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CORE DAMAGE FREQUENCY (REACTOR DESIGN) PERSPECTIVES BASED ON IPE RESULTS*

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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. Human actions found to be important in boiling water reactors (BWRs) and in pressurized water reactors (PWRs) are presented and the events most frequently found important are discussed.

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).

An important aspect of the IPE program is to identify human actions important to severe accident prevention and mitigation. In this context, the human reliability analysis (HRA) is expected to be a critical component of the probabilistic risk assessments (PRAs) for the IPEs. The determination and selection of human actions for incorporation into the event and fault tree models and the quantification of their failure probabilities can have an important impact on the resulting estimates of CDF. Thus the human actions important in the IPEs are summarized in this paper and the degree of variability in the results of the HRAs is addressed. Of particular concern is the degree of variability in the quantification of similar human actions across different plants.

Perspectives regarding factors that have the largest influence on the IPE results are provided in Sections II through IV. More specific perspectives for one of the key factors, HRA, are provided in Sections V through VIII.

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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 loss-of-coolant accidents (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, 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 for each of the individual LWR units are indicated by the dots in Figure1. As shown, the CDFs are lower on average for the BWR plants than for the PWR plants. 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.

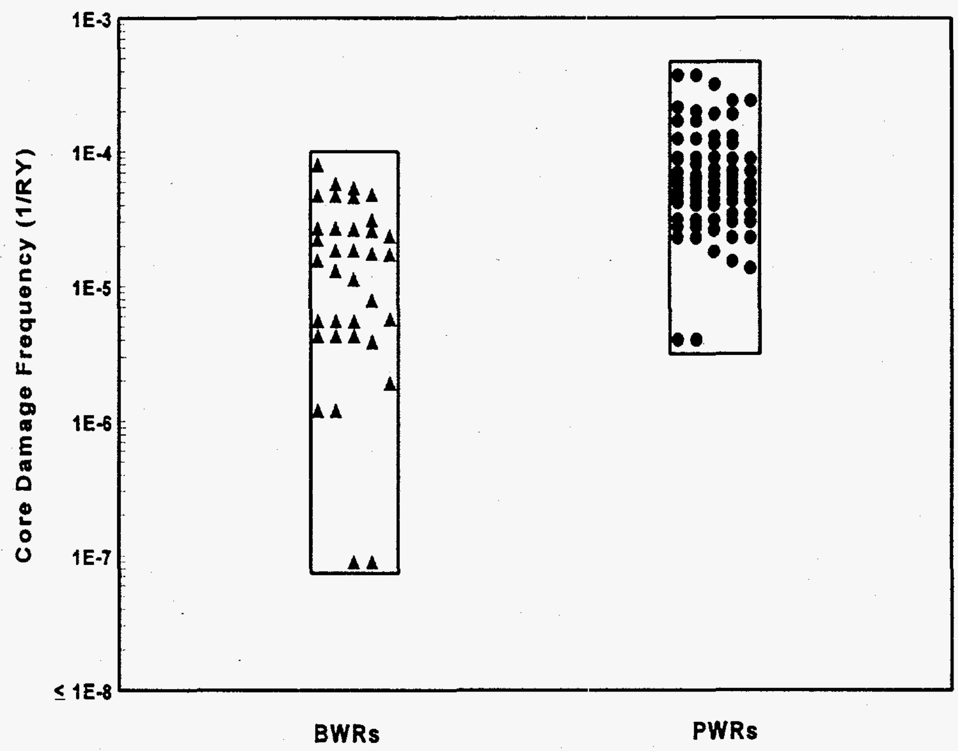

Figure 1 BWR and PWR CDFs as reported in the IPEs

However, if a LOCA is induced during a transient (e.g., reactor coolant pump (RCP) 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.

III. BOILING WATER REACTOR PERSPECTIVES

The total CDFs for all operating BWRs in each of the BWR plant groups (grouped by design vintage) 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 contributors to the total plant CDF: station blackouts, transients with loss of coolant injection, and transients with loss of decay heat removal (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 high pressure core spray (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 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.

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)
Anticipated Transient Without Scram (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	Interfacing System LOCAs (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

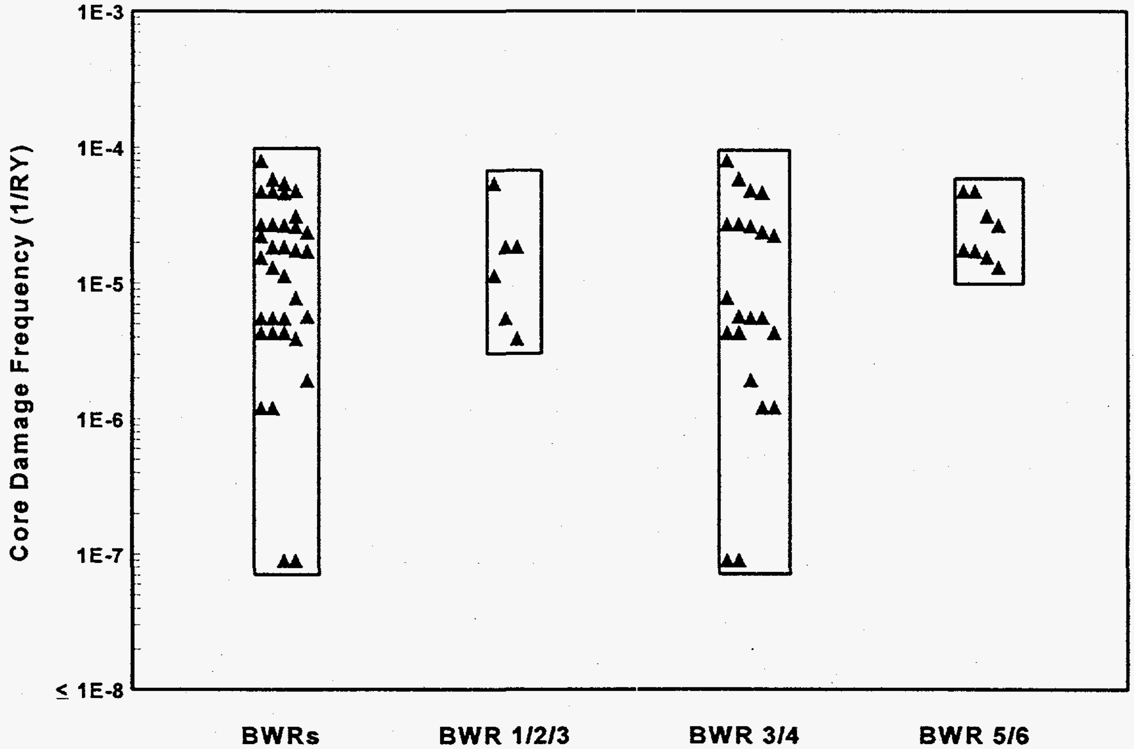

Figure 2 BWR plant group CDFs as reported in the IPEs

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 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.

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

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 (West) 3-loop plants generally have the highest CDFs, and the Babcock & Wilcox (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 (CE) 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 CDFs for the different PWR plant groups.

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, LOCAs, and station blackout. These three accident classes involve accident initiators and/or subsequent

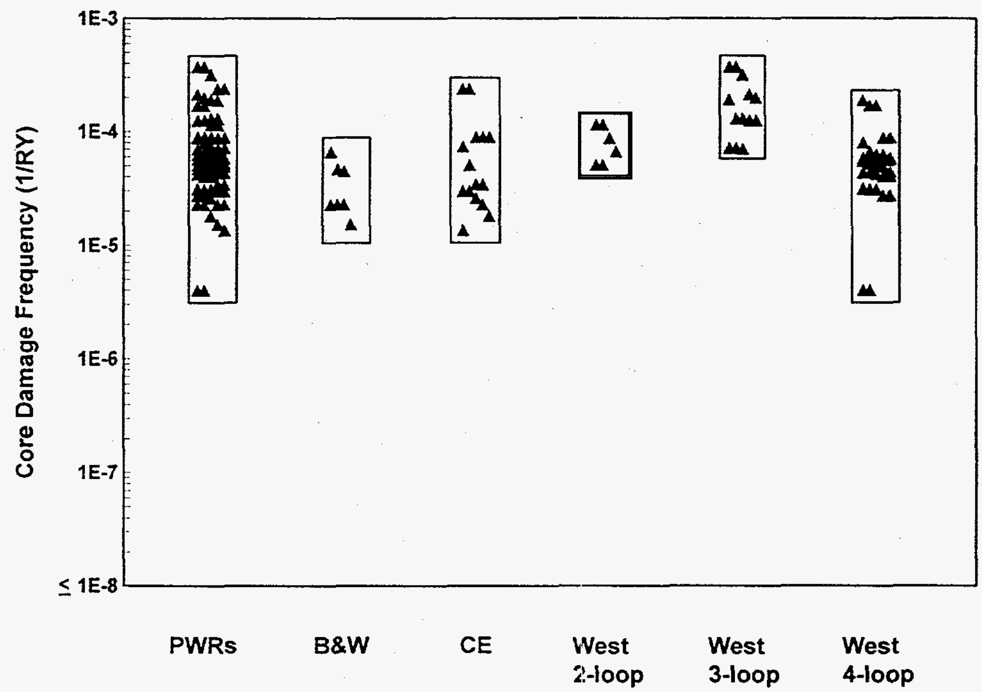

Figure 3 PWR plant group CDFs as reported in the IPEs

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, heating, ventilation and air conditioning (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 auxiliary feedwater/emergency feedwater (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

system failures that defeat the redundancy in systems available to mitigate potential accidents. Lesser contributions are generally reported for ATWS, steam generator tube ruptures, ISLOCAs, 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).

V. HUMAN ACTIONS GENERALLY IMPORTANT FOR BWRs

Table 4 lists the most important human actions identified in the staff's review of all 27 BWR IPE submittals (covering 35 units), along with the percentage of all BWR IPEs finding the action important, and the percentage of IPEs finding the action important as a function of BWR class. Of the 27 submittals reviewed, five are in the BWR 1/2/3 class (covering six units), 15 are in the BWR 3/4 class (covering 21 units), and seven are in the BWR 5/6 class (covering eight units).

Only a few specific human actions are regularly found to be important across all the BWR IPEs. That is, while many different events are indicated as being important, relatively few are important to most of the IPEs. Thus, the staff attempted to group the operator actions according to the function to be accomplished. For example, events related to aligning an alternative injection source during transients, LOCAs, and station blackouts (SBOs) are considered important to several licensees. Even though the alternative systems used ranged from firewater to suppression pool cleanup, the function accomplished by performing the action is similar. In order to help capture the general types of events that are important to BWRs, the staff grouped these actions with similar functions and presented them in Table 4 along with other important individual operator actions.

Manual depressurization of the vessel[1] so that low-pressure injection systems can be used after a loss or unavailability of high-pressure injection systems is important in most BWR IPE submittals. This action is particularly important in some plants for long-term SBO sequences where depressurization is required to allow injection from firewater systems, after loss of steam-driven systems such as reactor core isolation cooling (RCIC). This human action is important largely because of the fact that most plant operators are directed to inhibit automatic actuation of the ADS by the plant emergency operating procedures (EOPs). Thus, operators must manually depressurize the vessel when injection from low-pressure systems is required to cool the core. The percentage of total CDF accounted for by cutsets including this event ranged from 1 to 44%.

While human actions related to an ATWS are frequently found in the licensees' lists of the top ten important events, the contribution of ATWS events to overall CDF is usually relatively small. The human action to inhibit the ADS is important in the ATWS sequences of several submittals. In fact, some licensees assume that because of the instabilities created under low-pressure conditions during an ATWS, core damage will occur if the operators fail to inhibit the ADS. Given this position, it is somewhat surprising to find that only ~20% of the BWR licensees identify inhibition of the ADS as being important. The low percentage results in part from how licensees model ADS inhibition. Many licensees assume that failure to perform this action has a very low probability, or they do not model it at all. Other licensees model the failure to inhibit the ADS as resulting in core damage only if it occurs in conjunction with a second failure (e.g., failure of SLC or failure of low-pressure injection flow control). Such a model can reduce the importance of this type of accident sequence and thus the importance of the related human errors. The remaining licensees model the failure to inhibit the ADS during an ATWS as directly resulting in core damage. This human error is noted as being important for approximately 50% of the licensees that model ADS inhibition in any fashion.

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[1]Section VII discusses the variability in HEPs for this event across the BWR IPEs.

Table 4 Important human actions and percentage of BWR IPEs finding the action important

Important human actions	Percentage of BWR IPEs finding the action important			
	All BWR IPEs	BWR 1/2/3s	BWR 3/4s	BWR 5/6s
Perform manual depressurization	~80%	~80%	~80%	~60%
Containment venting	~55%	~35%	~60%	~60%
Align containment or suppression pool cooling	~55%	~70%	~50%	~50%
Initiate standby liquid control (SLC)	~50%	~70%	~50%	~40%
Level control in ATWS	~25%	~50%	~30%	0%
Align/initiate alternative injection	~25%	~30%	~30%	~15%
Recover ultimate heat sink	~20%	~20%	~20%	~25%
Inhibit automatic depressurization system (ADS)	~20%	~20%	~20%	~25%
Miscalibrate pressure switches	~15%	~20%	~15%	~10%
Initiate isolation condenser	N/A	~85%	N/A	N/A
Control feedwater events (e.g., loss of instrument air)	~15%	~15%	~20%	~15%
Manually initiate core spray or other low-pressure system	~15%	~20%	~20%	0%
Miscalibrate low-pressure core spray permissive	~10%	~20%	~15%	0%
Provide alternative room cooling (in the event of a loss of HVAC)	~10%	0%	~5%	~25%
Recover injection systems	~10%	0%	~15%	~15%

Two other ATWS-related events are found to be important by several licensees. The operator action to initiate boron injection during an ATWS is important in ~50% of the BWRs, and ~25% identify level control as being important. As with ADS inhibition, the modeling of these events partially impacts their importance to core damage. For example, some licensees model early initiation of SLC, while others consider both early and late initiation times. The initiation times (important in calculating the HEPs) are based on avoiding adverse conditions, such as high suppression pool temperatures, and are somewhat variable (ranging from one minute to 45 minutes). Some licensees take credit for alternative means of injecting boron, while others take credit for level control as a means of reducing core power to acceptable levels following SLC failure. All of these variables can contribute to the importance of the failure to manually initiate SLC. Modeling of level control is highly variable, with several different factors influencing the modeling. Whether these actions are important for particular licensees is, to some extent, a function of the contribution of the ATWS sequences to overall CDF. The contribution of these events to CDF is usually in the range of 1 to 3 %.

Many licensees identify human actions related to decay heat removal as being important. Two of the most frequently identified important actions in BWRs relate to decay heat removal (DHR) sequences in transients and LOCAs. With a loss of the power conversion system and safety relief valves (SRVs) open, containment temperature and pressure must be controlled. The actions to provide some form of containment or suppression pool cooling, or to vent containment when adequate cooling can not be provided, are important in more than 50% of the IPE submittals. Plant characteristics and modeling differences are important factors in determining the impact of these human actions.

Plants require DHR actuation before adverse conditions are reached. These conditions can range from reaching a high suppression pool temperature that results in a loss of emergency core coolant system (ECCS) pumps, to reaching a high containment pressure that results in closure of SRVs that are required to remain open to maintain the vessel at low-pressure (for coolant injection from low-pressure systems). However, some licensees did not model the failure of DHR as leading to a failure in the ability to inject water into the vessel from the ECCS or from alternative injection systems. In addition, some licensees identified the steam released following containment failure as having a negative impact on the operability of injection systems. In addition, some licensees do not model venting at all. They either do not have reliable venting systems, do not have a strong need to vent, or simply do not take credit for venting. The contribution from these events to CDF generally ranges from 1 to 5 %, with one licensee indicating a 12% contribution.

VI. HUMAN ACTIONS GENERALLY IMPORTANT FOR PWRs

Table 5 lists the most important human actions identified in the staff's review of all 48 PWR IPEs submittals, along with the percentage of all PWR submittals finding the action important, and the percentage of submittals finding the action important as a function of PWR class.

As with BWRs, only a few human actions are regularly found to be important across all PWR submittals. The human action most consistently important for PWRs is the switchover to recirculation during LOCAs. Other human actions frequently important include feed and bleed, and actions associated with depressurization and cooldown. Only these three actions are important in more than 50% of the submittals. They are discussed in more detail below, along with several other actions frequently found to be important by the licensees.

Switchover to recirculation on low ECCS level is important for LOCA sequences in most submittals for plants with semi-automatic or manual switchover. All ten CE plants (15 units) have an automatic switchover, as do four of the other plants. For the 35 plants (58 units) that require operator actions (either completely manual

Table 5 Important human actions and percentage of PWR submittals finding action important

Important human actions	Percentage of IPEs finding event important					
	All PWRs	B&W	CE	West 2-loop	West 3-loop	West 4-loop
Switchover to recirculation (plants with manual or semi-automatic switchover)	~80%	~85%	N/A	100%	~55%	~90%
Feed-and-bleed	~60%	~45%	~60%	~70%	~45%	~70%
Depressurization and cooldown	~50%	~60%	~30%	100%	~70%	~50%
Use of backup cooling water systems	~40%	~45%	~30%	~35%	~60%	~30%
Makeup to tanks for water supply	~35%	~30%	~20%	~35%	~40%	~40%
Restoration of room cooling (HVAC)	~30%	~15%	~50%	~35%	~30%	~30%
Restoration of main feedwater (MFW) or condensate to steam generators (SGs)	~30%	~30%	~35%	~35%	~50%	~30%
Proper control of AFW or EFW	~25%	~30%	~40%	~35%	0%	~30%
RCP Trips	~25%	~45%	~35%	~35%	~15%	~20%
Pre-initiators	~25%	0%	~50%	0%	~25%	~20%
ATWS reactivity control	~20%	0%	~20%	0%	~10%	~35%
Water supply for AFW or EFW	~15%	0%	~40%	~35%	~10%	~5%
Initiation of AFW or EFW	~15%	0%	~50%	0%	~10%	~10%

or semi-automatic) to complete the switchover, ~80% of the submittals find this action to be important. One possible reason some licensees fail to find this action important may be the fact that the sizes of refueling water storage tanks (RWSTs) vary from plant to plant. Licensees with plants that have larger RWST capacities may model the small LOCA and long-term transient sequences as not requiring the switchover to recirculation cooling, thereby lessening the importance of the recirculation function and hence human actions related to recirculation cooling. Additionally, some licensees model RWST refill as the action preferred over recirculation cooling, particularly in small LOCA and long-term transient cooling situations. This again lessens the overall importance of recirculation cooling and the corresponding related human actions. For licensees that find the switchover to recirculation to be an important operation (and report the related contribution to total CDF), the contribution to CDF ranges from less than 1% in several cases to as much as ~16%, with an average contribution of ~6%.

Many licensees identify the initiation of the feed-and-bleed operation as being important. This event is important in transient and steam generator tube rupture (SGTR) sequences when all feedwater has failed. In addition, a few licensees find the establishment of an reactor coolant system (RCS) bleed path with one power operated relief valve (PORV) to be important in small LOCAs. In all, about 60% of the submittals indicate that feed-and-bleed is one of the more important events. Some licensees may fail to find feed-and-bleed important for a variety of reasons that are interrelated and not easily discernible. For instance, the relative reliability of each plant's AFW or EFW system is a factor since it is only in sequences where AFW or EFW has failed that feed-and-bleed becomes another important action in the in-depth defense to provide core cooling. Thus, accident sequences involving AFW/EFW failure (and thus the need to use the feed-and-bleed function) can vary considerably in frequency, thereby affecting the overall importance of the feed-and-bleed function. Specific support system dependencies can also be important to the overall feed-and-bleed reliability and hence the importance of this human action. For plants with a higher susceptibility of failing feed-and-bleed because of support system failures, this mode of cooling is less reliable, and the human action of feed-and-bleed operation can be less important.

Additionally, many licensees spent considerable effort to model the ability to depressurize the plant and use condensate as yet another way to achieve core cooling. Taking credit for such action further lessens the overall importance of feed-and-bleed function and the related human action. Other factors related to the success criteria for feed-and-bleed, as well as the HEPs themselves, can contribute to the relative importance of this mode of cooling and the related human action. The CDF contribution for this event ranges from less than 1% to 11%, with most submittals showing relatively small contributions from this event, resulting in an average total CDF contribution of about 4%.

The depressurization and cooldown operation, in order to use available sources of core cooling (and in many cases to lessen SGTR leakage), is found to be important by more than half of the licensees. This action usually (but not always) involves depressurizing the steam generators to cool the RCS and is found to be important in all types of sequences except ATWS. It is most frequently deemed important in SGTR sequences. As a result, 52% of the licensees find this human action important. As discussed above regarding the feed-and-bleed function, licensees may neglect to find depressurization and cooldown important for numerous interrelated reasons (including those described for the feed-and-bleed event). Additionally, not all of the plants model this mode of cooling, in some cases because of the relatively low capacity to depressurize the SGs in some scenarios (depending on PORV, atmospheric dump valve, or other equipment sizes). The CDF contribution for this event ranges from less than 1% to ~7%, and is similar to feed-and-bleed. Most submittals show relatively small contributions from this event, resulting in an average total CDF contribution of approximately 4%.

None of the remaining human actions are important in more than 40% of the submittals, and none of them consistently contributes significantly to CDF. As shown in Table 5, the remaining human actions are not important in a large percentage of the submittals. Recovering and using backup cooling systems, supplying makeup for injection sources, and recovering loss of room cooling are important for accident sequences in approximately one-third of the submittals. Several actions related to restoration and appropriate use of MFW and AFW systems are found to be important in several submittals, and RCP trips upon loss of seal cooling is important in about 25% of the submittals. Similar to the BWRs, pre-initiator events, including both miscalibration and restoration errors, are found important in some submittals. The miscalibration errors tend to involve the traditional instruments such as level, pressure, and temperature sensors and transmitters, but the restoration errors tend to vary across submittals. Examples of important restoration errors include those associated with AFW and EFW systems, diesel generators, and several unique events such as leaving a nitrogen station manual valve closed and removing a jumper in the reactor protection system after refueling.

VII. VARIABILITY IN HUMAN ERROR PROBABILITIES

Numerous factors can influence the quantification of HEPs and introduce significant variability in the resulting HEPs, even for essentially identical actions. General categories of such factors include plant characteristics, modeling details, sequence-specific attributes (e.g., patterns of successes and failures in a given sequence), dependencies, HRA method and associated performance shaping factors (PSFs) modeled, application of HRA method (correctness and thoroughness), and the biases of both the analysts performing the HRA and the plant personnel from whom selected information and judgments are obtained. Although most of these factors introduce appropriate variability in results (i.e., the derived HEPs reflect "real" differences such as time availability and scenario-specific factors), several have the potential to cause invalid variability. A discussion of both appropriate and inappropriate influences is presented below, followed by a discussion of the variability in the HEPs for a specific event.

In order to examine the variability in HRA results from the IPEs and to assess the extent to which variability in results is caused by real versus artifactual differences, the staff examined HEPs from several of the more important human actions appearing in the submittals across plants. However, since the staff reached the same general conclusion after examining several important human actions for the BWRs and PWRs, this summary report presents the results from the examination of a single important human action. Discussions of the variability in HEPs for several other human actions from BWRs and PWRs are presented in the body of the main report.

Figure 4 presents the HEPs used in various BWR submittals for failure to depressurize the vessel during transients. As shown in the figure, a relatively large variability exists across the submittals for this event. However, there appears to be reasonable explanations for much of the variability in the HEPs. For values on the high end of the continuum, the events modeled appear to be special cases of depressurization. For example, the high value for Nine Mile Point 1 (N-1) involves depressurization using main steam isolation valves and the condenser, which is apparently not typically modeled. The high value for Peach Bottom 2&3 (PB) and the next to the highest value for Limerick 1&2 (LIM) pertain to the case in which a controlled depressurization is needed to allow use of the condensate system. The highest value for Limerick 1&2 (LIM) pertains to a recovery of a failed automatic depressurization. While the justification for the high values for Big Rock Point (BRP) is not apparent, it is unique relative to the other BWRs in that the plant has some characteristics similar to PWRs. The reason for the high value for Cooper (COP) is also not obvious, but the large range of values for that plant apparently relates to the number of SRVs to be used for depressurization.

The explanations for the large difference (approximately one and one-half to two orders of magnitude) between the HEP values in the middle range appear to be related, at least in part, to dependencies and initiator- and sequence-specific factors. Several licensees, such as Nine Mile Point 1 (N-1), Dresden 2&3 (DRE), Fermi 2 (FER), and Limerick 1&2 (LIM), conducted relatively detailed analyses and apparently derived multiple values in order to account for specific conditions. These specific conditions include LOOPs, SBOs, loss of DC power, use of turbine bypass valves for depressurization, and loss of feedwater and standby feedwater. Nevertheless, while much of the variability in the middle range of values is clearly explainable, some differences are less clear. For example, the generally lower values for Fermi 2 (FER) and Limerick 1&2 (LIM) relative to those from Nine Mile Point 1 (N-1) and Dresden 2&3 (DRE) are not explainable in a straightforward manner, but may very well result from valid, plant-specific characteristics.

Finally, the reasons for the relatively low HEP values at Cooper (COP), Duane Arnold (DA), Fitzpatrick (FIT), Vermont Yankee (VY), and Susquehanna 1&2 (SUS) are not clear. It can be argued that at least the top three or four values from these submittals fall within an acceptable range. It may also very well be the case that plant-specific characteristics support the HEPs on the lower end of the continuum. For example, the relatively low value for Cooper (COP) is for a long-term DHR sequence in which operators have up to 4 hours to depressurize. The lowest value, from Susquehanna (SUS), is clearly an outlier, but this value is consistent with many of that plant's HEP values and is a direct function of the HRA methodology used in the Susquehanna IPE.

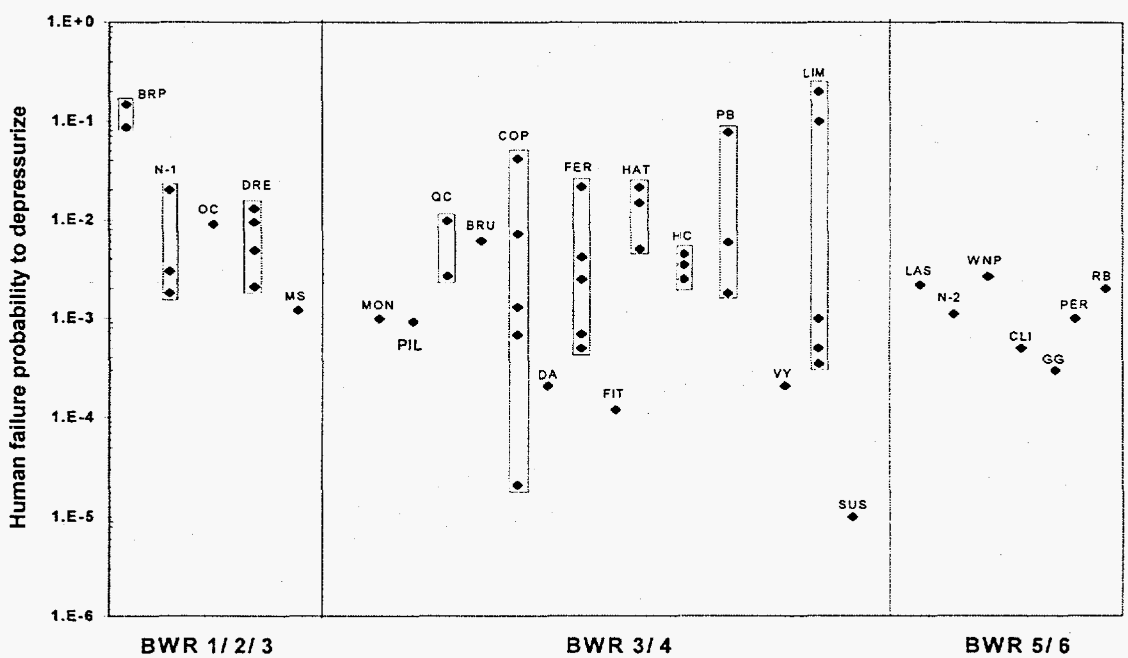

Figure 4 HEPs for depressurization failure by BWR class.[2]

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[2] HEPs shown on figure are on a submittal basis, and not necessarily a plant-unit basis.

At least some of the variability in HEP values can arise as an artifact of the way in which HRA methods are applied. Nonetheless, the main point to be derived from examining the HEPs for specific actions across plants, is that, in most cases, it also appears that there are reasonable explanations for much of the variability in HEPs and in the results of the HRAs across the different IPEs. However, such an assertion does not necessarily imply that the HEP values are generally valid. Reasonable consistency can be obtained in HRA without necessarily producing valid HEPs. An HEP is only valid to the extent that a correct and thorough application of HRA principles has occurred. For example, if a licensee simply assumes (without adequate analysis) that their plant is "average" in terms of many of the relevant PSFs for a given event, but appropriately considers the time available for the event in a given context, the value obtained for that event may be similar to those obtained for other plants. Yet, the resulting value may be optimistic or pessimistic relative to the value that would have been obtained if the licensee had conducted a detailed examination of the relevant plant-specific factors. Thus, to reiterate, consistency does not necessarily imply validity. In addition, because many of the licensees failed to perform high-quality HRAs, it is possible that the licensees obtained HEP values that are not appropriate for their plants.

VIII. SIMILARITIES AND DIFFERENCES IN HUMAN ACTION OBSERVATIONS ACROSS BWRs AND PWRs

Given the basic differences between BWRs and PWRs, the preceding discussion has for the most part provided separate observations regarding the submittals for the two different plant types. Nevertheless, the obvious commonalities across the plant types, prompt an examination of potential similarities or differences in the operational and HRA-related observations:

- Neither BWR nor PWR submittals show a broad consistency in terms of which human actions are found to be important. Given the numerous factors that can influence the IPE results, and the fact that functional redundancy creates the opportunity for quite a few operator actions to be taken to mitigate an accident scenario in both BWRs and PWRs, there is no reason to expect more consistency in what is found to be important for one type of plant as opposed to the other.

- Of the events frequently found to be important in BWRs and PWRs, the only similar actions are those related to depressurization and cool down.

- Events related to aligning or recovering backup cooling water systems (e.g., service water) are found to be important in approximately one-third of both BWRs and PWRs.

- In both BWRs and PWRs, no individual human action appears to account for a large percentage of the total CDF across multiple submittals. Taken together, however, human actions are clearly important contributors to operational safety.

- With the exception of the licensees using the IPE Partnership (IPEP) methodology, there is no indication that particular HRA methods are applied more frequently to one type of plant than another. Thus, except for the IPEP plants, there is no reason to expect that any general differences in the results of the PRAs for the two different plant types is related to HRA method (or to any of the more general influencing factors). The IPEP methods are primarily applied to PWRs.

In summary, it seems that most of the differences in the HRA results from the BWR and PWR submittals relate (not surprisingly) to the differences in the systems used in the two types of plants. In terms of more methodological aspects, general patterns of results, and the overall importance of humans in operating the plants, BWRs and PWRs are reasonably similar.