UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

GT3500205A

January 29, 2002

MEMORANDUM TO: ACRS Members

FROM: Hossein Nourbakhsh, Senior Fellow HPN

SUBJECT: LARGE BREAK LOCA INITIATING EVENT FREQUENCY ESTIMATES

Estimates of LOCA initiating event frequencies are being utilized in a number of programs under review by ACRS including:

- Westinghouse AP1000 passive plant design PRA that was discussed at the January 23-24, 2003 Reliability and Probabilistic Assessment Subcommittee meeting;
- PTS Reevaluation Project: Technical Bases for Potential revision to the Pressurized Thermal Shock (PTS) screening criterion;
- The risk-informed reevaluation of 10 CFR 50.46 (along with appendix K and GDC 35), the Emergency Core Cooling System (ECCS) requirements.

The NUREG/CR-5750 (Ref. 1) pipe break LOCA estimates or their NRC interim updated values were used to represent the initiating event frequencies in the above programs. The large break LOCA initiating event frequency estimates reported in NUREG/CR-5750 are more than one order of magnitude smaller than the values used in WASH-1400 (Ref. 2) and NUREG-1150 study (Ref. 3). A comparison of PWR large LOCA frequencies reported in various studies is presented in Figure 1.

This memorandum provides a summary overview of the bases for estimates of PWR large LOCA initiating event frequencies reported in various studies.

Reactor Safety Study (WASH-1400)

The first systematic attempt to provides estimates of the pipe LOCA frequencies was contained within the Reactor Safety Study (WASH-1400), completed in 1975. Because of limited combined years of reactor service experience at the time of WASH-1400 study, the pipe LOCA frequencies were derived based on nuclear as well as non-nuclear operating experience. WASH-1400 examined data sources from experimental reactor experience, military applications

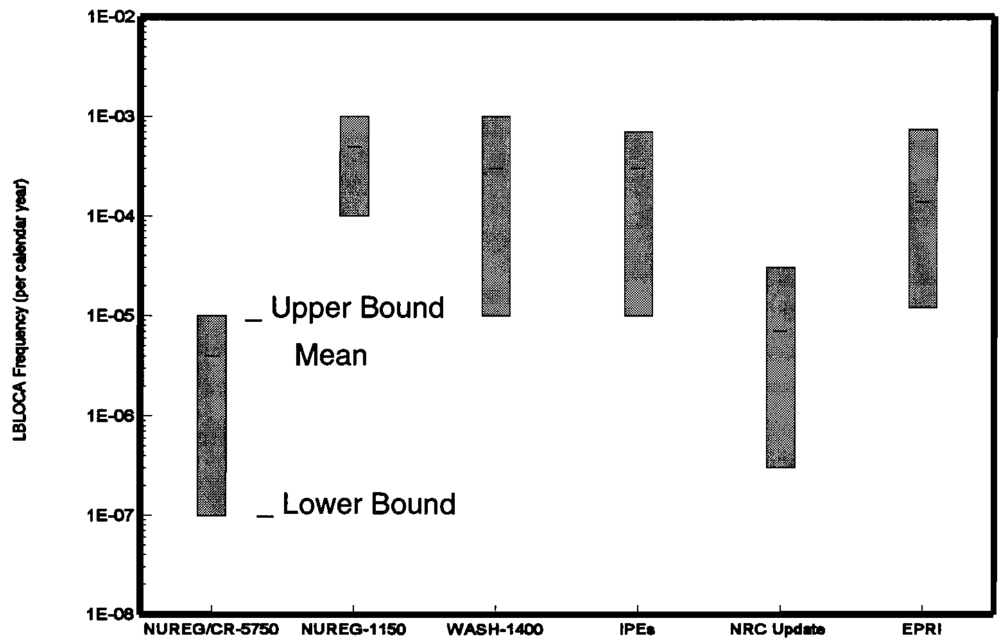

Figure 1. Comparison between NUREG/CR-5750 PWR large LOCA estimated frequency and values from WASH-1400, NUREG-1150, IPEs, EPRI and NRC interim update.

experience (naval), U.S. non-nuclear power utility experience, and the United Kingdom historical information on pipe failures.

Each of the various data sources was individually evaluated to obtain pipe rupture assessments. To incorporate the resulting uncertainty and possible variations that could exist in the assessments, a lognormal distribution with an error factor of 10 was assigned to the LOCA frequency ranges. The median value for the large break (pipe rupture size of grater than 6 inches) LOCA initiating event frequency was assigned to be $1X10^{-4}$ per plant per year (for both PWRs and BWRs).

NUREG-1150 Study

The WASH-1400 pipe break estimates formed the basis for the large LOCA initiating event frequencies used in NUREG-1150. NUREG-1150 used the generic value of large LOCA initiating frequency from WASH-1400 for BWRs. However, for PWRs, the probabilistic risk assessments (PRAs) that was performed prior to 1987 were reviewed and a mean value of $5X10^{-4}$ per reactor year typical of the range was selected for large LOCA initiating event frequency. NUREG-1150 used error factors of 3 (and a lognormal distribution) for all LOCA frequencies (for both PWRs and BWRs).

EPRI Study on Pipe Failures in U.S. Commercial Nuclear Power Plants

EPRI and Northeast Utilities Service Company sponsored a study, completed in 1992 (Ref.4), to provide a nuclear plant pipe failure data base reflecting more recent experience, and to provide an updated assessment of pipe failure rates. The principal sources of nuclear plant component failure data used in this study were Licensee Event Report (LER) abstracts, Nuclear Power Experience (NPE), and the Nuclear Plant Reliability Data System (NPRDS) covering plant experience up to 1990. Other sources such as WASH-1400 were also used to augment the data base on catastrophic pipe failures (ruptures).

A new parameter that was introduced in this study served as an important tool in the quantification methodology. This parameter, called the failure severity code, accounts for the fact that the effective break area can be significantly smaller than the area calculated using pipe inner diameter. This parameter was used to estimate the conditional probability of having a given effective break size for a given size pipe (Ref. 4). The methodology also accounted for factors that were postulated to significantly affect the values of the failure rates and that could be quantified from the data base. These include the nuclear steam system supplier (NSSS), system type, pipe size, and plant age.

Generic LOCA frequencies were calculated by using the average number of pipe-sections per reactor type(PWR and BWR). The potential LOCA contributors from sources other than Pipe failures (e.g. failure of reactor coolant pump seals) were excluded in the calculations. The estimates were averaged over all plant ages, and are thus time-independent. The generic PWR point estimate large LOCA frequency was calculated to be $1.4X10^{-4}$. The upper-bound for large LOCA frequency was approximately obtained by multiplication of the point estimate frequency by a factor of five. The corresponding lower-bound was obtained by dividing the point estimate by a factor of twelve. The assumption on the exact form of distribution on the LOCA frequencies was considered to be largely arbitrary and was not specified in the study.

NUREG/CR-5750 LOCA Frequency Estimates

Pipe break LOCA frequency estimates were updated within Appendix J of NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995" published in 1999 (Ref. 1). Separate frequencies were estimated for PWRs and BWRs

The approach taken in NUREG/CR-5750 (Ref.1) to estimate the frequency of a large break LOCA uses the available experience to estimate the number of reactor calendar years and through-wall crack events in large-diameter piping to first estimate a leak frequency. A conditional rupture probability (given a through-wall crack or leak) was then estimated and factored into the frequency calculation to produce a rupture (LOCA) frequency estimate. An error factor of 10 (assuming a lognormal distribution) was used to capture the uncertainties in the large break LOCA frequency estimates.

A total of 3362 (1392 U.S. and 1970 non-U.S.) PWR reactor-calendar-years operating experience for year 1967 through 1997 was used to calculate LOCA frequency estimates. The Russian build VVERs (except for the two Finnish-built VVER reactors) and pressurized heavy water reactors (e.g., CANDU designs) were not included in this analysis because of limited data and the differences between these reactor designs and U.S. LWRs.

A search of the available literature identified only a single through-wall crack or leak event in large diameter piping. In this foreign reactor (Genkai, Unit 1, Japan) event (Ref.5), a 203 mm (8-inch) schedule 140, type 316 stainless steel residual heat removal system line was found leaking (0.2 gpm) in 1988. The un-isolated leak was in the weld joint between an elbow and a horizontal pipe section located between the hot leg and the first isolation valve. The crack extended 3.8 inches (97 mm) circumferentially around the pipe on the inside surface of the weld. About 0.06 inches (1.5 mm) of this crack extended completely through the wall. The crack was reported to be caused by thermal fatigue.

Using this single leak event in 8-inch diameter piping, 3362 PWR-calendar years of worldwide PWR operating experience, the corresponding probability of leakage was calculated to be 3.E-4 (1/3362).

For LBLOCA, NUREG /CR-5750 analysis used a simplified correlation for the conditional probability of a rupture given a leak, proposed by Beliczey and Schulz (Ref. 6). This simplified correlation was derived from results and insights from structural mechanics models, experimental data, and operating experience with German PWRs. The conditional probability of rupture is inversely proportional to pipe diameter and is defined as:

$$P_{R/TW} = 2.5/DN \quad \text{(Eq.1)}$$

Where

$P_{R/TW}$ = mean probability of rupture given a through-wall (TW) crack
DN = nominal pipe diameter in mm

NUREG/CR-5750 reported that the above correlation for various piping diameter is supported by results presented in a recent report from Swedish Nuclear Power Inspectorate or SKI (Ref. 7), which used a Bayesian statistics and the worldwide SKI pipe failure database to estimate conditional break probabilities for stainless steel piping in nuclear power plants. Furthermore, NUREG/CR-5750 states that the results from probabilistic fracture mechanics analyses on PWR and BWR piping systems performed by Lawrence Livermore National Laboratory (Ref. 8), Battelle (Ref. 9), and Pacific National laboratory (Ref. 10) also support the Beliczey and Schulz correlation (Eq. 1). Comparison to these studies are discussed in NUREG/CR-5750 (Appendix J, Section J.5.1).

Using Eq. 1 for 8-inch (203-mm) piping:

$$P_{R/TW} = (2.5/203) = 0.0123$$

Therefore, the PWR LBLOCA frequency was estimated as 3.6 E-06 (3.E-4 X 0.012).

NRC Interim LOCA Frequencies

Several failure mechanisms were recently unfolded which were not previously experienced. These include Primary Water Stress Corrosion Cracking (PWSCC) of PWR alloy 82/182 welds which occurred at VC Summer and Ringhals, PWR Vessel degradation at Davis Bessie, hydrogen combustion failures at Hamaoka and Brunsbuettel, and Control Rod Drive Mechanism (CRDM) and housing cracking at Oconee and other plants. In response to the concern that these and other potential aging-related mechanisms may not be adequately represented within the NUREG/CR-5750 LOCA frequency estimates, the NRC Office of Regulatory Research (RES) recently completed a near- term elicitation study to determine the suitability of NUREG/CR-5750 LOCA frequency estimates. The interim LOCA frequencies were determined from an expert judgment process based on elicitation of a panel of eleven NRC staff from RES and NRR. The interim LOCA frequency were determined to be in the order of two to four times greater than NUREG/CR-5750 values (Ref. 11).

It should be noted that RES has initiated a formal expert judgment process to pursue the technical issues raised by the near term elicitation study and to provide a more rigorous basis for updated LOCA frequency distributions. This effort is expected to provide results within the next 8 to 12 months.

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