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Mr. Scott F. Newberry
Director, Risk Analysis and Applications
U.S. Nuclear Regulatory Commission
Mail-Stop T10E50
Washington, DC 20555-0001

Dear Mr. Newberry:

Enclosed for your review are the results of the EPRI expert elicitation meeting on determining the probability of a loss of offsite power (LOOP) given a large break loss of coolant accident (LOCA).

The paper is forwarded to you, in advance of the May 2, 2002, industry-NRC meeting on the elimination of the LOCA-LOOP requirement, to facilitate a more constructive and productive meeting. The paper builds on the previous industry-NRC interactions that are aimed at improving the safety focus of NRC regulations through the adoption of risk-informed, performance-based concepts.

If you or your staff have any questions, please contact Adrian Heymer, (aph@nei.org, 202-739-8094), or me at 202-739-8081.

Sincerely,

Anthony R. Pietrangelo

Anthony R. Pietrangelo

Enclosures

Enclosure

Probability of LOOP Given Large LOCA

Results of Expert Elicitation Meeting
EPRI, Palo Alto, CA
March 20, 2002

Introduction

As part of improving safety-focus and regulatory efficiency, the NRC has proposed an initiative to assess whether the specific statements in General Design Criterion 35 of Appendix A to Part 50 and in 10 CFR 50.46 are unnecessary. The statements relate to the need to analyze and design for a coincident loss of offsite power (LOOP) with a large break loss of coolant accident (LOCA). In support of the initiative, the industry, with BWROG as the lead entity, is developing the basis to support the deletion of the coincident LOCA-LOOP requirement. This paper provides the basis for the probability of a LOOP given a large break LOCA event.

Background

The LOCA-LOOP task is part of a larger project to risk-inform NRC technical requirements. The acceptance criteria are based on the concepts and metrics described in Reg. Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant-specific Changes to the Licensing Basis.*

The specific sequence of concern is LOOP as a consequence of the large LOCA. To achieve success, the core damage frequency (CDF) and the change in CDF must be below the criteria described in Reg. Guide 1.174, with a similar approach for the large early release frequency metric. In this case, the low risk-significance is achieved by demonstrating a low initiating event frequency.

The industry and/or NRC have developed the following information relative to this problem:

- The database of LOOP events given a plant trip contains between 8 and 10 events in 3415 trips from 1984 through 2001. This probability is estimated to be 0.003. Both EPRI and Brookhaven have independently determined this value. This probability includes any effects of inadequate grid voltage and VAR support. EPRI has researched recent grid support data and has reviewed grid support contractual requirements between BWR plants and grid operators. The research showed no adverse trend in grid support, and contracts will assure that no adverse trend develops.

- The database of LOOP events given a full ECCS actuation signal contains one event in 14 full ECCS actuations. This probability is 0.07. The failure cause, for the event, which occurred, was overloaded safety buses from out-of-date bus transient analysis. Industry response to this and other events has greatly reduced the likelihood of this failure cause. Therefore, this event is not used. Instead, using zero failures in 14 events yields a best estimate probability of 0.035 to 0.05 using accepted statistical approaches. Given the few number of events, the uncertainty on this estimate is quite large.

The database of 14 events is too small to yield an accurate best estimate value for conditional LOOP. It is not practical to develop a more accurate estimate from data analysis alone. Therefore an expert elicitation process was used to adjust the conditional LOOP probability.

Expert Elicitation Process

The bases and input into the expert panel process was derived from the events database using a facilitated expert meeting, which considered the events data, the equipment design and operation, and the nature of the ECCS signal transient.

The process was generally consistent with the approach described in the following references:

- NUREG/CR-6372: R.J. Budnitz, G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on the Use of Experts", U.S. Nuclear Regulatory Commission and Lawrence Livermore National Laboratory, Report NUREG/CR-6372, 1997

- NUREG-1653: J.P. Kotra, M.P. Lee, N.A. Eisenberg, and A.R. DeWispelare, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program", U.S. Nuclear Regulatory Commission, Office of Nuclear Materials Safety and Safeguards, Report NUREG-1563, 1996

- ASME PRA Standard (in publication)

The experts were:

- Harvey L. Wyckoff, Consultant to EPRI. 53 years nuclear plant experience including engineering and construction of electrical switchyards, and transmission and distribution facilities. Includes 19 years with EPRI's Nuclear Safety Analysis Center, responsible for EPRI series of Causes of Loss of Offsite Power reports since 1986, and other work related to safety of nuclear plants.

- Kiang Zee, P.E., ERIN Engineering. Electrical engineer with over 20 years of experience in electrical system design, analysis, and risk assessment. Experience also includes vertical induction motor design with General Electric.

Participated in original design and construction of nuclear power plants as well as system design and regulatory compliance assessments (SSFI/EDSFI). Actively engaged for the past 10 years in the performance of nuclear power plant risk assessments.

- J. D. Wolcott, TVA Browns Ferry. Nuclear Safety Specialist, familiar with nuclear power systems design, safety system functional assessments for power systems, and nuclear plant operations for their respective plants.

- Gerry Nicely, TVA, Corporate Staff, Senior Electrical Specialist. 30 years experience with all TVA Nuclear Plants, Expertise in AC Auxiliary Power Systems and Plant/Grid Interfaces. Experienced with nuclear power systems design, electrical short circuit analyses, load flow/transient analysis, safety system functional assessments for power systems, grid interfaces including implementation of NERC Planning Standards. Senior Member IEEE, member IEEE Nuclear Power Engineering Committee, Vice-Chair of IEEE Subcommittee 4 (Auxiliary Power Systems), member SERC Generator Owners Working Group.

- David C. Alstad, PE, NMC, Monticello Station. Electrical engineer, BWR Senior Reactor Operator Certified, Lead AC Power Systems Load Flow Design and Analysis, EDG ECCS loading, safety system functional assessments for power systems, grid interface and analysis.

- Michael S. Tucker, Senior Staff Engineer, Exelon Nuclear, Mid West Regional Operating Group. BSEE, 1977, IIT, Electrical Power; 22 years in the nuclear industry. Presently responsible for oversight of the AC Auxiliary Power Systems analysis for Byron, Braidwood, Clinton, Dresden, LaSalle and Quad Cities, including load flow, short circuit, degraded voltage, transient analysis, diesel generator loading and interface with the offsite Transmission and Distribution organizations.

LOOP Given LOCA Sequence Model

The sequence of events that must occur to avoid a LOOP/LOCA event is depicted by the following event sequence:

1.	Stable Grid	2.	Successful Bus Transfer	3.	Successful Pump Sequencing	4.	Other Issues

The events are discussed below:

1. The grid must be stable, having adequate VAR and voltage support to maintain stability given loss of plant generation and rapid loading of equipment,

particularly ECCS pumps. For a BWR, this typically includes six medium-voltage pump motors including RHR or LPCI pumps, low-pressure core spray pumps, and sometimes a high-pressure core spray pump. The specific issue being examined was whether the response of the offsite power system given a plant trip is substantially different than that due to a LOCA. The predominate impact given a plant trip or a LOCA is loss of the unit generator and its associated watt and VAR support to the system. The response of the plant to the LOCA includes the automatic starting of ECCS loads. It may also include the automatic tripping of non-ECCS loads such as the Recirculation Pump MG set drive motor. Other loads such as the Condensate and Condensate Booster Pump motors may also trip or have their load considerably reduced due to feedwater flow isolation.

The net difference in plant loading due to the sequencing of the ECCS loads could still result in additional watt and VAR demand and a corresponding increase in the grid voltage sag. However, this additional sag was not considered to be a significant factor given the analyses performed examining system performance. The analysis performed to confirm acceptable T&D system performance typically includes the consideration of a pre-existing significant upset (contingency condition) combined with a subsequent second contingency condition concurrent with postulated loss of generation from the nuclear power plant and LOCA event.

The conclusion of the Expert Elicitation Process is that the conditional loss of offsite power due to grid-centered factors given a LOCA would be the same as that for a plant trip. The probability for this event is represented by the LOOP-given-trip probability of 0.003. This is the value based on industry experience of 3415 trips and the EPRI/BWROG investigation to confirm that the grid support of nuclear plants has no adverse trend.

EPRI/BWROG investigated both grid contracts and degraded grid events within the previous two-year period. The investigation confirmed that: 1) contracts either exist, or are in the process of being negotiated, between transmission grid providers and all BWR plants, and 2) these contracts vary in detail but require transmission grid operators to maintain the grid adequate to meet plant technical specifications. These agreements also meet the intent of INPO SOER 99-01 which requires improved means of communication between plants and transmission providers and re-evaluating electrical system transients such that grid voltage support would be sufficient to support loss of the unit including power demands in response to a LOCA. Whenever the transmission providers believe that this capability is likely to be compromised, the nuclear unit operators are to be alerted and actions are taken to eliminate this alert condition.

Such alert conditions occurred at several plants during the previous two-year period. No instances were found for which the grid could not provide voltage support needed in the event of a LOCA. Even on the west coast, where prolonged alert conditions existed during early 2001, the transmission provider (CAISO) ensured that the power stability requirements specified in the contracts with Diablo Canyon and SONGS were met. Therefore, the likelihood of such a degraded condition is considered to be small, and its contribution to LOOP probability is adequately represented by the industry trip experience. Furthermore, the alert system could provide adequate performance monitoring for the future, so that any trend toward increasing probability would be evident.

2. Power must transfer from its at-power source -- the plant generator through the auxiliary transformer – to its shutdown source – the grid through the startup transformer. Although every plant configuration is somewhat different, many plants have safety loads, including ECCS, powered from the grid at all times. The experts believe that at least one division of ECCS is normally powered from the grid. Nonetheless, it was recognized that random failures that cause a failure of the bus transfer function are included in the industry experience data in item 1 above. Therefore, the assessment of this event focused on examining whether there are differences in the bus transfer function during a plant transient as compared to a LOCA event. This assessment concluded that there are no differences. The occurrence of a LOCA would result in the same demand for tripping of the bus normal feeder breaker and the closure of the alternate supply breaker. The occurrence of a postulated LOCA event would not alter the performance of, or demand on, the bus transfer function. Because there is no difference, the contribution of bus transfer failures to the conditional loss of offsite power probability is considered to be included in the value developed for Item 1.

3. The ECCS pumps must sequence onto their respective safety buses without inducing an automatic transfer to the EDGs. This transfer occurs at a specific voltage drop for a specific duration (say, 6% drop for 5 seconds). This load sequencing during a postulated LOCA event creates a substantially different demand on the plant power distribution system. During a postulated plant trip, loads are transferred, but there is no addition of large system loads. The starting of a large induction motor is expected to cause a momentary dip in the bus voltage to below the setpoint of the 2nd level undervoltage relays. Upon successful load acceleration the bus voltage must recover to a level high enough to reset the undervoltage relay.

   There are two types of potential failures for this event. First, latent human errors could be present from inaccurate system voltage analysis or inaccurate calibration. Second, equipment failures could impact logic relays such that they fail to respond to the voltage recovery and initiate separation from the offsite power system due to a false signal. For some plants, the failure of the transformer automatic load tap changer would result in a bus voltage below that required for relay reset. Latent human errors could also change the loading sequence to induce the low voltage condition.

The probability of a latent human error in the system voltage analysis is estimated to be 0.003 [4]. This value is based on a 0.03 probability of errors being introduced by the analysis originator and a 0.10 probability that those errors are not identified and corrected through the analysis checking and verification process. An additional factor could be applied to reflect the fraction of errors that causes a materially significant error, which results in the state of knowledge with respect to LOCA response to be flawed. This additional factor is not considered in this assessment.

The probability of a latent human error in the calibration process is estimated to be 0.003. This value is based on a 0.03 probability of error by the technician performing the calibration and a 0.10 probability that the error was not identified and corrected via the checking process.

The probability of an equipment failure that prevents the logic relays from properly responding to the bus voltage recovery is estimated to be 4.3E-4. Each bus has multiple undervoltage relays arranged in a coincident logic arrangement. These undervoltage relays may have integral time delays or may be connected to an external timer. For the purposes of this assessment a logic circuit configuration consisting of a single external timer is assumed. The failure of that timer to 'de-energize' or properly 'delay' would result in the spurious separation of the buses from the offsite supply. This failure of the time delay relay is estimated to be 1.0E-4 [1]. The failure of an undervoltage relay is estimated to be 0.0033 based on a failure rate of 3E-6/hr [2] and quarterly testing. Since multiple undervoltage relays must fail, a common cause factor of 0.10 is applied reducing the overall failure probability to 3.3E-4. Combining this value with the time-delay relay failure results in a total failure probability of 4.3E-4. The treatment of a typical logic circuit configuration wherein multiple individual timers are used would reduce this value.

The probability that an automatic tap changer failure prevents the recovery of bus voltage to acceptable levels is estimated to be 4.0E-4. This value is based on a failure rate of 4.7E-8/hr [3]. The tap changers are conservatively treated as being exercised only on a refueling cycle basis. This refueling is assumed to occur every 24 months.

The total failure probability for this item is the sum of the contributions in Item 3:

$$3.0E\text{-}3 + 3.0E\text{-}3 + 4.3E\text{-}4 + 4.0E\text{-}4 = 6.8E\text{-}3.$$

4. During the course of the Expert Elicitation and review of industry LOOP experience it was noted that latent errors involving plant configuration control issues could be important. For example, a 1997 plant event illustrates the concern. The system voltage analysis referenced in Item 3 appears to have been accurate and valid, and the calibration procedures followed. However, the failure to properly translate the design requirements into the plant may have contributed to a LOOP following a plant trip. The calculation stipulated a required transformer tap position, the analytical limit for the 2^{nd} level undervoltage relays, and/or the required time delays for the voltage protection circuitry. The failure to properly translate these requirements into the plant procedures could have resulted in incorrect tap setting and relay setpoints. The numerical methods available for estimating a probability for such a latent error results in a low probability value that can be enveloped by the latent human failure probability developed in Item 3 above. However, the experts were concerned that the complexity of the voltage analysis created many opportunities for inconsistencies between the analysis and the actual plant setup that might not be accounted for by the quantification model. Therefore, it is recommended that this consistency be verified as part of any plant change that relies on this probabilistic assessment.

<u>Probability of LOOP Given Large LOCA</u>

The sum of the probabilities from each of the four items above yields the best estimate for the conditional probability of a LOOP given a LOCA:

$$LOOP_{LOCA}\ PROB = 0.003 + \varepsilon + 0.0068 + \varepsilon = 0.0098 \sim 0.01$$

This result is likely to have a much higher confidence limit than one based on the 14 events involving full ECCS actuation, and takes into account reasons that might render an estimate based solely on the 3415 plant trips somewhat suspect.

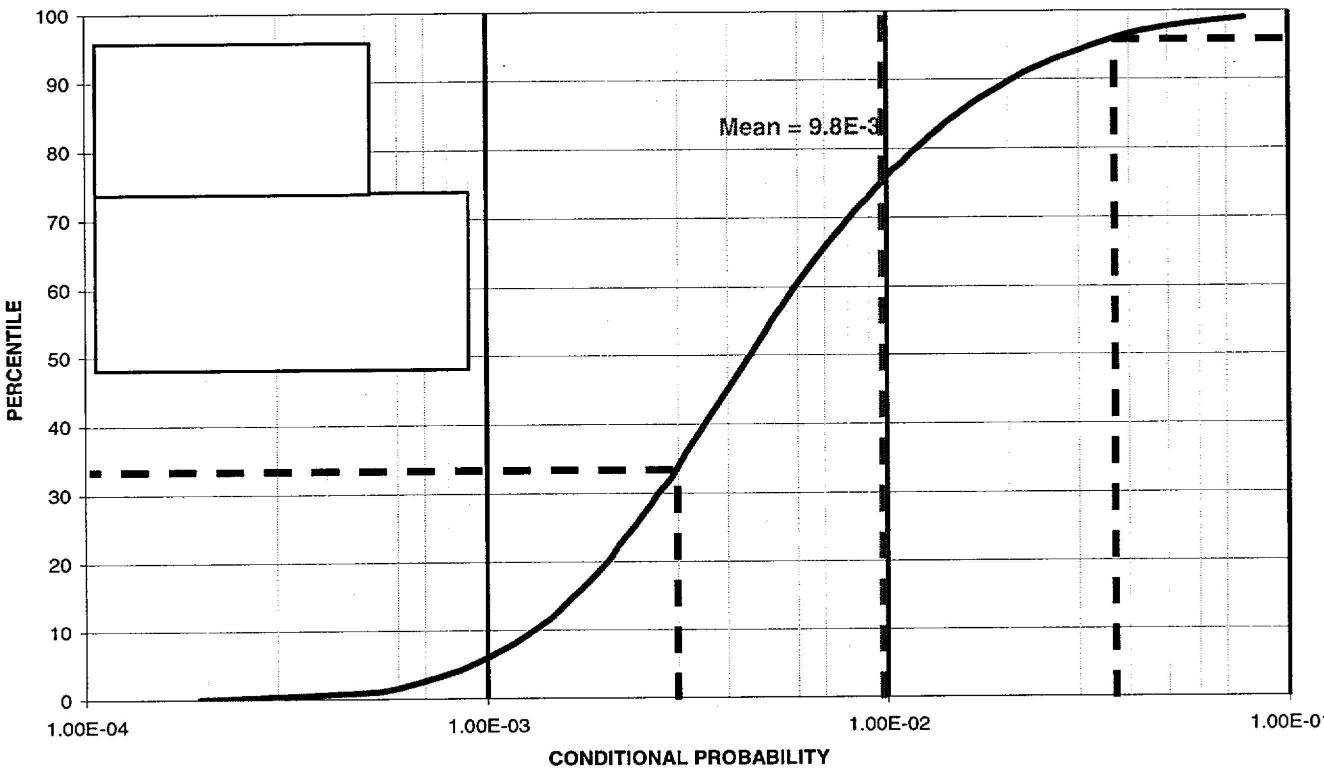
The reasonableness of the resultant 0.0098 probability can be assessed by comparison to industry experience data. As discussed earlier, the experience data for LOOP given a plant trip is 0.003. It is expected that the LOOP probability given a LOCA would be higher than this value. The other data point involves 0 failures in 14 events. This translates to a probability of approximately 0.035. It is expected that the LOOP probability given a LOCA would be lower than this value. Based on discussions during the Expert Elicitation and the industry activities related to SOER 99-01, it would be reasonable to expect that the estimate for a LOOP given a LOCA would be between the statistical mean of the two bounding values.

Alternatively, a probability distribution for the result of the expert panel elicitation can be generated by combining the probability distributions for each of the constituent parts considered in detail above. Using the mean values above, representative error factors, and assumed lognormal distributions yields the following percentile and mean values:

5 percentile	0.0011
95 percentile	0.03
Mean	0.0098

The figure below presents details of the result. This result gives confidence that the actual probability is well below the value derived from ECCS actuation experience of 14 actuation events. It also indicates that the value is likely higher than the trip experience of 3415 events, but this experience base does lie within the 95 percentile confidence bounds.

CUMULATIVE DISTRIBUTION FUNCTION FOR LOOP FOLLOWING A LOCA



Conclusion

The best value to use for the conditional probability of LOOP given a LOCA based on experience data and this expert elicitation process is 0.01.

References

1. ALWR Requirements Document. EPRI TR-016780, Rev. 6, 1997.
2. Probabilistic Safety Analysis Procedures Guide, NUREG/CR-2815, August 1985.
3. IEEE Guide to the Collection and Presentation of Electrical, Electronic, Sensing Component, and Mechanical Equipment Reliability Data for Nuclear Power Generating Stations, IEEE-500, November 1993.

4. The quantification of the conditional LOOP probability was developed with the assistance of Dr. Ed Burns and Mr. Doug True.

   - Dr. Ed Burns, ERIN Engineering. Nuclear engineer with considerable experience in the application of probabilistic risk assessment technology to the solution of engineering problems. Significant experience in the field of probabilistic risk assessment, severe accident analysis, and emergency procedure examination.

- Doug True, ERIN Engineering. Technical background includes engineering, safety analysis and operations of a variety of nuclear facilities. He has served as the technical director of numerous large-scale PSA projects ranging from the preparation of nuclear power plant risk assessments to Safety Analysis Reports for DOE facilities to chemical process industry safety analyses.