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CORE DAMAGE FREQUENCY PERSPECTIVES BASED ON IPE RESULTS

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ABSTRACT

This paper provides perspectives gained from reviewing 75 Individual Plant Examination (IPE) submittals covering 108 nuclear power plant units. Variability both within and among reactor types is examined to provide perspectives regarding plant-specific design and operational features, and modeling assumptions that play a significant role in the estimates of core damage frequencies in the IPEs.

I. INTRODUCTION

In November 1988, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter 88-20 requesting that all licensees perform an Individual Plant Examination (IPE) "to identify any plant-specific vulnerabilities to severe accidents and report the results to the Commission." The purpose and scope of the IPE effort includes examination of internal events, including those initiated by internal flooding, occurring at full power. In response, 75 IPE submittals covering 108 nuclear power plant units were received by the staff. These IPE submittals were examined to determine which factors are most influential for core damage frequencies (CDFs).

Perspectives regarding the factors that are most influential for the IPE CDFs were obtained by comparing results within and between groups of plants with similar characteristics. The boiling water reactor (BWR) plants were categorized by reactor vintage: BWR 1/2/3 plants with isolation condensers, BWR3/4 plants with reactor core isolation cooling (RCIC) systems, and BWR 5/6 plants. The pressurized water reactor (PWR) plants were categorized by reactor vendor, with the Westinghouse plants further subdivided according to the number of reactor coolant loops. The PWR groups are Babcock & Wilcox (B&W), Combustion Engineering (CE), Westinghouse 2-loop, Westinghouse 3-loop, and Westinghouse 4-loop.

To provide a common basis for comparison, the IPE results were examined for the following accident classes: station blackout accidents (SBO), anticipated transients without scram (ATWS), transients, loss-of-coolant accidents (LOCAs), internal floods, interfacing system LOCAs (ISLOCAs), and steam generator tube ruptures (SGTRs) for PWRs. For the BWRs, the contributions from transient sequences with loss of decay heat removal (DHR) were considered separately from transient sequences with loss of coolant injection. This distinction was not made for the PWRs because DHR sequences are relatively low contributors to transients in PWRs, on average.

II. GENERAL CDF PERSPECTIVES

Consistent with the results of previous NRC and industry risk studies, the IPEs indicate that the plant CDF is determined by a collection of many different sequences, rather than being dominated by a single sequence or failure mechanism. The accident class that is the largest contributor to plant CDF and the dominant failures contributing to that accident class vary considerably among the plants (e.g., some are dominated by LOCAs while others are dominated by station blackout). However, for most of the plants, support systems are important to the results because support system failures can result in failures of multiple front-line systems. The support system designs and the dependencies of front-line systems on support systems vary considerably among the plants, which explains much of the variability in the IPE results. This variability is consistent with the perspectives of the Severe Accident Policy Statement, that is,

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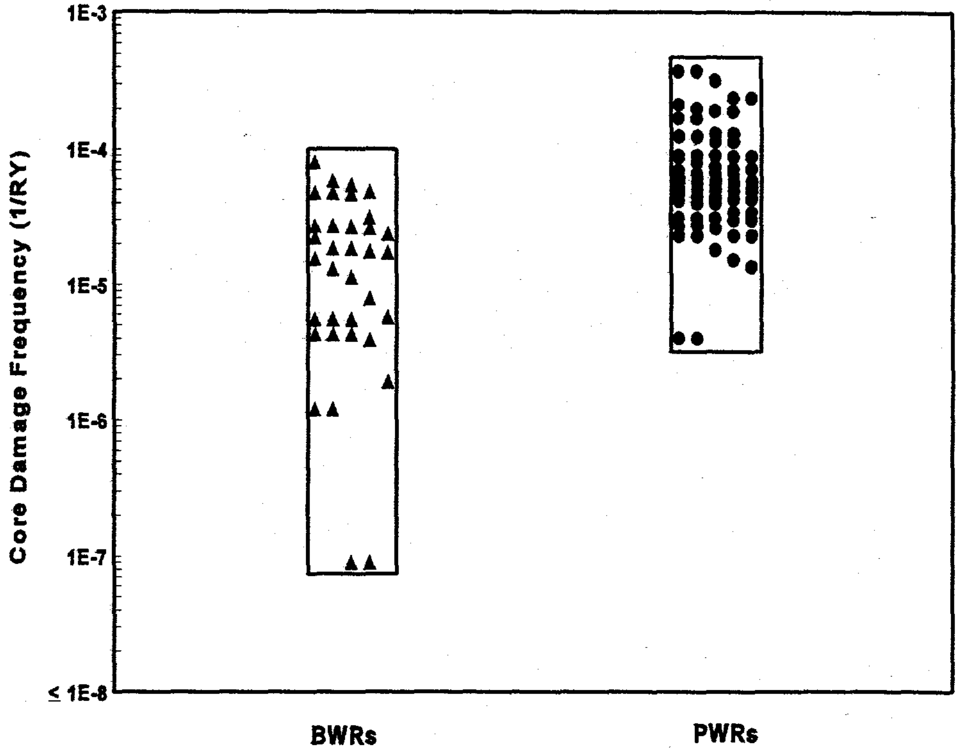

Figure 1 BWR and PWR CDFs as reported in the IPEs

that plant-specific factors are important in determining the risk for the various light water reactor (LWR) plants.

The CDFs reported in the IPE submittals are lower on average for the BWR plants than for the PWR plants, as shown in Figure 1. Although the BWR and PWR results are strongly affected by the support system considerations discussed above, there are a few key differences among the plant types that cause this tendency for lower BWR CDFs. BWRs have more injection systems and can depressurize more easily than PWRs to use low pressure injection systems. This results in a lower average contribution from LOCAs for BWRs. Most PWRs can remove decay heat during transients either through the steam generators or by using primary system feed and bleed, which gives considerable redundancy for coping with transient sequences. However, if a LOCA is induced during a transient (e.g., reactor coolant pump seal LOCA or stuck-open relief valve), injection is needed to maintain the reactor coolant system inventory. This is not as significant a problem for most BWRs because the normal means of decay heat removal is through injection systems, and as noted above, BWRs have more injection systems available than PWRs. However, many BWRs are more susceptible to transients with loss of containment heat removal because the sequence results in an adverse environment that fails emergency core coolant system (ECCS) pumps and other injection systems. This type of transient sequence is not generally important for PWRs. Station blackout sequences tend to be important contributors for both PWR and BWR plant groups because they result in the unavailability of numerous systems, leaving relatively few systems available to respond to the accident.

The results for some of the individual plants vary from the general trends noted above for some plants. As shown in Figure 1, there is considerable variability in CDFs within the BWR and PWR plant groups, which results in considerable overlap between the CDFs of the PWR and BWR plants. The variability is driven by a combination of factors, including plant design differences (primarily in support systems such as cooling water, electrical power, ventilation, and air systems), variability in modeling assumptions (including whether the models accounted for alternate accident mitigating systems), and differences in data values (including human error probabilities) used in quantifying the models. A summary of the key observations regarding the importance and variability of each accident sequence is provided in Table 1. Further details are provided in Sections III and IV for BWRs and PWRs, respectively.

Table 1 Overview of key IPE observations for LWRs

Accident Class	Key Observations
Transients	Important contributor for most plants because of reliance on support systems whose failure can defeat redundancy in front-line systems Both plant-specific design differences and IPE modeling assumptions contribute to variability in results. Major factors are: • capability to use alternate injection systems for BWRs • capability to use feed & bleed cooling and susceptibility to RCP seal LOCAs for PWRs
Station Blackouts	Significant contributor for most plants, with variability driven by: • number of emergency AC power sources • alternate offsite power sources • battery life • availability of firewater as injection sources for BWRs • susceptibility to reactor coolant pump seal LOCAs for PWRs
LOCAs	LOCAs are significant contributors for many PWRs BWRs generally have lower LOCA CDFs than PWRs • BWRs have more injection systems • BWRs can depressurize more readily to use low-pressure systems
Internal Floods	Small contributor for most plants, but significant for some because of plant-specific designs Largest contributors involve water system breaks that fail multiple mitigating systems (directly or through flooding effects)
ATWS	Normally a low contributor to plant CDF because of reliable scram function and successful operator responses BWR variability mostly driven by modeling of human errors; PWR variability mostly driven by plant operating characteristics and IPE modeling assumptions
Bypass Sequences	ISLOCAs are a small contributor to plant CDF for BWRs and PWRs because of low frequency of initiator Steam generator tube rupture normally a small contributor to CDF for PWRs because of opportunities for operator to isolate break and terminate accident

III. BOILING WATER REACTOR PERSPECTIVES

The total CDFs for all operating BWRs in each of the BWR plant groups are shown in Figure 2. With the exception of a few outliers, the total CDFs for most BWRs fall within an order of magnitude range. The variability in the results is attributed to a combination of factors, including plant design differences, especially in support systems such as electrical power, cooling water, ventilation, and instrument air systems; modeling assumptions; and differences in data values, including human error probabilities. The largest variation exists in the BWR 3/4 group, which is the group with the largest number of plants. Variability in plant design and modeling assumptions results in several plants in the BWR3/4 group having CDFs below the remaining BWRs, and one plant (2 units) considerably below the others. Significantly smaller variability in the total plant CDFs was found for the other two BWR plant groups. A summary of the importance of the various accident classes to the BWR CDFs and the factors influencing the results is provided in Table 2.

A large variability exists for each BWR group in the contributions of the different accident classes to the total plant CDF. However, licensees in all three BWR groups generally found that three types of accidents are the major

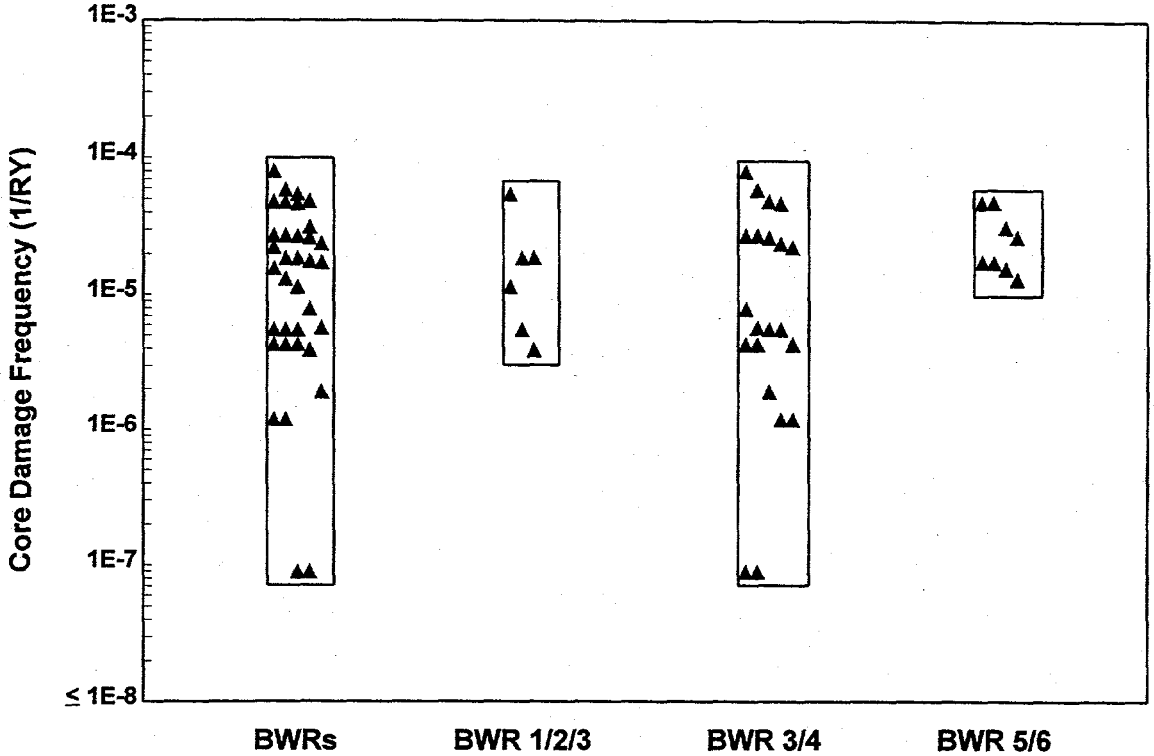

Figure 2 BWR plant group CDFs as reported in the IPEs

contributors to the total plant CDF: station blackouts, transients with loss of coolant injection, and transients with loss of DHR. These three accident categories involve accident initiators and/or subsequent system failures that defeat the redundancy in systems available to mitigate potential accidents. Station blackouts involve a loss of both offsite and emergency onsite power sources (primarily diesel generators but a few plants also have gas turbine generators) that fail most available mitigating systems except those that do not rely on AC power (the definition of station blackout for BWR 5/6s does not include failure of the diesel generator supplying the HPCS system). Most of the accident sequences contributing to the transients with loss of coolant injection category involve the failure of high-pressure injection systems such as feedwater, RCIC, high pressure coolant injection (HPCI), and high pressure core spray (HPCS), with a subsequent failure to depressurize the plant for injection by low-pressure injection systems. The failure to depressurize effectively defeats a large part of the redundancy in the coolant injection systems. Support system failures (e.g., loss of cooling water systems, AC or DC buses, or instrument air) that impact many of the available accident mitigating systems contribute to the importance of this accident category and also to the transient with loss of DHR category. In all loss of DHR sequences involving transient or other initiators, redundancy in mitigating systems can be lost due to harsh environments in the containment prior to containment failure or in supporting structures following containment venting or failure.

Lesser contributions from LOCAs, ATWS, and internal flooding are generally reported for all BWRs. These three accident categories are not important contributors primarily because they involve low frequency initiating events. However, there are a few BWRs that did report significant contributions from these accident categories. Although interfacing system LOCAs are potentially risk-significant contributors since the containment is bypassed, none of the licensees reported significant CDFs from this category of accident, again primarily because it involves low-frequency initiating events.

Many of the factors that impact the CDF contributions from these accident categories are the same for each plant group. However, there are factors worth highlighting that explain some of the differences across the BWR groups. For example, it was noted that some of the accident class frequencies for the BWR 1/2/3 plant group are generally lower than for the other two BWR plant groups, partially because isolation condensers appear to be more reliable than the RCIC systems that replaced them in the later BWR models. RCIC systems have more possible failure modes related to protective trip signals, ventilation failures, and pump operability requirements. Some of these failure modes

Table 2 Summary of CDF perspectives for BWRs

Accident Importance	Important Design Features, Operator Actions, and Model Assumptions
Station blackout accidents	
Important for most BWRs, regardless of plant group	Availability of AC-independent systems (i.e., high-pressure coolant injection system, diesel-driven firewater system, reactor core isolation cooling interface with suppression pool) Turbine bypass and isolation condenser capacity Battery life DC dependency for diesel generator startup Service water system design and heating, ventilating and air conditioning dependency AC power reliability (number of diesel generators, cross-tie capability between buses and units, diverse AC power sources)
Transients with loss of injection accidents	
Relatively unimportant at BWR 1/2/3 plants Important for most BWR 3/4 and 5/6 plants	Injection system dependencies on support systems, defeating redundancy Availability and redundancy of injection systems (e.g., control rod drive, motor-driven feedwater pumps, service cross-tie to residual heat removal, firewater system) Failure to depressurize influenced by operator direction to inhibit the automatic depressurization system
Transients with loss of decay heat removal accidents	
Important for most BWRs, regardless of plant group	Limited analysis to support success criteria — no credit for decay heat removal system (e.g., venting) Dependency of support systems for decay heat removal Net positive suction head problems with emergency core cooling systems on suppression pool Availability of injection system located outside containment and reactor building Capability of emergency core cooling systems to pump saturated water
Anticipated transient without scram accidents	
Relatively unimportant for most BWRs, regardless of plant group	Operator failure to initiate standby liquid control in timely manner, maintain main steam isolation valves open, control vessel level, and/or maintain pressure control Use of alternate means of injecting boron Availability of high-pressure core spray to mitigate
Loss-of-coolant accidents	
Relatively unimportant at all but one of the BWR plants	High redundancy and diversity in coolant injection systems
Interfacing systems LOCAs	
Not important for BWR plants	Compartmentalization and separation of equipment
Internal flood accidents	
Relatively unimportant at most BWRs, regardless of plant group	Plant layout: separation of mitigating system components and compartmentalization

are only prevalent in the BWR 5/6 IPEs and partially account for the higher station blackout CDFs for this group. However, it should be noted that some of the licensees with isolation condenser plants generally ignored the potential for recirculation pump seal failures, which would effectively defeat the use of the isolation condensers. Finally, the BWR 5/6 plants had lower contributions on average from sequences involving loss of high-pressure injection systems coupled with failure to depressurize the vessel for low-pressure injection than BWR 3/4s since the HPCS system in the BWR 5/6 plants tends to be more reliable than the HPCI system in the BWR 3/4 plants.

IV. PRESSURIZED WATER REACTOR PERSPECTIVES

There is generally a larger variability in plant CDFs within the individual PWR plant groups than among plant groups. The Westinghouse 3-loop plants generally have the highest CDFs, and the B&W plants generally have the lowest CDFs, with the CDFs for most of the B&W plants falling below the CDFs for the Westinghouse 3-loop plants. However, the difference in average CDFs between these two plant groups is about the same as the variability within either of the two plant groups. The variability in the PWR results is attributed to a combination of factors, including plant design differences (especially in support systems such as electrical power, cooling water, ventilation, and instrument air systems), modeling assumptions, and differences in data values (including human error probabilities). The largest variation exists in the Westinghouse 4-loop plant group, which is the group with the largest number of plants, but the other plant groups also show considerable variability. The Combustion Engineering plant group contains a 2-unit plant with a CDF well above the other plants in the group while the Westinghouse 4-loop plant group contains a 2-unit plant with a CDF considerably below the other plants in the Westinghouse 4-loop group. Figure 3 presents the CDF for the different PWR plant groups.

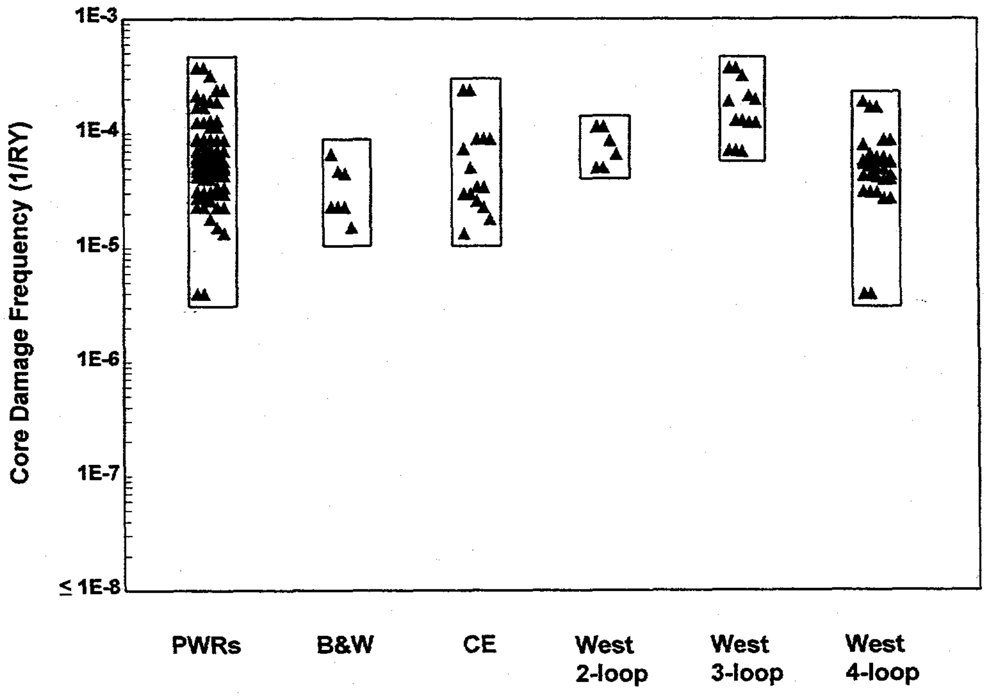

Figure 3 PWR plant group CDFs as reported in the IPEs

A summary of the importance of the various accident classes for the PWR CDFs and the factors driving variability in the results is provided in Table 3. Considerable variability exists for each PWR group in the contributions of the different accident classes to the total plant CDF. However, licensees in all five PWR groups generally find that three types of accidents are the major contributors to the total plant CDF: transients, loss-of-coolant accidents, and station blackout. These three accident classes involve accident initiators and/or subsequent system failures that defeat the redundancy in systems available to mitigate potential accidents. Lesser contributions are generally reported for ATWS, steam generator tube ruptures, interfacing system LOCAs, and internal flooding. However, a few PWRs do report significant contributions from these accident classes, and steam generator tube ruptures are found to be significant contributors for the Westinghouse 2-loop plants.

Some of the factors that have the largest influence on the CDF contributions reflect concerns that are more prevalent in a particular PWR plant group, but most reflect design differences or modeling assumptions that are applicable to all of the PWR plant groups. Differences that tend to reflect design differences among the PWR plant groups are summarized below.

One of the most important factors affecting PWR CDFs is the susceptibility to RCP seal LOCAs for transient and station blackout sequences. To prevent core damage in RCP seal LOCA sequences, inventory makeup is required in addition to core heat removal. Both the B&W and CE plant groups have less susceptibility to RCP seal LOCAs in the IPE models because most plants in these groups have a seal design that the industry believes to be less prone to seal damage. However, there is at least one plant in each group that has indicated a significant CDF contribution that involves RCP seal LOCAs. This lower susceptibility to RCP seal LOCAs in the B&W and CE IPEs tends to cause lower contributions from transient and station blackout sequences for the B&W and CE plants relative to the Westinghouse plants.

Because the probability of RCP seal LOCAs is generally lower in the B&W and CE IPEs, these plants tend to show more benefit than Westinghouse plants from plant characteristics that improve the reliability of heat removal through the steam generators (e.g., reliable or redundant feedwater pumps, sustained source of water for feedwater, or longer battery life for control of auxiliary feedwater during station blackout). The importance of these factors is less for many Westinghouse plants because RCP seal LOCAs lead to core damage despite the cooling provided through the steam generators.

Feed-and-bleed cooling is often an important backup for transient sequences with loss of steam generator heat removal. All but one of the B&W plants have high- pressure injection pumps with high shutoff heads that can provide adequate flow for feed-and-bleed cooling even at the safety relief valve setpoint. Some CE plants do not have power-operated relief valves (PORVs) or other means to depressurize. The inability to feed and bleed for these CE plants is generally compensated for by the ability to depressurize the steam generator and use condensate for cooling. Therefore, the lack of PORVs has less influence on the IPE results than might otherwise be expected.

The final factor that tends to show similarities within plant groups is the configuration for ECCS recirculation. Plants with a higher degree of automation in performing the switchover and plants that can achieve high-pressure recirculation with fewer components operating tend to have lower failure rates resulting from the switchover to recirculation. For the plants with manual switchover, variability in the assessment of operator performance in performing the action is also important. The B&W plants require manual actions for ECCS switchover from injection to recirculation, and the high-pressure injection pumps must draw suction from the low-pressure pumps to operate in the recirculation mode. The CE plants have automatic switchover, and the high-pressure pumps can draw water directly from the sump rather than drawing suction from the discharge of the low-pressure pumps. The Westinghouse plants are mixed on these factors. Some Westinghouse plants require operator actions to perform the switchover while other plants have automatic switchover. For some Westinghouse plants, the high-pressure pumps draw directly from the sump during recirculation, while at other plants the high-pressure pumps must be aligned to draw suction from the low-pressure pumps (which draw from the sump).

Table 3 Summary of CDF Perspectives for PWRs

Accident Importance	Important Design Features, Operator Actions, and Model Assumptions
Station blackout accidents	
Important for most PWRs	Susceptibility to RCP seal LOCAs (o-ring design, alternate cooling, and seal LOCA model) Redundancy in emergency AC power sources (e.g., number of diesel generators) Battery life Use of plant operating data indicating low frequencies for loss of offsite power and high reliability of emergency diesel generators
Loss-of-coolant accidents	
Important for most PWRs	Whether manual action required for switchover to recirculation Alternate actions to mitigate LOCA (e.g., depressurizing the reactor coolant system using the steam generator atmospheric dump valves when high pressure injection fails during LOCA) Size of refueling water storage tank
Transient accidents	
Important for most PWRs	Susceptibility to RCP seal LOCAs (pump design, seal cooling capabilities, seal LOCA model) Capability for feed-and-bleed cooling Ability to cross-tie between systems/units Dependence on support systems (component cooling water and/or service water systems, HVAC and instrument air) Ability to depressurize the steam generators and use condensate for heat removal Ability to supply long-term water to the suction for AFW/EFW
Anticipated transients without scram accidents	
Relatively unimportant for most PWRs	Ability to mitigate by pressure control, boration, and heat removal
Interfacing system LOCAs	
Relatively unimportant for PWRs	Compartmentalization and separation of equipment
Steam generator tube rupture accidents	
Relatively unimportant to CDF for most PWRs	Credit for operator actions and equipment used to mitigate accidents
Internal flood accidents	
Important for some PWRs	Plant layout: separation of mitigating system components and compartmentalization