves random failure of a primary S/RV to reclose, resulting in a small LOCA.

The sixth sequence results from random failures of the HPCI and RCIC systems along with a mix of random and seismic failures of the ADS depressurization valves. The seismic failures are due to failure of the nitrogen accumulators which provide pressure to open these air-operated valves. These failures combine to render all core cooling inoperative, however they do not prevent the use of RHR for containment cooling.

The seventh and ninth sequences are similar, and involve core melt due to random failures in the RHR systems when required for long term cooling. In both cases, core cooling is initially successful, but in different manners. In the seventh sequence, ECC is provided by RCIC or HPCI. In the ninth sequence, RCIC and HPCI fail due to random causes and ECC is provided by utilizing depressurization and LPCI.

### 4.1.6 BWR-F

The results for plant BWR-F are presented in Table 4.6. The first sequence results from seismic failure of the service water system discharge lines, which fails all plant service water. This results in a loss of all AC power due to loss of cooling to the emergency diesels, which in turn fails all RHR and low pressure ECC systems. Loss of all high pressure cooling pumps (which are turbine driven and do not require AC power) also occurs due to loss of room cooling (which requires AC).

The second and seventh sequences are similar, and result from seismic failure of the 480 VAC breaker cabinets and the 125/250 VDC switchgear. This causes loss of all power and control signals to the safety systems, and results in failure of all those systems which require positive power or control to be successful (all fluid systems). RPS is not directly affected. The difference between the two sequences is that the seventh sequence also includes failure of the RPS (either seismically or through random faults not related to the power/control failures).

The third, tenth, and eleventh sequences are all similar, and are the result of seismic failures of the RCS resulting in very large LOCAs and extensive damage to safety system piping. This is assumed to lead to core melt due to the inability of any ECC or RHR systems to mitigate the event. The difference between the sequences is that the third sequence results from seismic failure of the recirculation pump support lugs, the tenth sequence from seismic failure of reactor vessel support (the shield wall), and the eleventh sequence from both of these seismic failures.

The fourth and eighth sequences are also similar, both resulting from seismic failure of the reactor and control building shear walls. This causes extensive damage to all of the ECC, RHR, and other systems which require through building piping, power, or positive control. Only the RPS is unaffected. The difference between the two sequences is that the eighth sequence also includes failure of the RPS (either seismically or through random faults not related to the structural failures).

The fifth sequence results from seismic failure of either the RHR outboard injection valves, the RHR pump supports, or the RHR heat exchangers, which fail long term cooling when it is required. Core cooling is initially supplied by either the RCIC or HPCI systems, which are unaffected.

The sixth sequence results from a failure of the RPS to scram due either to seismic or random causes. Subcriticality fails due to the operator failing to properly initiate the standby liquid control system (SLCS). A high human error probability for this act is a direct result of the seismic event (thus it could be deemed a seismic induced human error). The other emergency systems are not capable of responding to an event where subcriticality is not achieved.

The ninth sequence results from a seismic or random failure of the RPS combined with the seismic or random failure of HPCI. The seismic HPCI failures are failure of the turbine, pump, or condensate storage tank. In this sequence, SLCS is successful in achieving subcriticality, but core melt results when HPCI is unavailable to provide the necessary RCS water level control. RHR is unaffected, and is available for containment cooling.

The twelfth sequence is similar to the ninth sequence. The difference in this case is that, following failure of RPS and success of SLCS, it is an operator error which causes failure of inventory control and depressurization (rather than the HPCI failures observed in the ninth sequence).

### 4.2 Discussion of Seismic PRA Results vs. Screening Insights

The results presented above were used to identify screening insights which would help to simplify the margins analysis required to determine an estimate of the HCLPF for a particular BWR. This was done much in the manner used for developing the PWR functional/systemic screening insights presented in NUREG/CR-4334. In discussing these insights, a two character nomenclature will be used. The five BWR functions will be identified by number as follows:

1 - Reactor Subcriticality
2 - Normal Cooldown
3 - Vapor Suppression
4 - Emergency Core Cooling
5 - Residual Heat Removal

When referring to the state of a function, a letter will be added following the number. An "S" will signify success of the function and an "F" will signify failure of the function. Thus, 1S signifies that reactor subcriticality has succeeded, while 4F signifies that emergency core cooling has failed, and so on. Event probability is signified by the function "P". Thus, P(4F) is the probability that emergency core cooling has failed.

The first thing that was noted was that there did not appear to be an obvious delineation of functional groups for the BWRs as there was for the PWRs. The PWR insights seemed to just drop out of a quick review and the functional group hypothesis which was later adopted was clear. Since this is not the case for the BWRs, the process which we used was to look for initial insights on a function-by-function basis and then aim for further refinements.

The first insight which can be determined from the PRA results is that normal cooldown always fails. In terms of a screening insight, this means that it is not necessary to perform a detailed evaluation of the systems used to perform function 2. Rather, it is reasonable to assume that;

$$P(2F) = 1.0$$

The potential error that could be made would be to identify core damage sequences which result directly from the failure of the other functions and which could have been prevented if function 2 had been available. We will refer to this as a Type III error. (The definition of Type I and Type II errors was established in NUREG/CR-4334 as they apply to PWR functional/systemic insights. In order to avoid confusion, those designations will be used only to discuss errors of the same type for BWRs.) In reviewing the 42 dominant sequences included, it is found that all 42 include failure of normal cooldown. The root cause of this failure is that every sequence involves a loss of offsite power due to the extremely low seismic capacity of key switchyard components. All seismic sequences which have been found to contribute to seismic risk have been large enough to assure failure of these components. Since function 2 requires offsite power in order to succeed (which is true in all nuclear power plants, both BWRs and PWRs) the failure of function 2 is assured. Thus, in the PRAs reviewed, no Type III errors would have occurred if the assumption were used and thus the screening insight is supported. It should be noted that this insight also leads directly to the conclusion that it is not necessary to consider any transient initiating events other than loss of offsite power, since only loss of offsite power initiators contribute to seismic risk.

The second insight identified is that function 3 always succeeds. In terms of a screening insight, this means that it is not necessary to perform a detailed evaluation of the systems which are required for function 3. Rather, it is reasonable to assume that;

$$P(3S) = 1.0$$

Since failure of function 3 will always result directly in core damage, regardless of the states of the other functions, the error which would result from the use of this assumption would be that if there were any dominant sequences which involved failure of this function, they would be missed. This will be called a Type IV error. If this type of error were potentially important, we would expect to see some dominant sequences from the PRAs (where this assumption was not made) which involved failure of function 3. However, of the 42 sequences reviewed, no Type IV errors would have occurred if the assumption were used and thus the screening insight is supported. (Note - see Section 5).

The first two insights leave three functions to be considered. Function 1 is interesting in that it is a basically independent function. That is, if it fails, the states of all of the subsequent functions are unimportant. An early core melt with early containment overpressure failure will occur. Now, no assumption regarding the success or failure of this function can be made as was done for functions 2 and 3 since it fails in some dominant sequences but not in others. However, when considering the contribution of this functional failure to plant HCLPF, it can be analyzed separately from the other functions. Once the independent HCLPF for function 1 is identified it can be taken as an independent element of the plant HCLPF equation. This does simplify the analysis somewhat, although it is not specifically an "insight" in the sense meant in the past work. It falls not from the seismic PRA results themselves as from the results of BWR PRAs in general and the limitations of BWR plant design. There are no significant potential error types of any kind associated with this insight.

With the final two functions, the review has been reduced to a point similar to that of the PWR review. That is, the consideration of early cooling versus late cooling. Briefly, in the PWR analysis, the insight identified was that it was reasonable to assume that if early cooling succeeded then late cooling also succeeded and that if early cooling failed then late cooling also failed. This permits the margins review to concentrate the detailed analysis only on the systems which provide the early cooling functions and to consider only those aspects of the late cooling functions which are the result of very gross plant unique design features. This saves a large amount of analysis. In order to see if a similar insight to the PWR insight is possible, it will be hypothesized that a similar insight exists in the BWR case. Put in terms of the PWR insight, this would be expressed as follows:

$$P(5S|4S|EQ) = 1.0$$

$$P(5F|4F|EQ) = 1.0$$

That is, the probability of 5S given 4S given the the occurrence of an earthquake is 1.0 (a certainty) and the probability of 5F given 4F given the occurrence of an earthquake is also 1.0. Direct corollaries of these statements are:

$$P(5S|4F|EQ) = 0.0$$

$$P(5F|4S|EQ) = 0.0$$

and, interestingly, combining with the insight discussed above regarding function 1:

$$P(CM|EQ) = P(1F|EQ) + P(4F|EQ)$$

$$P(NCM|EQ) = P(1S|EQ) + P(4S|EQ)$$

where "CM" is core melt (core damage) and "NCM" is no core melt (no core damage). Thus, what the above two equations signify is that, if the assumption can be shown to be viable, it is possible to evaluate core damage probability (or, obviously, HCLPF) by only performing detailed evaluations of function 1 and function 4, and what's more, the system analysis part of the evaluations could be completely independent. As stated in NUREG/CR-4334, this kind of assumption is prone to two types of error, as follows:

> Type I Error - This error involves seismic core damage sequences which would result from the success of function 4 and failure of function 5. These sequences will be missed since the core damage is caused by failure of function 5 and the assumption would be that if function 4 succeeds then function 5 succeeds, which is a non-core damage sequence. This type of error would be obviously non-conservative.
>
> Type II Error - This error involves seismic core damage sequences which result from failure of function 4 but which are followed by success of function 5. These sequences would not be entirely missed since the core damage results from failure of function 4, which the assumption would allow to be found. However, they would erroneously be considered to have included failure of function 5 since the assumption states that success of function 5 is not probable given failure of function 4. This type of error would be conservative, since these 4F/5S sequences would be placed into 4F/5F plant damage states, which have higher consequences. Further, since the HCLPF concept is transparent to plant damage state, the calculation of HCLPF would not be affected by this type of error.

Going back and reviewing the 42 dominant seismic sequences in Tables 4.1 through 4.6, it can be seen that if the assumption presented above had been used in the PRAs there would have been ten Type I errors and four Type II errors. Further, four of the Type I errors are potentially quite significant. Type I errors represent the top two dominant sequences for BWR-C, the most dominant sequence for BWR-E, and one of the only two dominant sequences for BWR-B. These Type I errors could cause a potentially serious overestimation of the HCLPF for all three plants, which is half the available data base. Thus, the seismic PRA results do not support this assumption, and the hypothesized functional insight is not valid.

The final consideration with regard to the functional insights is which initiating events need to be considered in the margins review. As previously discussed, the transient initiator loss of offsite power is the only transient required to be analyzed, and loss of offsite power should be assumed for all sequences evaluated. This leaves the consideration of the three traditional LOCA sizes, vessel rupture, and direct containment failure. It was decided that any initiating event which appeared in the forty-two sequences in the data base should be included in the list of initiators to be evaluated in the margins review and those which did not appear could be eliminated. Reviewing the forty-two sequences, two other initiators appeared (in combination with loss of offsite power), Large LOCA and Vessel Rupture. Thus, these should also be considered in the margins review.

The functional insights which resulted from the above process were presented in summary form in Section 2 and on Table 2.1.

## 5. LIMITATIONS

There are a number of limitations on the insights presented in this report. In general, they are the same as those discussed for the entire methodology in NUREG/CR-4334. However, one particular limitation applies more strongly to the BWR functional/systemic insights presented here than to the equivalent PWR insights from the previous report. That is the fact that the insights are based on the results of a limited number of seismic PRAs. The PWR insights were founded on a base of 10 seismic PRAs containing 60 dominant accident sequences. In contrast, the BWR insights have a base of only 6 seismic PRAs containing 42 dominant accident sequences. Thus, we were much stricter in identifying BWRs insights. In the PWR analysis, we permitted exceptions to the insights to exist as long as they would not have significantly altered an assessment of plant HCLPF and identification of plant weaknesses had a margin study been performed on that plant. That is, all 60 of the dominant sequences identified did not have to agree perfectly with the insights in order to allow the insight to be proclaimed a valid and reasonable approximation. However, in the BWR analysis, any insight which would allow simplification of the systems model was considered valid only if it were valid for all 42 dominant sequences. Thus, function 2 being assumed to fail and function 3 being assumed to succeed are designated as valid insights because these functional states were present in all 42 sequences. Similarly, initiating events were eliminated from consideration only if they did not appear in any of the 42 sequences. An interesting note is that even if an occasional insignificant exception were allowed in the development of the BWR insights, the conclusions would not have changed. That is because the results of the six PRAs considered did not reveal any additional simplifying assumptions whose exceptions could be judged to be broadly insignificant to the assessment of a plant HCLPF.

In addition, there is one other limitation which applies specifically to these BWR results. All of the seismic PRAs considered in this report have either MARK II or MARK III containments (no seismic PRA were available for plants with MARK I containments). Therefore, the insight with regard to the vapor suppression function (function 3) always being available and the insight that it is not necessary to consider containment integrity failure as an initiating event <u>may</u> not apply to plants with Mark I containments. This is not necessarily a critical weakness, since in their functional system behavior, BWRs are not all that different from one to the next. Nevertheless, it appears that it will be necessary (and prudent) to give some consideration to these failures for the first few MARK I plants to be subjected to a margins review. After a fragilities analysis has been performed on a few representative MARK I containments in the context of a margins study, it will be possible to positively determine whether this insight can be extended to that containment type.

## TABLE 2.1

## FUNCTIONAL SCREENING REQUIREMENTS FOR BWR SEISMIC MARGINS REVIEW

| Function | Screening Requirement |
|---|---|
| Initiators: | |
| Offsite Power | Assume Failure |
| RCS Integrity - Small LOCA | Assume Success |
| RCS Integrity - Medium LOCA | Assume Success |
| RCS Integrity - Large LOCA | Margin Evaluation Required |
| RCS Integrity - Vessel Rupture | Margin Evaluation Required |
| Containment Integrity | Assume Success [1] |
| Plant Functions: | |
| Reactor Subcriticality | Margin Evaluation Required |
| Normal Cooldown | Assume Failure |
| Vapor Suppression | Assume Success [1] |
| Emergency Core Cooling | Margin Evaluation Required |
| Residual Heat Removal | Margin Evaluation Required |

[1] See discussion in Section 5 regarding MARK I containments.

TABLE 3.1

PLANT FUNCTIONS vs. FRONT LINE SYSTEMS MATRIX

| PLANT FUNCTIONS / FRONT LINE SYSTEMS | Reactor Protection System | Standby Liquid Control System | Power Conversion System | Safety/Relief Valves | Suppression Pool | Reactor Core Isolation Cooling | High Pressure Core Spray System | Low Press Cool Inj/ Residual Heat Rem. | Low Pressure Core Spray System | Condensate System |
|---|---|---|---|---|---|---|---|---|---|---|
| Reactor Subcriticality | X | X | | | | | | | | |
| Normal Cooldown | | | X | | | | | | | |
| Vapor Suppression | | | | X | X | | | | | |
| Emergency Core Cooling | | | | X | | X | X | X | X | X |
| Residual Heat Removal | | | | | | | | X | | |

TABLE 3.2

FRONT LINE SYSTEMS vs. SUPPORT SYSTEMS MATRIX

| FRONT LINE SYSTEMS / SUPPORT SYSTEMS | AC Power System | DC Power System | Residual Heat Removal Service Water System | Diesel Generator Service Water System | Circulating Water System | Turbine Building Closed Cooling Water | Reactor Building Closed Cooling Water |
|---|---|---|---|---|---|---|---|
| Reactor Protection System | | X | | | | | |
| Standby Liquid Control System | X | X | | | | | |
| Power Conversion System | Y | X | | | X | X | |
| Safety/Relief Valves | | X | | | | | |
| Suppression Pool | | | | | | | |
| Reactor Core Isolation Cooling | | X | | X | | | |
| High Pressure Core Spray | X | X | | X | | | |
| Low Pres Cool Inj/ Residual Heat Rem. | X | X | Z | X | | | |
| Low Pressure Core Spray | X | X | | X | | | |
| Condensate System | Y | X | | | | X | |

TABLE 3.3

SUPPORT SYSTEMS vs. SUPPORT SYSTEMS MATRIX

| "SUPPORTED" SUPPORT SYSTEM / "SUPPORTING" SUPPORT SYSTEM | AC Power System | DC Power System | Diesel Generator Service Water System | Service Water System |
|---|---|---|---|---|
| AC Power System | N | X | X | |
| DC Power System | | N | | |
| Residual Heat Removal Service Water System | X | X | | |
| Diesel Generator Service Water System | X | X | N | |
| Circulating Water System | Y | X | | |
| Turbine Building Closed Cooling Water System | Y | X | | X |
| Reactor Building Closed Cooling Water System | Y | X | | X |
| Service Water System | Y | X | | N |

TABLE 4.1

SEISMIC PRA RESULTS FOR PLANT BWR-A

| Core Melt Cause | Sequence Description | RS | NC | VS | ECC | RHR |
|---|---|---|---|---|---|---|
| ECC | Seismic Induced Loss of Offsite and Onsite Power, Failure of High Pressure Injection, Failure to Depressurize RCS | S | F | S | F | F |
| ECC | Seismic Induced Collapse of Reactor or Control Building, Loss of All Systems Except RPS | S | F | S | F | F |
| ECC | Seismic Induced Rupture of Reactor Pressure Vessel, Loss of All Systems Except RPS | - | F | S | F | F |
| RS | Seismic Induced Loss of Offsite Power, Failure of RPS and Standby Liquid Control System | F | F | - | - | - |
| RS | Seismic Induced Collapse of Reactor or Control Building, Loss of All Systems | F | F | - | - | - |
| RHR | Seismic Induced Loss of Offsite Power, Failure of Long Term Heat Removal | S | F | S | S | F |

TABLE 4.2

SEISMIC PRA RESULTS FOR PLANT BWR-B

| Core Melt Cause | Sequence Description | RS | NC | VS | ECC | RHR |
|---|---|---|---|---|---|---|
| ECC | Seismic Induced Loss of Offsite Power, Failure of Onsite Power, Loss of All Systems Except RPS | S | F | S | F | F |
| RHR | Seismic Induced Loss of Offsite Power, Failure of Onsite Power, Failure of Long Term Heat Removal | S | F | S | S | F |

TABLE 4.3

SEISMIC PRA RESULTS FOR PLANT BWR-C

| Core Melt Cause | Sequence Description | RS | NC | VS | ECC | RHR |
|---|---|---|---|---|---|---|
| RHR | Seismic Induced Loss of Offsite Power, Failure of Long Term Cooling | S | F | S | S | F |
| RHR | Seismic Induced Loss of Offsite Power, Failure of Onsite Power, Failure of Long Term Cooling | S | F | S | S | F |
| ECC | Seismic Induced Collapse of Auxiliary Building, Loss of All Systems Except RPS | S | F | S | F | F |
| RS | Seismic Induced Loss of Offsite Power, Failure of RPS, Failure of Standby Liquid Control System | F | F | - | - | - |
| ECC | Seismic Induced Loss of Offsite Power, Failure of High and Low Pressure Injection, Failure of Long Term Cooling | S | F | S | F | F |
| ECC | Seismic Induced Loss of Offsite Power, Failure of RPS, Failure of High Pressure Injection and Automatic Depressurization | S | F | S | F | S |

TABLE 4.4

SEISMIC PRA RESULTS FOR PLANT BWR-D

| Core Melt Cause | Sequence Description | RS | NC | VS | ECC | RHR |
|---|---|---|---|---|---|---|
| ECC | Seismic Induced Loss of Offsite Power, Failure of Onsite Power, Failure of High Pressure Injection, Failure to Depressurize RCS, Failure of Long Term Cooling | S | F | S | F | F |
| ECC | Seismic Induced Loss of Offsite Power, Failure of High and Low Pressure Injection, Failure of Long Term Cooling | S | F | S | F | F |
| RS | Seismic Induced Loss of Offsite Power, Failure of RPS, Failure of High Pressure Injection. | F | F | - | F | - |
| ECC | Seismic Induced Loss of Offsite and Onsite Power, Loss of All Systems Except RPS | S | F | S | F | F |
| ECC | Seismic Induced Large LOCA with Loss of Offsite and Onsite Power, Failure of Low Pressure Injection, Failure of Long Term Cooling | S | F | S | F | F |
| ECC | Seismic Induced Loss of Offsite Power, Failure of Onsite Power, Loss of All Systems Except RPS | S | F | S | F | F |

**TABLE 4.5**

**SEISMIC PRA RESULTS FOR PLANT BWR-E**

| Core Melt Cause | Sequence Description | RS | NC | VS | ECC | RHR |
|---|---|---|---|---|---|---|
| RHR | Seismic Induced Loss of Offsite Power and Service Water, Loss of Onsite Power and Long Term Heat Removal | S | F | S | S | F |
| ECC | Seismic Induced Rupture of Reactor Coolant System, Loss of All Systems Except RPS | - | F | S | F | F |
| ECC | Seismic Induced Collapse of Control Building or Cable Trays, Loss of All Systems Except RPS | S | F | S | F | F |
| RHR | Seismic Induced Loss of Offsite and Onsite Power, Loss of Long Term Heat Removal | S | F | S | S | F |
| ECC | Seismic Induced Loss of Offsite Power, Failure of High and Low Pressure Injection, Failure of Long Term Heat Removal | S | F | S | F | F |
| ECC | Seismic Induced Loss of Offsite Power, Failure of High Pressure Injection, Seismic/Random Failure to Depressurize | S | F | S | F | S |
| RHR | Seismic Induced Loss of Offsite Power, Failure of Long Term Heat Removal | S | F | S | S | F |
| RHR | Seismic Induced Loss of Offsite and Onsite Power, Transient Induced Small LOCA, Loss of Long Term Heat Removal | S | F | S | S | F |
| RHR | Seismic Induced Loss of Offsite Power, Failure of High Pressure Injection and Long Term Heat Removal | S | F | S | S | F |
| ECC | Seismic Induced Loss of Offsite Power, Transient Induced Small LOCA, Failure of High and Low Pressure Injection, Failure of Long Term Heat Removal | S | F | S | F | F |

TABLE 4.6

SEISMIC PRA RESULTS FOR PLANT BWR-F

| Core Melt Cause | Sequence Description | RS | NC | VS | ECC | RHR |
|---|---|---|---|---|---|---|
| ECC | Seismic Induced Loss of Offsite Power and Service Water, Loss of Onsite Power, Loss of All Systems Except RPS | S | F | S | F | F |
| ECC | Seismic Induced Loss of Offsite Power and All Control and Instrumentation Power, Loss of All Systems Except RPS | S | F | S | F | F |
| ECC | Seismic Induced Rupture of Reactor Coolant System, Loss of All Systems Except RPS | - | F | S | F | F |
| ECC | Seismic Induced Collapse of Reactor or Control Buildings, Loss of All Systems Except RPS | S | F | S | F | F |
| RHR | Seismic Induced Loss of Offsite Power and Long Term Heat Removal | S | F | S | S | F |
| RS | Seismic Induced Loss of Offsite Power, Seismic/Random Failure of RPS, Failure to Initiate Standby Liquid Control | F | F | - | - | - |
| RS | Seismic Induced Loss of Offsite Power and All Control and Instrumentation Power, Loss of All Systems | F | F | - | - | - |
| RS | Seismic Induced Collapse of Reactor and Control Building, Loss of All Systems | F | F | - | - | - |
| ECC | Seismic Induced Loss of Offsite Power, Failure of RPS, Random/Seismic Failure of High Pressure Injection | S | F | S | F | S |
| ECC | Seismic Induced Vessel Rupture, Loss of All Systems | - | F | S | F | F |
| ECC | Seismic Induced Vessel and Reactor Coolant System Rupture, Loss of All Systems | - | F | S | F | F |
| ECC | Seismic Induced Loss of Offsite Power, Failure of RPS, Failure to Maintain Proper Event Control | S | F | S | F | S |

# APPENDIX A

## LASALLE UNIT 1 SYSTEM DESCRIPTIONS

The purpose of this appendix is to provide background information on the systems design for LaSalle Unit 1, which is used in Section 3 as a plant specific example of the identification of plant systems that are required to provide each plant safety function. Each system mentioned in that section is described in some detail in this Appendix.

### A.1 Reactor Protection System (RPS)

The RPS constantly monitors plant parameters and initiates the actions necessary to shut down the nuclear reaction. It examines selected plant parameters and determines whether plant operating limits are exceeded. If they are, the system transmits actuation signals to the components required to operate in order to shut down the nuclear reaction.

The signal is processed by the main RPS sensors and logic and is transmitted to the scram solenoids. This is a two train arrangement, and either train is capable of sending the required signal. In addition, there is an alternate set of sensors, logic and solenoid valves (called alternate rod insertion (ARI)) which backs up this portion of the RPS. The opening of the scram solenoids results in the opening of a set of redundant scram valves. This admits water from a pressurized scram system to enter the control rod drives and force the control rods into the core, shutting down the nuclear reaction.

### A.2 Standby Liquid Control System (SLCS)

The SLCS provides a backup means of shutting down the nuclear reaction by injecting a concentrated boron solution into the reactor coolant system. The system is entirely manual in operation, and in addition to initiating it the operator must take other actions to reduce core power in order to give the boron concentration sufficient time to build up. It takes about 30 minutes to fully shut down the nuclear reaction using the SLCS.

The SLCS consists of two independent SLCS pumps along, with their associated piping, valves, and control circuitry. Both trains take suction from a single boron injection tank, which contains a concentrated boric acid solution. Success of at least one pump train is required to shut down the reaction.

### A.3 Power Conversion System (PCS)

The PCS is an extremely complex plant system whose function it is to convert heat produced in the reactor core into electrical power. It is the major part of the secondary side of the plant. It is always operating when the plant is at power, and is also capable of removing decay heat after plant shutdown.

There are five major systems that make up the PCS: main steam, turbine, condenser, condensate, and feedwater. The main steam systems takes steam from the reactor and brings it to the turbine. It consists of four main steam lines, each of which has a pair of isolation valves (one inside containment and one outside containment) and other associated valves and piping. The turbine system consists of the turbine itself with its associated turbine stop and control valves along with piping and control circuitry, and the turbine bypass valves with their piping and control circuitry. Steam passes through the turbine during normal operation and around the turbine (through the bypass) when a shutdown occurs. In either case, the steam is passed on to the condenser.

The condenser is a large heat exchanger that cools the steam by passing it over a large number of water cooled tubes. The cooling for these tubes is provided by the circulating water system. The steam is condensed to water and falls into the condenser hotwell, a kind of tank which provides suction for the condensate system.

The condensate and feedwater systems take water from the hotwell and return it to the reactor vessel to be heated again. These two systems contain a number of components, the primary of which are the motor-driven condensate pumps, the main feedwater pumps (two turbine-driven, one motor-driven), and the feedwater stop and control valves. They also contain a number of other pumps and valves, as well as piping, heat exchangers, and control circuitry.

### A.4 Safety/Relief Valves (S/RV)

The S/RVs are designed to protect the reactor vessel from the rupture of any part of its pressure boundary, thus preventing a LOCA caused by overpressure. It does this by providing a group of valves that open when the vessel pressure is too high, removing heat and maintaining vessel pressure within design limits. The valves then reclose, returning the integrity to the pressure boundary. The valves can also be used to depressurize the reactor coolant system by being locked open. This mode of operation is either automatically initiated by low reactor water level or manually initiated by the operator, depending on the scenario. The valves discharge steam into the suppression pool.

The major function of the S/RVs is to relieve pressure from the reactor coolant system when the RCS becomes isolated from the condenser. After the valves open, they must reclose in order to prevent a LOCA condition from existing. A stuck open valve results in a small LOCA. The depressurization mode is used in cases where the RCS is isolated and no high pressure cooling is available. In this case, locking the valves open reduces the RCS pressure so that low pressure cooling systems can be used to provide the necessary cooling.

### A.5 Suppression Pool (SP)

The SP is essentially a large pool of water located inside the containment and is used to condense steam and prevent containment overpressure. The containment is arranged such that steam escaping from the RCS, either through the S/RVs or from a break elsewhere in the RCS pressure boundary, is routed through the SP. The SP contains cool water, and the steam is condensed in passing through the water, causing the water temperature to rise but preventing pressure build-up in the containment from a build-up of steam. The SP also provides an alternate suction source for a number of emergency cooling systems. After a period of time, the SP water gets too hot and must be cooled it it is to continue providing its vapor suppression function.

### A.6 Reactor Core Isolation Cooling (RCIC) System

The RCIC is an engineered safeguards system that is designed to provide reactor core cooling by injecting water into the reactor vessel at high pressure. It is automatically actuated by a low reactor water level signal.

The RCIC consists of a turbine-driven pump along with the associated piping, valves, and control circuitry. Upon receipt of a signal, the system is automatically aligned and the steam admission valve to the pump turbine opens, starting the pump. The steam is provided from the reactor vessel. The pump takes suction from the condensate storage tank (CST) and discharges directly to the reactor vessel. If desired, the pump suction lines can be realigned to take suction from the suppression pool, but this is generally not essential since the RCIC's mission should be completed prior to draining the CST.

### A.7 High Pressure Core Spray (HPCS) System

The HPCS is an engineered safeguards system that is designed to provide reactor core cooling by injecting water into the reactor vessel at high pressure. It is automatically actuated by a low reactor water level signal.

The HPCS consists of a motor-driven pump along with the associated piping, valves, and control circuitry. Upon receipt of a signal, the system is automatically aligned and the pump starts. The pump takes suction from the condensate storage tank (CST) and discharges directly to the reactor vessel. The pump suction lines will be automatically realigned to take suction from the suppression pool when high SP level is detected, but this is generally not essential since the HPCS's mission should be completed prior to draining the CST.

## A.8 Low Pressure Coolant Injection/Residual Heat Removal (LPCI/RHR) System

The LPCI/RHR is an engineered safeguards system designed to provide core cooling by injecting large amounts of water into the reactor vessel at high pressure and also to provide long term decay heat removal from the containment. Depending on its mode of operation, it is actuated by a low reactor vessel water level or high drywell pressure signal or by manual action.

For the core cooling mode, the LPCI/RHR consists of three redundant pump trains along with their associated valves, piping, and control circuitry. Upon receipt of a signal, the system is automatically aligned and the pumps start. The pumps take suction from the suppression pool (SP) and discharge directly to the reactor vessel.

For the residual heat removal mode, the LPCI/RHR consists of the two redundant pump trains (of the three mentioned above) which contain the RHR heat exchangers. Each train has two heat exchangers. Cooling to the heat exchangers is provided by the Residual Heat Removal Service Water System (RHRSWS), which is automatically actuated (see Section A.13). The alignment of the system is performed manually (if required) and depends on the existing plant conditions. For LOCA conditions (the reactor is not isolated and coolant is being lost to the suppression pool), no alignment is required since the suction and discharge is the same as for the core cooling mode. For transient conditions (no coolant loss from the reactor vessel), the pump suction lines are realigned to take suction directly from the reactor vessel and, if the system has not been automatically placed in operation, the pumps are started.

## A.9 Low Pressure Core Spray (LPCS) System

The LPCS is an engineered safeguards system that is designed to provide reactor core cooling by injecting water into the reactor vessel at low pressure. It is automatically actuated by a low reactor water level signal.

The LPCS consists of a motor-driven pump along with the associated piping, valves, and control circuitry. Upon receipt of a signal, the system is automatically aligned and the pump starts. The pump takes suction from the suppression pool (SP) and discharges directly to the reactor vessel.

## A.10 Condensate System (CS)

The condensate system was discussed previously as part of the power conversion system (Section A.3). It is capable of performing a low pressure injection function when the rest of the PCS is unavailable. In this mode, it takes suction from the condenser hotwell and injects it into the reactor vessel through the idle feedwater pumps, using the motor driven feedwater pump control valve as its injection path.

### A.11 AC Power System (ACPS)

The ACPS is a major support system that supplies power to the normal and emergency systems in the plant. The normal power system supplies power from the offsite power grid or the output of the main generator to all plant systems. If these power sources are not available, the emergency AC power system is capable of supplying power to those systems that are required to ensure safe shutdown or to mitigate the effects of any accident condition.

During normal operation, the power is supplied to the electrical system through the normal station service transformer (NSST) from the main generator. In the event of a unit trip, the generator is isolated and power is supplied through the NSST or the reserve SST from the offsite grid.

When offsite power is not available, the emergency AC power supplies a limited amount of power to the two main emergency busses and a third smaller emergency bus. The emergency AC power source consists of three emergency diesel/generator units. The first two are redundant, and each is dedicated to one of the two main emergency busses and is capable of supplying power to all engineered safety features and safe shutdown equipment fed from that bus. The third unit feeds the third emergency bus, which is specifically dedicated to the high pressure core spray system. All three units are started and tied to the emergency busses automatically upon receipt of a loss of offsite power signal. Cooling to the units is supplied by the diesel generator service water system (DGSWS).

### A.12 DC Power System (DCPS)

The DCPS is the support system which provides power to plant controls and instruments, to the switchgear breakers for large pieces of equipment, and to actuate and load the emergency diesel/generators when offsite power is lost. Power is provided by a DC battery for each train (when AC power is not available) or by a battery charger unit (when AC power is available).

The DC power system consists of three trains (each containing a battery and battery charger), along with associated busses, motor control centers, instrument AC inverters, distribution panels, switches, wires, fuses, etc. Each train is dedicated to one of the emergency AC power trains.

### A.13 Residual Heat Removal Service Water System (RHRSWS)

The RHRSWS is the support system that cools the LPCI/RHR system equipment. Raw water suction is taken from the ultimate heat sink and pumped through the RHR heat exchangers and equipment coolers. The water is returned to the ultimate heat sink.

The RHRSWS consists of two pump trains (each containing two pumps), along with associated piping, valves, and control circuitry. Each train is dedicated to a LPCI/RHR train, and provides cooling to the associated RHR pump and heat exchanger. This cooling is only required during the RHR cooling mode of the LPCI/RHR system. The RHRSWS automatically starts on receipt of a signal which would initiate the LPCI/RHR system.

### A.14 Diesel Generator Service Water System (DGSWS)

The DGSWS is the support system that cools most of the engineered safety features equipment. Raw water suction is taken from the ultimate heat sink and pumped through various heat exchangers and equipment coolers. The water is returned to the ultimate heat sink.

The DGSWS consists of three pump trains, along with associated piping, valves, and control circuitry. Each train is dedicated to a particular diesel generator train, and provides cooling to the associated diesel generator and to all the safety equipment which is powered by that electrical train. The system automatically starts on receipt of any signal which would initiate any safety system or diesel generator unit.

### A.15 Circulating Water System (CWS)

The CWS is a support system which provides cooling to the main condenser. It does this by injecting cool water through the tubes of the main condenser. It is a normally operating system.

The CWS consists of a series of circulating water pumps, along with piping, valves, and control circuitry. The pumps take suction directly from the ultimate heat sink and inject it through the circulating water piping to the main condenser. The water is then discharged back to the ultimate heat sink.

### A.16 Turbine Building Closed Cooling Water System (TBCCWS)

The TBCCWS is a support systems which provides cooling to the secondary plant equipment used to operate the secondary plant cycle. It does this by circulating water through various equipment coolers in that equipment. It is a normally operating system.

The TBCCWS consists of the TBCCW pumps, TBCCW heat exchangers, along with piping , valves, and control circuitry. The pumps circulate water in a closed loop, pumping the water first through the TBCCW heat exchangers and the discharge of the heat exchangers is routed to the secondary equipment coolers. The discharge of the equipment coolers is then returned to the TBCCW pump suction header. The cycle is cooled by the service water system.

## A.17 Reactor Building Closed Cooling Water System (RBCCWS)

The RBCCWS is a support systems which provides cooling to the secondary plant equipment used to operate the primary (reactor coolant system) plant cycle. It does this by circulating water through various equipment coolers in that equipment. It is a normally operating system.

The RBCCWS consists of the RBCCW pumps, RBCCW heat exchangers, along with piping , valves, and control circuitry. The pumps circulate water in a closed loop, pumping the water first through the RBCCW heat exchangers and the discharge of the heat exchangers is routed to the secondary equipment coolers. The discharge of the equipment coolers is then returned to the RBCCW pump suction header. The cycle is cooled by the service water system.

## A.18 Service Water System (SWS)

The SWS is a support system which provides cooling the the TBCCW and RBCCW systems. It does this by injecting water into the secondary side of the heat exchangers in those systems. It is a normally operating system.

The SWS consists of a series of service water pumps, along with piping, valves, and control circuitry. The pumps take suction directly from the ultimate heat sink and inject it through the service water piping to the TBCCW and RBCCW heat exchangers. The water is then discharged back to the ultimate heat sink.

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In NUREG/CR-4334 ("An Approach to the Quantification of Seismic Margins in Nuclear Power Plants"), the Expert Panel on Quantification of Seismic Margins presented a technique for studying the issue of quantifying seismic margins. As part of that technique, the panel included methods for simplifying the margins assessment by screening out components and systems using both systems and fragilities screening out components and systems using both systems and fragilities screening guidelines. At the time of that report, the panel was able to develop fragilities screening guidelines for all plants, however the systems screening guidelines applied only to PWRs (due to a shortage of BWR seismic PRAs upon which to base BWR systems screening guidelines). This report develops the BWR systems screening guidelines by utilizing the results of a number of BWR PRAs which have become available since the publication of NUREG/CR-4334.


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