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CORE DAMAGE FREQUENCY PERSPECTIVES FOR BWR 3/4 AND WESTINGHOUSE 4-LOOP PLANTS BASED ON IPE RESULTS

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This paper discusses the core damage frequency (CDF) insights gained by analyzing the results of the Individual Plant Examinations (IPEs) for two groups of plants: boiling water reactor (BWR) 3/4 plants with Reactor Core Isolation Cooling systems, and Westinghouse 4-loop plants. Wide variability was observed for the plant CDFs and for the CDFs of the contributing accident classes. On average, transients with loss of injection, station blackout sequences, and transients with loss of decay heat removal are important contributors for the BWR 3/4 plants, while transients, station blackout sequences, and loss-of-coolant accidents are important for the Westinghouse 4-loop plants. The key factors that contribute to the variability in the results are discussed. The results are often driven by plant-specific design and operational characteristics, but differences in modeling approaches are also important for some accident classes.

The U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter 88-20 in November 1988 requesting that all licensees perform an Individual Plant Examination (IPE) to identify any plant-specific vulnerabilities to severe accidents, and to report the results to the Commission. The scope of the IPE effort includes examination of internal events, including those initiated by internal flooding, occurring at full power. A memorandum from the Executive Director of Operations to the Office of Nuclear Regulatory Research in NRC on May 12, 1993 recommended that the NRC document the significant safety insights resulting from this program and show how the safety of reactors has been improved by the IPE initiative.

The IPE Insights Program was initiated to document significant safety insights, based on the IPEs, for the different reactor and containment types and plant designs. The major insights to be gained through this program include:

- How has the IPE program impacted reactor safety?
  - How many of the plants have identified vulnerabilities or other safety issues, and what safety enhancements have been made as a result?
  - How have the improvements impacted the safety of plants?
  - Are there any "generic" improvements that have significantly affected the plant core damage frequencies (CDFs) and containment performance, or are the plant improvements plant specific?

- What is driving the CDF and containment performance?
  - What are the important design and operational features that affect the CDF and containment performance?
  - How important is the role of the plant operators?
  - How much influence do the IPE methodology and assumptions have on the results?

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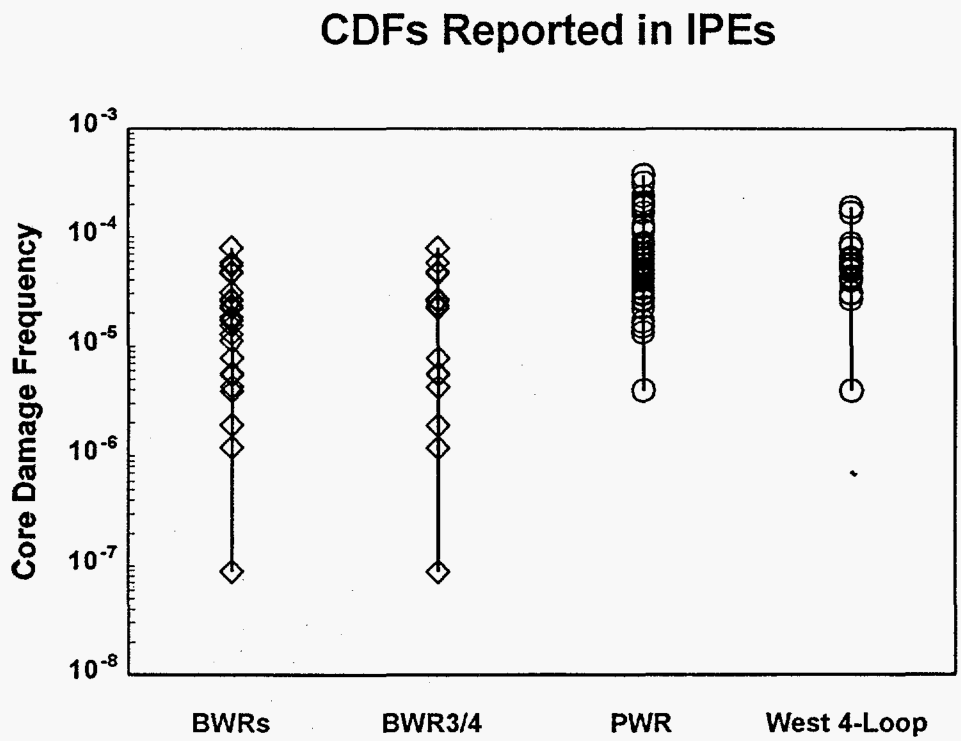

Figure 1. CDFs Reported in IPEs for Various Plant Groups

To gain these insights, the IPEs were examined to determine what the collective IPE results imply about the safety of U.S. nuclear power plants. Variations and commonalities among plant results were studied to determine which factors were most influential on the results (CDF and containment performance). In addition, the improvements that have been made at the plants, and the impact of these improvements on the plant CDF and containment performance were examined. This paper will focus on the insights regarding CDF results; other papers at this meeting discuss the impact of plant improvements and insights regarding containment performance.

This paper presents insights from the IPEs regarding CDF results for two plant groups: boiling water reactors (BWRs) 3 and 4 with Reactor Core Isolation Cooling (RCIC) systems[1] and Westinghouse 4-loop plants. The plant CDFs reported in the IPE submittals for the individual plants in these two plant groups are shown in Figure 1 along with the plant CDFs for the entire BWR group and the entire pressurized water reactor (PWR) group. On average, the BWR CDFs fall below the PWR CDFs, but there is some overlap between the two plant groups. There is a wide spread in the reported CDFs for both the BWR 3/4 and Westinghouse 4-loop plant groups. In fact, the range for these two groups spans nearly the full range for the BWR and PWR plants collectively. Both plant groups contain a single low outlier plant with a CDF about an order of magnitude below the CDF for the plant with the next highest CDF. The reasons for the variability in CDFs among the plants are discussed in the following two sections.

[1] Some BWR 3 plants have isolation condensers instead of RCIC; they are not considered in this plant group.

BWR 3/4 Perspectives

Twenty-one units (15 IPE submittals) make up the BWR 3/4 group of plants with RCIC. All of the units are housed in Mark I containments except for Limerick 1 and 2 and Susquehanna 1 and 2, which are in Mark II containments.

Figure 2 shows the contribution to plant CDF from the following accident classes: station blackout (SBO), anticipated transient without scram (ATWS), transients with loss of injection (T), transients with failure of decay heat removal (DHR), loss-of-coolant accidents (LOCA), and internal floods (FLD). The importance of specific accident classes to CDF varied significantly from plant to plant; however, the following accident classes were important for many of these plants:

- Station blackout - the loss of all offsite and onsite AC power, and
- Transients with loss of coolant injection.

In general terms, these accident classes are important contributors to CDF since they involve initiating events and/or subsequent system failures that defeat the redundancy in systems available to mitigate potential accidents. Lesser contributions were identified for the group on average from accident classes involving transients with loss of decay heat removal, ATWS, LOCAs, and internal floods. However, some IPEs did report important contributions from these accident classes. Although interfacing systems LOCAs are potentially important risk contributors since the containment is bypassed, none of the plants reported significant CDFs or radionuclide releases from interfacing system LOCAs since this accident class involves low frequency initiating events. The variation in the reported IPE results is attributed to many factors including plant-specific design features, modeling assumptions, and variation in data (including the probability of operator errors). These factors are discussed below for each accident class.

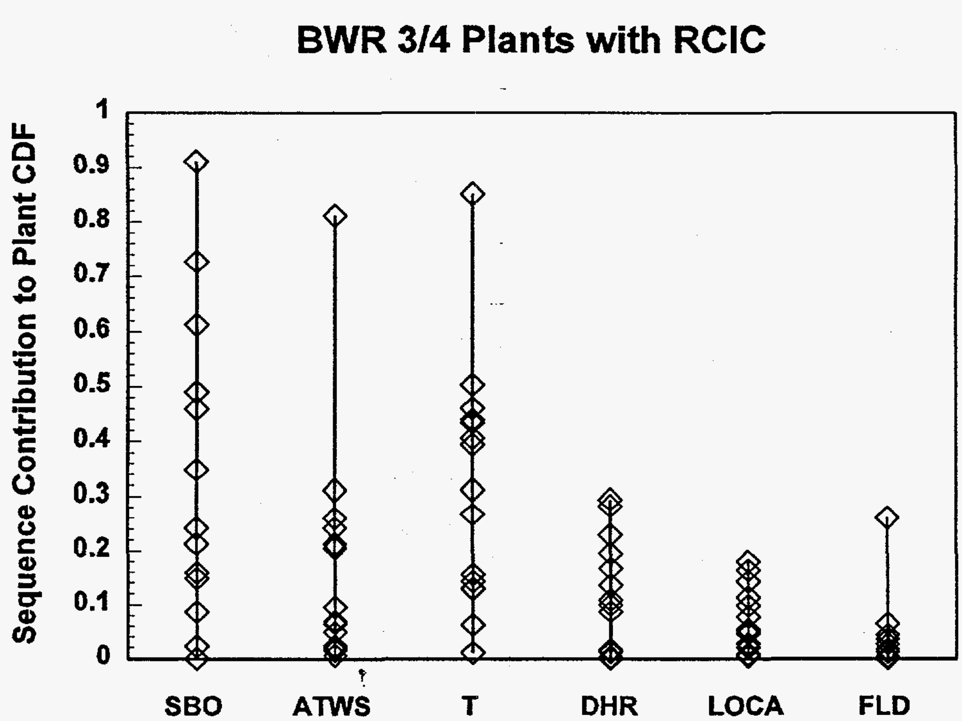

Figure 2. Accident Class CDFs for BWR 3/4 Plants with RCIC

Station blackout accidents are important contributors to CDF for most of the plants in this group. Station blackout accidents involve an initial loss of offsite power followed by failure of the emergency onsite AC power sources. The failure of AC power sources results in failure of multiple mitigating systems, leaving only steam-driven systems such as RCIC and the High Pressure Coolant Injection (HPCI) system available for coolant injection.

Generally, plant design and operational features had a larger impact on the station blackout CDF than did modeling characteristics, but no single factor dominated. Usually, combinations of contributors were important, and those combinations varied from plant to plant. With that in mind, the most influential plant features and modeling characteristics are identified and discussed below.

- Number of emergency AC power sources - The number of emergency diesel generators (usually from two to four per unit) directly affects the reliability of the emergency AC power system. Generally, the higher the number of emergency diesel generators, the lower the station blackout contribution. However, plant-specific features or modeling assumptions (such as the diesel generator cooling water system alignment at Hope Creek) can reduce the diesel generator reliability or, alternatively, increase the reliability (such as occurred in the Susquehanna IPE due to the elimination of common cause failures). The availability of additional and diverse AC power sources (such as the gas turbine generator at Fermi 2) or a separate offsite power source in addition to the normal grid connection (such as exists at Pilgrim and Vermont Yankee) reduced the station blackout contributions at those units.

- Battery depletion time - When AC power is lost, the only injection systems available are turbine driven systems (HPCI and RCIC) or, for some plants, diesel-driven firewater. Battery power is needed to provide control for HPCI and RCIC, or to maintain the automatic depressurization system (ADS) valves open so that the low-pressure firewater system can be used. Thus, when the batteries are depleted, all cooling is lost and core damage follows. Battery depletion times range from 2 hours at Brunswick 1 and 2 to 14 hours for Pilgrim, with the longer times reflecting plants making extensive use of load shedding. The contribution from station blackout accidents is generally lower for units with longer battery depletion times since the probability of recovering AC power and AC-powered mitigating systems increases with time. In fact, units with battery depletion times greater than 4 hours had significantly lower station blackout CDFs than plants with 4-hour or shorter battery depletion times.

- Use of diesel-driven firewater - Some units use diesel-driven firewater systems as a diverse means of supplying coolant injection when HPCI and RCIC have failed. The vessel must be depressurized and maintained at low pressure in order for firewater to be used. Further, this nonstandard use of the firewater system requires that piping connections and power to certain valves be available, along with appropriate procedures. The ability and the time required to inject coolant water using the firewater system thus varies from unit to unit. The station blackout contribution for units with firewater injection capability generally was dominated by sequences with early failure of RCIC and HPCI and with insufficient time available to align firewater for injection.

Transients with loss of coolant injection are important contributors to the CDF for most plants in this group. This accident class is dominated by sequences involving loss of the relatively few high-pressure coolant injection systems (typically feedwater, HPCI, and RCIC), and failure to depressurize the vessel so that the multiple low-pressure coolant injection systems can be used. Transients with loss of high-pressure injection, successful vessel depressurization, and failure of low-pressure injection systems are of lesser importance because of the significant redundancy in low-pressure injection systems. The most common transient initiating events are those that fail feedwater such as loss of offsite power, loss of feedwater, and main steam isolation valve (MSIV) closure. Loss of DC buses are also important initiating events at some plants because DC power is needed to provide control for HPCI and RCIC, and to maintain the ADS valves open so that the vessel can be depressurized.

For this group of accidents, there are two issues that are critical to the CDF. One issue involves plant-specific design characteristics while the other issue involves plant operating procedures and training along with modeling assumptions.

- Availability of alternate high-pressure injection systems - The availability of high-pressure injection systems in addition to feedwater, RCIC, and HPCI reduces the contribution of this class of accident. Several licensees (Brunswick, Cooper, and Susquehanna) used plant-specific calculations to show that the Control Rod Drive (CRD) hydraulic system in the enhanced flow mode can provide sufficient coolant injection immediately after a reactor scram to maintain core cooling. The Quad Cities IPE credited a unique safe shutdown injection system which helped reduce the importance of all loss of injection and loss of DHR sequences. Several units with motor-driven feedwater pumps (Monticello, Fermi, and Vermont Yankee) also calculated lower contributions since, unlike steam-driven feedwater pumps, these pumps can continue to operate during a transient with main steam isolation valve closure.

- Operator failure to depressurize - During the past few years, operating procedures have changed to direct the operators to inhibit ADS for most transients. If high-pressure injection then fails, the operators must recognize this condition and manually depressurize the reactor vessel to allow use of low-pressure systems. Operator error probabilities for failure to manually depressurize the vessel varied widely in the IPEs and significantly impacted the results. Whether this wide variability in the human error probabilities is due to plant-specific factors such as training or procedures or is due to some other factors is not clear.

Transients with loss of DHR are important for many of the BWR 3/4 plants. Loss of DHR transient sequences involve accidents where coolant injection succeeds, but containment heat removal fails. In this situation the suppression pool heats up, leading to containment pressurization, and if the containment is not vented, it will eventually fail. Coolant injection eventually fails, either as a result of a hot suppression pool, or the adverse conditions created in containment or the reactor building when the containment is vented or fails. These adverse conditions include loss of net positive suction head in the suppression pool or steam in the reactor building.

The key factors affecting the CDF from loss of DHR sequences involve plant-specific design and operating conditions as well as the assumptions made in the IPEs. The modeling issues are important to the results and represent an important area of uncertainty. The key factors are identified and discussed below:

- Ability of emergency core cooling system (ECCS) pumps to continue operating under severe containment conditions - ECCS pumps can fail for a variety of reasons during these accidents. Net positive suction head (NPSH) requirements may not be met if the containment fails or is vented. The pumps may fail due to high suppression pool temperature or due to steam in the reactor building. The IPEs vary significantly in their assessments of pump operation under these conditions. Some of these differences are due to actual variations in pump design or venting procedures; however, significant uncertainty remains in this area.

- Availability of alternate injection sources - Many of these plants have injection sources available that are located outside the containment and reactor building, and thus are not subject to the potential harsh environments noted above. Examples of such systems are CRD and condensate. Plants with such systems have lower CDFs, and these differences are based on actual plant design differences as opposed to modeling assumptions.

- Treatment of venting - Most of the Mark I containments are now equipped with hardened vents to prevent containment failure and harsh environments in the reactor building. Use of these vents can reduce the CDF for this accident class. However, all plants did not model these vents in their IPEs, and some IPEs accounted for loss of NPSH upon venting. Therefore, there are significant differences from plant to plant based on their treatment of venting and its effects.

ATWS sequences are significant contributors for some plants in the BWR 3/4 group. ATWS sequences involve a transient, followed by failure to shutdown the nuclear chain reaction by inserting the control rods. Power generation continues at levels far in excess of normal decay heat. An ATWS sequence can be mitigated by boron injection using the Standby Liquid Control (SLC) system, control of coolant injection, and heat removal.

The ATWS results were impacted more by modeling assumptions than by plant-specific design features. The key factors varied from plant to plant and are identified and discussed below:

- Failure of boron injection - The SLC system provides boron injection that can shut down the nuclear reaction. An important failure mode of the SLC reported in the IPEs was the failure to initiate the system. The probabilities for operator failure to initiate SLC vary by orders of magnitude among the IPEs. Some of these variations are due to different assumptions about timing, but uncertainties remain. Monticello and Fitzpatrick also credited alternate means of injecting boron using systems such as the CRD system. Generally plants with low operator error probabilities for initiating SLC and with alternate means of injecting boron had lower ATWS contributions from loss of SLC sequences.

- Power reduction using level control - Some licensees (e.g., Hatch 1 and 2) assumed that if boron injection failed, controlling the water level in the core would reduce power to within the turbine bypass capacity of the plant and allow for alternate means of placing the plant in cold shutdown. However, most licensees assumed that level control could only be successful in conjunction with the use of SLC. The ATWS contribution from loss of boron injection sequences was reduced for licensees that credited level control by itself as a means of reducing core power to a stable level.

- Operator failure to inhibit ADS and control coolant injection - If the operator fails to take action to manually inhibit ADS, the HPCI will be lost and low-pressure systems will inject at a high flow rate that is not easily controlled, possibly leading to boron flushing and to a large power surge that repressurizes the system and causes low-pressure injection to cease. Repeated cycles of pressurization may eventually lead to failure of either the reactor coolant system boundary or the low-pressure injection system. In the IPEs the probabilities for operator failure to inhibit ADS vary by several orders of magnitude. This action could be the object of further study, as the reasons for the wide variation are not obvious from the submittals. Some licensees, including Pilgrim and Brunswick, assumed that failure to inhibit ADS would not lead to core damage if the operator controlled low-pressure coolant injection flow. These plants generally had lower ATWS contributions from sequences involving failure to inhibit ADS.

LOCAs are not dominant contributors to most BWR 3/4 plants. Because LOCAs are low-frequency events and since BWR 3/4 plants have a variety of diverse injection sources to mitigate a LOCA, LOCAs are not usually important to either the core damage frequency or risk for these plants. The amount of credit given for alternate injection systems was the major parameter accounting for the variability in the LOCA results. For example, Susquehanna reported a low contribution from small-break LOCAs partially due to the credit given for enhanced CRD flow for mitigating this size LOCA. Other plants such as Fitzpatrick credited limited-volume systems such as condensate for partially mitigating a large LOCA. Most licensees assumed that vessel depressurization was needed for medium LOCAs (including stuck-open relief valves) in those cases where HPCI failed. This assumption contributed to medium LOCAs being the dominant LOCA core damage contributor at several plants.

Internal flooding is not important for most BWR 3/4 plants. Internal flooding events involve rupture of water lines or operator errors that result in a release of water that can directly fail required mitigating systems (e.g., through loss of cooling) and/or fail other mitigating systems due to submergence or spraying of required components. The most important factor in determining the importance of flooding is the plant layout. Separation of mitigating system components and compartmentalization reduces the impact of internal flood initiators. Internal flooding events were not dominant at most plants because no internal flood initiator was identified (due to the above factors) that would completely fail all systems required to mitigate a flood-induced transient without additional random failures. For a few plants, important internal flooding sequences were identified, generally involving service water system breaks that impacted equipment through both loss of cooling and through flood impacts on other mitigating systems.

Westinghouse 4-Loop Perspectives

Thirty-two plant units (20 IPE submittals) make up the Westinghouse 4-loop plant group. Twenty-two of the plant units are in large dry containments, one is in a subatmospheric containment, and nine are in ice-condenser containments.

Figure 3 shows the contribution to plant CDF from the following accident classes: station blackout, anticipated transient without scram , transients, loss of coolant accidents, internal floods, interfacing systems LOCAs (ISLOCAs), and steam generator tube ruptures (SGTR). The importance of specific accident classes to CDF varied significantly from plant to plant; however, the following accident classes were important for many of these plants:

- Transients,
- LOCAs, and
- Station blackout - the loss of all offsite and onsite AC power.

In general terms, transients are important contributors to CDF because they involve relatively high initiating event frequencies coupled with system failures that defeat the redundancy in systems available to mitigate potential accidents. LOCAs are more important for these PWR plants than for BWR plants because there are fewer systems available in the PWRs to provide low-pressure coolant injection. Station blackout accidents are relatively important for the Westinghouse 4-loop plants because they leave few systems available to prevent core damage. A few plants identified internal flooding accidents as being important, generally reflecting plant-specific weaknesses. With the exception of a single outlier plant in each category, none of the plants found ATWS or steam generator tube rupture sequences to be important contributors, and none of the plants reported significant CDFs from interfacing system LOCAs. These accidents are normally low contributors because of the low frequency of the initiating event. Although steam generator tube rupture and interfacing system LOCAs were generally found to be low contributors to CDF, they can be important risk contributors since they bypass containment. The variation in the reported IPE results is attributed to many factors including plant-specific design features, modeling assumptions, and variation in data (including the probability of operator errors). These factors are discussed below for each accident class.

Transients are important contributors to CDF for nearly all of the Westinghouse 4-loop plants. This accident class involves events that cause the reactor to trip, followed by failure to bring the reactor to safe shutdown [either failure to remove decay heat or failure to replace reactor coolant inventory following an accident-induced LOCA, normally a reactor coolant pump (RCP) seal LOCA]. Transients represent a broad category, covering both general initiators (such as reactor trip or loss of main feedwater) as well as support-system initiators (such as loss of service water or AC/DC bus).

The specific failures leading to core damage for transients were found to be quite plant-specific. However, there are some key factors affecting the CDF from transients that are common among many of the submittals. These key factors involve plant-specific design and operating conditions as well as the assumptions made in the IPEs. The modeling issues are important to the results and represent an important area of uncertainty. The key factors are identified and discussed below:

- Service water and component cooling water dependencies - At most of the plants, there is a relatively high dependence of other plant systems on component cooling water and/or service water. Thus, loss of either of these support systems is important to the overall transient CDF. For example, the dominant transient sequence at D. C. Cook is loss of component cooling water, leading to an RCP seal LOCA that cannot be mitigated. The configurations vary considerably among the plants, however, and plants with the ability to use alternate cooling configurations when the primary cooling system is lost generally have lower transient CDFs.

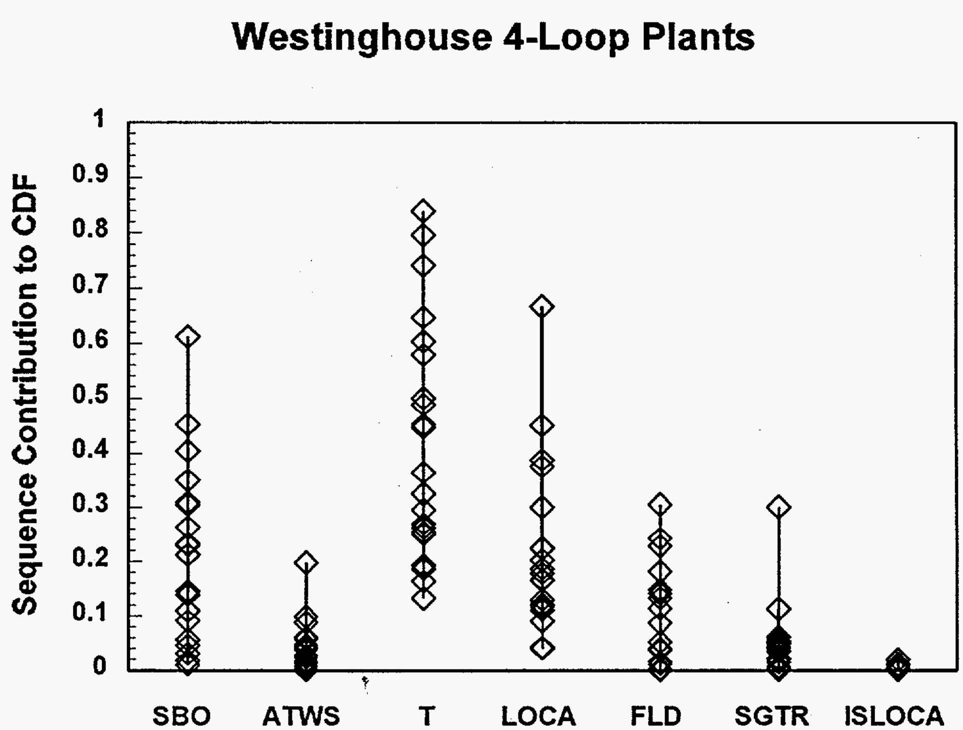

Figure 3. Accident Class CDFs for Westinghouse 4-Loop Plants

- Susceptibility to RCP seal LOCAs - For some of the plants, the importance of RCP seal LOCAs is reduced because of plant design characteristics that reduce the threat from RCP seal LOCAs. These include the use of the newer seals that are less susceptible to leakage than the older design (e.g., Braidwood, Byron, Vogtle had small RCP LOCA contributions because of using the newer seals), availability of backup systems to cool the seals when the normal configuration fails (e.g., the safe shutdown facility at Catawba and McGuire can provide RCP seal cooling), and use of different support systems for cooling the RCP seals than for providing injection (so that loss of a single system does not lead to an unmitigable LOCA). The modeling of RCP seal LOCAs varied considerably among the IPEs, with some using low probabilities of leakage and low leak rates while others used much higher values for both parameters. This variability in design and modeling had a significant impact on the results.

- Ability to cross-tie - The ability to cross-tie between systems or between units was more important for the PWRs than the BWRs because there are fewer systems available in the PWRs to provide core cooling. The ability to cross-tie somewhat compensates for this reduced redundancy. However, many plants with capability to cross-tie did not credit this in the IPE submittals.

LOCAs are important contributors to CDF for many of the plants in this group. The most common LOCA contributors are small LOCAs, but some plants are instead dominated by large or medium LOCAs. The small LOCAs are generally the larger contributors to CDF because they have a higher frequency. The dominant contributor to core damage involves failure of injection systems after the borated water storage tank is depleted and the injection pumps switch to the recirculation mode (in which water is drawn from the containment sump). Recirculation involves realigning systems, and typically involves more components than required for the injection mode, and this complexity leads to a higher failure probability.

Generally, plant design and operational features had a larger impact on the LOCA CDF than did modeling characteristics. Overall, the most influential factors are those discussed below.

- Switchover to recirculation - Most of the plants require some manual actions to initiate recirculation because high-pressure recirculation draws suction from low-pressure systems during recirculation at most plants. However, at many of the plants, the switchover of the low-pressure systems from the refueling water storage tank to the containment sump is automatic, which simplifies the actions required by the operators, and thus increases the probability of successfully completing the action. At Haddam Neck, for example, the switchover is manual, and this plant has the highest LOCA CDF for the Westinghouse 4-loop plants. For ice-condenser plants especially, the lower containment design pressure results in earlier actuation of containment sprays, so there is less time to perform the switchover. Therefore, the degree of automation is particularly important for ice-condenser plants.

- Size of refueling water storage tank - Some plants have large refueling water storage tanks so that the switchover to recirculation (and the associated complications discussed above) is either not necessary or is significantly delayed, which gives the operators more time to complete the necessary actions for the switchover to recirculation. Larger refueling water storage tanks were found to have an important effect on the LOCA CDF. For example, the two plants with the lowest LOCA CDFs, Braidwood and Byron, have relatively large refueling water storage tanks, and do not require switchover to recirculation for small LOCAs.

- Alternate actions to mitigate a LOCA - Some plants credited alternate actions such as depressurizing the reactor coolant system using the steam generator relief valves when high-pressure injection fails during a LOCA or refilling the refueling water storage tank if recirculation fails. The ability of these strategies to succeed is plant specific, but the strategies were found to be important for those submittals that credited the actions.

Station blackout accidents are important contributors to CDF for many of the Westinghouse 4-loop plants. Station blackout accidents involve an initial loss of offsite power followed by failure of the emergency onsite AC power sources. The failure of AC power sources results in failure of all injection systems and failure of motor-driven auxiliary feedwater (AFW). This leaves only turbine-driven AFW available for cooling the core and no systems available to provide injection to make up the loss through any RCP seal LOCAs that develop during the transient.

Generally, plant design and operational features have a larger impact on the station blackout CDF than do modeling characteristics, but no single factor dominates. Usually, combinations of contributors are important, and those combinations vary from plant to plant. Overall, the most influential factors are those discussed below.

- Modeling of RCP seal LOCAs and seal design - Because seal cooling is lost during a station blackout, RCP seal LOCAs are important for many of the Westinghouse 4-loop plants. As noted in the discussion of transients above, RCP seal LOCA modeling varies among the submittals, and has a significant impact on the results. Also important is whether the plants have replaced the RCP seals with the newer, temperature-resistant design. Vogtle, for example, has installed the new seals, and because of this has a relatively low contribution from station blackout accidents that involve RCP seal LOCAs (although station blackout accidents without seal LOCAs are relatively high for this plant).

- Number of emergency AC power sources - The number of emergency diesel generators (usually two or three per unit) directly affects the reliability of the emergency AC power system. Further, a few plants have a diverse AC power source, such as a gas turbine generator or an independent safe shutdown facility, and are less susceptible to common cause failures of diesel generators. Plant-specific operating history can also be important, and for example, drives Vogtle to have the highest station blackout CDF for this plant group.

- Battery depletion time - Although some plants have indicated that they have the capability to manually control AFW when DC power is lost, most plants need battery power to provide control for AFW. Thus, for most plants, when the batteries are depleted, all cooling is lost and core damage follows. Battery depletion times range from 1 to 12 hours for this group, with the longer times reflecting plants making extensive use of load shedding.

Internal flooding is important for some of the Westinghouse 4-loop plants. Internal flooding events involve rupture of water lines that result in a release of water that can directly fail required mitigating systems and/or fail other mitigating systems due to submergence or spraying of required components. The effects of internal flooding are highly plant specific, depending on the layout of equipment within the plant and the relative isolation of rooms. Because of this diversity of design and layout, each plant has different vulnerabilities to flooding and generic conclusions regarding flooding cannot be drawn. The plants with the largest flood contributions typically were dominated by floods that affected support systems such as electric power and service water, which have plant-specific designs.

ATWS is not a dominant contributor for most Westinghouse 4-loop plants. ATWS sequences involve a transient, followed by failure to shutdown the nuclear chain reaction by inserting the control rods. Power generation continues at levels far in excess of normal decay heat. An ATWS sequence can be mitigated by pressure control and heat removal. Because of the low frequency of failure to scram, ATWS is a relatively low contributor (less than 10% of CDF) for nearly all the plants in this group. The single plant with a significant ATWS contribution, Indian Point 3, operates with the power-operated relief valve (PORV) block valves closed, which reduces the relief capacity of the primary system during the early phase of an ATWS.

Steam generator tube rupture is not a dominant contributor to CDF for most Westinghouse 4-loop plants. Steam generator tube rupture sequences involve leakage from the primary to the secondary through a ruptured steam generator tube, followed by either failure to mitigate the leak or failure to establish long-term core heat removal. Steam generator tube rupture accidents are a minor contributor to plant CDF at most Westinghouse 4-loop plants because of the low frequency of the rupture occurring. However, the contribution to risk is more significant at many plants because the releases bypass containment.

Although nearly always minor, the contribution to CDF from steam generator tube rupture is primarily driven by assumptions made in the IPEs. Steam generator tube rupture accidents require considerable involvement from operators, and modeling of operator errors as well as alternate strategies varies widely among the IPEs. Although this is an area containing large uncertainties, it does not have a large impact on the overall plant CDF.

Summary

The objective of the IPE Insights Program is to document the significant safety insights, based on the IPEs for different reactor and containment types and plant designs. That objective is being achieved by examining the IPE CDF and containment performance results for individual plants in various groups and searching for commonalities and differences. For the BWR 3/4 and Westinghouse 4-loop Plants, transients, station blackout, and LOCAs (for PWRs) tend to dominate, but individual plant results vary considerably, reflecting both differences in plant design and operational characteristics, and differences in IPE modeling.