50-335

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
April 23, 1998

Mr. T. F. Plunkett
President - Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF OPERATIONAL CONDITION AT ST. LUCIE UNIT 1

Dear Mr. Plunkett:

Enclosed for your review and comment is a copy of the preliminary Accident Sequence Precursor (ASP) analysis of an operational event which occurred at St. Lucie Unit 1, on November 2, 1997 (Enclosure 1), and was reported in Licensee Event Report (LER) No. 335/97-011. This analysis was prepared by our contractor at the Oak Ridge National Laboratory. The results of this preliminary analysis indicate that this condition may be a precursor for 1997. In assessing operational events, an effort was made to make the ASP models as realistic as possible regarding the specific features and response of a given plant to various accident sequence initiators. We realize that licensees may have additional systems and emergency procedures, or other features at their plants that might affect the analysis. Therefore, we are providing you an opportunity to review and comment on the technical adequacy of the preliminary ASP analysis, including the depiction of plant equipment and equipment capabilities. Upon receipt and evaluation of your comments, we will revise the conditional core damage probability calculations where necessary to consider the specific information you have provided. The object of the review process is to provide as realistic an analysis of the significance of the event as possible.

In order for us to incorporate your comments, perform any required reanalysis, and prepare the final report of our analysis of this event in a timely manner, you are requested to complete your review and to provide any comments within 30 days of receipt of this letter. We have streamlined the ASP Program with the objective of significantly improving the time after an event in which the final precursor analysis of the event is made publicly available. As soon as our final analysis of the event has been completed, we will provide for your information the final precursor analysis of the event and the resolution of your comments.

We have also enclosed several items to facilitate your review. Enclosure 2 contains specific guidance for performing the requested review, identifies the criteria which we will apply to determine whether any credit should be given in the analysis for the use of licensee-identified additional equipment or specific actions in recovering from the event, and describes the specific information that you should provide to support such a claim. Enclosure 3 is a copy of LER No. 335/97-011, which documented the event.

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Please contact me at (301) 415-1479 if you have any questions. This request is covered by the existing OMB clearance number (3150-0104) for NRC staff followup review of events documented in LERs. Your response to this request is voluntary and does not constitute a licensing requirement.

Sincerely,

/s/

William C. Gleaves, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-335

Enclosures: As stated

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cc w/encls: See next page

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LER No. 335/97-011

Event Description: Non-conservative recirculation actuation signal set point

Date of Event: November 2, 1997

Plant: St. Lucie, Unit 1

Event Summary

St. Lucie, Unit 1 was defueled in support of a steam generator replacement refueling outage. Utility personnel determined that the engineered safety feature actuation system (ESFAS) recirculation actuation signal (RAS) bistable set point for indicating the water level in the refueling water tank (RWT) had been set less conservatively than the Technical Specification set point. Plant personnel determined that the unit was more susceptible to core damage in the event of a large-break loss-of-coolant accident (LBLOCA) since changing the span of the RWT level indication during a 1993 refueling outage.[1] The nominal core damage probability (CDP) over a 1-year period estimated using the IRRAS model for St. Lucie is 1.7×10^{-5}. The increase in the CDP (i.e., the importance) for this event over a 1-year period because of an improper RAS set point is 1.7×10^{-5}. Hence, the less conservative RAS set point results in an estimated conditional core damage probability (CCDP) for a 1-year period of operation of 3.4×10^{-5}. Uncertainty in the frequency of a LBLOCA (none have occurred) and uncertainty in the amount of emergency core cooling system (ECCS) flow required when recirculation is initiated contribute to a substantial uncertainty in this estimate.

Event Description

On October 27, 1997, St. Lucie, Unit 1 was defueled in support of a steam generator replacement refueling outage. As part of the outage, obsolete ESFAS bistables were being replaced to improve system reliability and calibration methods. The equipment to be replaced included all four channels of the RWT low level bistables. A signal from these RWT low-level bistables causes the operating mode of the safety injection system to change from the injection mode to the recirculation mode following a loss-of-coolant accident (LOCA).

Because of the RWT bistable changes, a system engineer performed additional verification to ensure that the RWT level set point agreed with the instrument loop scaling requirement. This review showed that the Technical Specification set point of 48 in from the bottom of the RWT correlated to a bistable set point of 5.28 mA.[1] The functional test procedure required an assigned set point of 4.96 mA, which corresponds to a water level in the RWT of 36 in above the tank bottom.[1] The less conservative set point dictated by the functional test procedure was applied to all 4 channels of the Unit 1 RWT level instrument bistables.

In January 1993, an engineering calculation was issued by the St. Lucie engineering staff to change the span of the RWT level measurement loop such that zero feet would reference the bottom of the tank. The level instruments are actually 1 ft above the bottom of the RWT and the 0 ft mark previously referenced the height of the level instruments. Before the change in the span of the RWT level, the set point procedure correctly

initiated the RAS 48 in above the true bottom of the RWT (36 in set point + 12 in instrument height).[1] After the change in the span of the RWT level, the set point procedure was not changed. Subsequently, the set point procedure incorrectly initiated RAS at 36 in above the bottom of the RWT.

Additional Event-Related Information

The RAS signal causes the suction source for the ECCS system to transfer from the RWT at the end of the injection phase to the containment sump. This begins the recirculation phase. The injection phase is not influenced by the set point error and, under worst case conditions, the injection phase will last for at least the initial 20 min following a LOCA.[1] When the RAS set point is reached, the containment sump isolation valve is opened in ~30 s and the RWT isolation valve is closed in ~90 s. The operator is directed by the emergency operating procedure (EOP) to initiate recirculation manually if the automatic signal fails to initiate recirculation when RWT level reaches 48 in from the bottom of the tank.[1]

The ECCS suction pipe outer diameter is ~24 in. The top of the inner diameter of the ECCS suction pipe is 42.25 in above the bottom of the RWT. The bistable set point error would delay the automatic initiation of recirculation until the water level in the RWT was ~6 in below the top of the ECCS suction piping. Without operator action, a condition known as open channel flow would occur. From calculations made by the licensee, open channel flow conditions could not support full ECCS flow (~13,000 gpm). The mismatch between available open channel flow and full ECCS flow would drain down the level in the ECCS suction piping leading to air ingestion and reduction of the available net positive suction head for the ECCS pumps. Under these conditions, chugging flow would result until the ECCS pumps became air bound. This interruption of ECCS flow would prevent a further drain down of the RWT and the lower RAS set point might never be reached.[1]

The licensee further calculated that ECCS flows below 7,000 gpm could be supported by open channel conditions until the lower RAS set point was reached.[1] This limits concern for ECCS failure, as a result of the less conservative RWT bistable set point, to a LBLOCA. Additionally, the peak containment temperature and pressure will be mitigated within the first 20 min following a LOCA; when containment pressure falls below 5 psig, operators are directed by the EOPs to secure the containment spray pumps. Because the spray pumps provide more than 6,000 gpm flow (6,750 gpm, Ref. 2, Table 6.3-6), this would reduce the total ECCS flow required below 7,000 gpm before reaching open channel flow conditions.

Modeling Assumptions

This event was modeled as a failure of the automatic recirculation actuation signal for a LBLOCA during a 1-year period. A failure to initiate recirculation could air bind all ECCS pumps following a LBLOCA, leading to core damage. Therefore, the significance of this event can be estimated directly from the change in ECCS pump failure probabilities because of the improper RAS set point and the probability of a LBLOCA during a 1-year period. The St. Lucie individual plant examination (IPE) estimates the frequency of a LBLOCA to be 2.7×10^{-4}/year.

The increased probability of core damage because of ECCS pump failures is estimated by considering the probability that the operators will fail to initiate recirculation manually when the water level in the RWT drops

to 48 in and the probability that the operators will fail to secure the spray pumps when conditions allow. Hence, the increased probability of core damage is

P(LBLOCA) × P(operator fails to initiate recirculation manually) × P(operator fails to secure spray pumps)

Operator fails to initiate recirculation manually

Manual initiation of recirculation following a LOCA is directed by the EOPs if the automatic RAS signal should fail. The operators had correct indication of the RWT level in the control room and the EOP directs operator attention to the RWT level when the water level drops to 6 to 8 ft. Manual initiation of recirculation when directed by the EOP would prevent any damage to the ECCS pumps because of the RAS set point error. It was assumed that the high-pressure injection (HPI) pumps would not be required to prevent core damage for the LBLOCA of concern in this event. Therefore, the low-pressure injection (LPI) pumps were expected to be the key to preventing core damage. Ref. 1 indicated 240 s were available after reaching the nominal RAS set point before the LPI pumps would fail. (The HPI pumps would fail within 90 s.) A simulator test suggested that operating crews maintained positive control of the RWT level and recognized the need to manually initiate recirculation ~40 s after the nominal automatic set point was reached. The *Human Reliability and Safety Analysis Data Handbook*[3] suggests that for post-accident task actions, a 1-min delay for travel/manipulation should be allowed (Ref. 3, p. 66). So, 100 s was considered the median time response for an operating crew. Allowing ~10 s for the containment sump valves to reposition enough to allow significant flow, the critical time was assumed to be 230 s before the LPI pumps would fail because of gas binding (i.e., 240 s − 10 s). The probability of the operating crew failing to initiate recirculation manually can be estimated by assuming that the failure probability can be represented as a time-reliability correlation (TRC) as described in *Human Reliability Analysis*.[4] Operator response was assumed to be rule-based and without hesitancy. For the 230-s period of interest, a failure probability of 1.2×10^{-1} was estimated.

Operator fails to secure spray pumps

If total ECCS flow could be reduced below 7,000 gpm before reaching the automatic RAS set point, then the potential for ECCS pump failure, because of the non-conservative bistable set point, is eliminated (Ref. 1, p. 9). Analysis of containment pressure curves in the FSAR[2] indicates that containment pressure should be reduced to 5 psig in ~20 min following a LBLOCA. If the RWT level were at the technical specification minimum level [401,800 gal (Ref. 2)] and ECCS flows were at the maximum indicated by reference 1 (13,000 gpm), then a minimum of 25.8 min would be available until the intended automatic RAS set point is reached [66,200 gal (Ref. 2)]. The EOPs direct the operating crew to secure the containment spray pumps when containment pressure returns below 5 psig. The *Human Reliability and Safety Analysis Data Handbook*[4] suggests that for post-accident task actions, a 5-min delay for diagnosis/analysis be allowed and a 1-min delay for travel/manipulation be allowed (Ref. 4, p. 66). With this median response prediction, the probability of the operating crew failing to secure the spray pumps can be estimated by assuming that the failure probability can be represented as a TRC as described in *Human Reliability Analysis*.[3] Operator response was assumed to be rule-based, but with hesitancy, because conditions to secure the spray pumps are not met immediately. This operator action was assumed to be independent of the operator action to manually establish the recirculation line-up, because of the way the EOP directs these activities be performed and the independence of the instrumentation required. Allowing 0.2 min for equipment response, a failure probability of 5.2×10^{-1}

was estimated for the 5.6-min period of interest [(25.8 - 20 - 0.2) min]. A sensitivity study on the operator response time is presented at the end of the Analysis Results section.

Analysis Results

The nominal core damage probability (CDP) over a 1-year period estimated using the IRRAS model for St. Lucie is 1.7×10^{-5}. The increase in the CDP (i.e., the importance) for this event over a 1-year period because of an improper RAS set point is 1.7×10^{-5}. Hence, the less conservative RAS set point results in an estimated conditional core damage probability (CCDP) for a 1-year period of operation of 3.4×10^{-5}. A large uncertainty is associated with this estimate because it relies on an estimated LBLOCA frequency (none have occurred) and estimations of operator response during accident conditions.

The dominant core damage sequence (sequence 4 in Fig. 1) for this event involves

- a LBLOCA,
- successful injection from the safety injection tanks,
- a failure of automatic RAS signal,
- operator fails to manually backup the RAS, and
- operators fail to secure the containment spray pumps before the RAS set point is reached.

The conditional core damage probabilities are shown in Table 1, while Table 2 lists the sequence logic associated with the sequences listed in Table 1. Table 3 provides the definitions and failure probabilities for event tree branch points in Fig. 1.

The HPI pumps were expected to fail 90 s following a failure of the RAS.[1] If it is assumed that HPI pump failure impacts adequate decay heat removal following some LOCA events of concern, the probability of the operating crew failing to initiate recirculation manually before HPI pump failure can be estimated by assuming that the failure probability can be represented as a TRC as described in *Human Reliability Analysis*.[3] Operator response was assumed to be rule-based and without hesitancy. Again allowing 10 s for valves to reposition and allow significant flow, a failure probability of 6.2×10^{-1} was estimated for the 80-s period of interest. In this case, the estimated increase in the core damage probability is 8.7×10^{-5} for a 1-year period with the RAS set point too low. The CCDP then is increased to 1.0×10^{-4}.

It could be assumed that the operator focus on current EOP steps and accident conditions overall would preclude any consideration of securing the containment spray pumps so early in the event. In this case, the probability of failing to secure the spray pumps before reaching the RAS set point would be 1.0. The estimated CCDP for this event increases to 4.9×10^{-5} for a 1-year period with the RAS set point lower than the design basis. On the other hand, with increased attention on RWT water management per the EOP, it is possible that the operating crew would be quick to secure containment spray pumps when it became permissible. If the median response were considered to be 3 min instead of the 6 min assumed previously, the probability that the operator fails to secure the spray pumps is 1.9×10^{-1}. The estimated CCDP for this event decreases to 2.3×10^{-5} for a 1-year period with the RAS set point too low. Similar results are obtained if the water level in the

RWT is assumed to start well above the minimum water level allowed by Technical Specifications when demanded.

Acronyms

CCDP	conditional core damage probability
CDP	core damage probability
ECCS	emergency core cooling system
EFW	emergency feedwater system
EOP	emergency operating procedure
ESFAS	engineered safety feature actuation system
HPI	high-pressure injection
IPE	integrated plant examination
IRRAS	Integrated Reliability and Risk Analysis System
LBLOCA	large-break loss-of-coolant accident
LOCA	loss-of-coolant accident
LPI	low-pressure injection
RAS	recirculation actuation signal
RWT	refueling water tank
TRC	time-reliability correlation

References

1. LER 335/97-011, Rev. 0, "Non-Conservative Recirculation Actuation Signal Set Point Resulted in Operation Prohibited by the Technical Specifications," December 2, 1997.

2. St. Lucie, *Final Safety Analysis Report (Updated Version)*.

3. D. I. Gertman and H. S. Blackman, *Human Reliability and Safety Analysis Data Handbook*, John Wiley and Sons, 1994.

4. E. M. Dougherty and J. R. Fragola, *Human Reliability Analysis*, John Wiley and Sons, 1988.

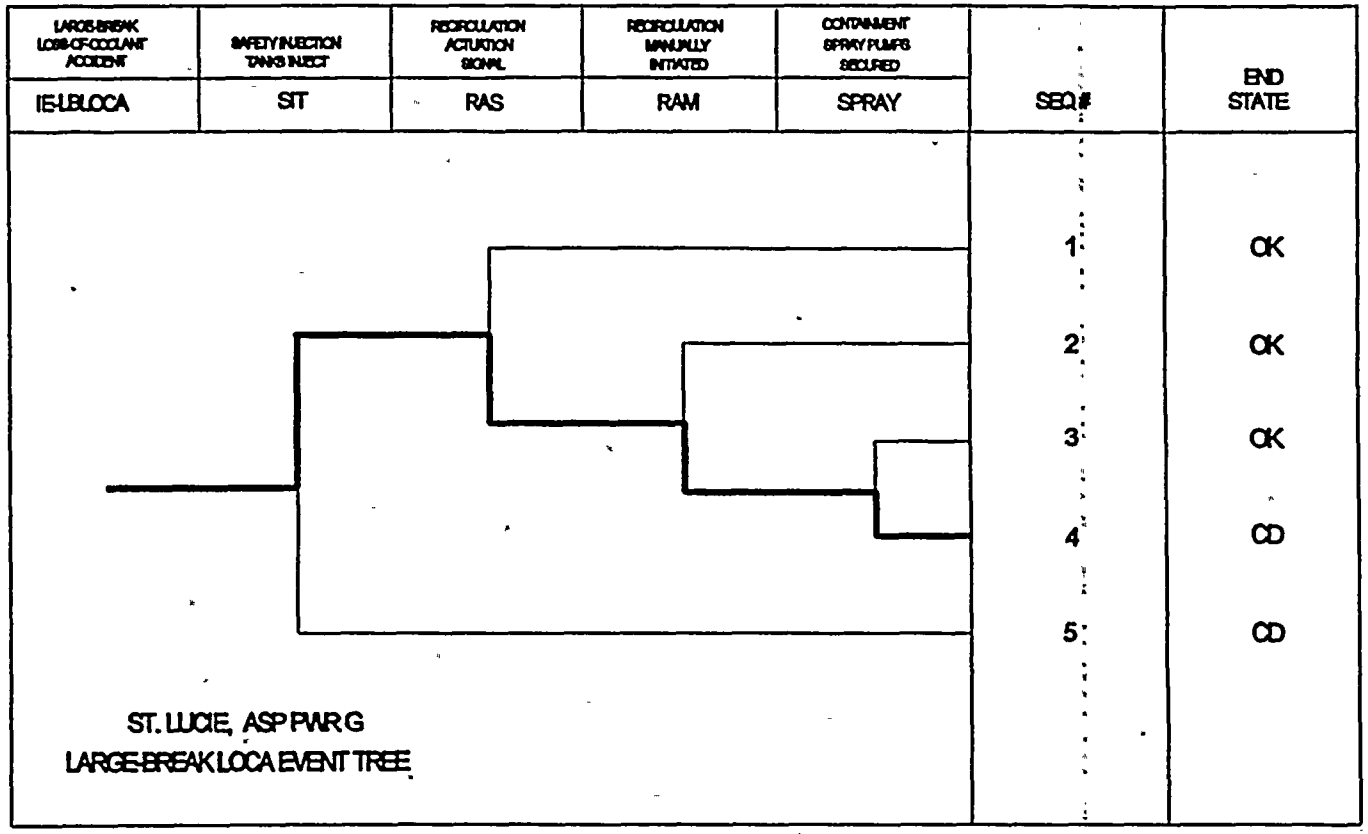

Fig. 1. St. Lucie large-break LOCA event tree.

Table 1. Sequence Conditional Probabilities for LER No. 335/97-011

Event tree name	Sequence number	Conditional core damage probability (CCDP)	Core damage probability (CDP)	Importance (CCDP-CDP)	Percent contribution
LBLOCA	4	3.4 E-005	1.7 E-005	1.7 E-005	99.9
LBLOCA	5	2.7 E-008	2.7 E-008	0.0 E+000	0.1
Total (all sequences)		3.4 E-005	1.7 E-005	1.7 E-005	

Table 2. Sequence Logic for Dominant Sequences for LER No. 335/97-011

Event tree name	Sequence number	Logic
LBLOCA	4	/SIT, RAS, RAM, SPRAY
LBLOCA	5	SIT

Table 3. System Names for LER No. 335/97-011

System name	Description	Failure probability
IE-LBLOCA	Initiating Event–LBLOCA	2.7 E-004
RAS	The RAS bistable fails to change the safety injection mode from injection to recirculation	1.0 E+000
RAM	Operator fails to manually initiate recirculation when the water level in the RWT drops to 48 in	1.2 E-001
SIT	The safety injection tanks fail to inject water properly	1.0 E-004
SPRAY	Operator fails to secure the spray pumps when conditions allow	5.2 E-001

GUIDANCE FOR LICENSEE REVIEW OF PRELIMINARY ASP ANALYSIS

Background

The preliminary precursor analysis of an operational event that occurred at your plant has been provided for your review. This analysis was performed as a part of the NRC's Accident Sequence Precursor (ASP) Program. The ASP Program uses probabilistic risk assessment techniques to provide estimates of operating event significance in terms of the potential for core damage. The types of events evaluated include actual initiating events, such as a loss of off-site power (LOOP) or loss-of-coolant accident (LOCA), degradation of plant conditions; and safety equipment failures or unavailabilities that could increase the probability of core damage from postulated accident sequences. This preliminary analysis was conducted using the information contained in the plant-specific final safety analysis report (FSAR), individual plant examination (IPE), and the licensee event report (LER) for this event.

Modeling Techniques

The models used for the analysis of 1995 and 1996 events were developed by the Idaho National Engineering Laboratory (INEL). The models were developed using the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software. The models are based on linked fault trees. Four types of initiating events are considered: (1) transients, (2) loss-of-coolant accidents (LOCAs), (3) losses of offsite power (LOOPs), and (4) steam generator tube ruptures (PWR only). Fault trees were developed for each top event on the event trees to a supercomponent level of detail. The only support system currently modeled is the electric power system.

The models may be modified to include additional detail for the systems/components of interest for a particular event. This may include additional equipment or mitigation strategies as outlined in the FSAR or IPE. Probabilities are modified to reflect the particular circumstances of the event being analyzed.

Guidance for Peer Review

Comments regarding the analysis should address:

- Does the "Event Description" section accurately describe the event as it occurred?
- Does the "Additional Event-Related Information" section provide accurate additional information concerning the configuration of the plant and the operation of and procedures associated with relevant systems?
- Does the "Modeling Assumptions" section accurately describe the modeling done for the event? Is the modeling of the event appropriate for the events that occurred or that had the potential to occur under the event conditions? This also includes assumptions regarding the likelihood of equipment recovery.

Appendix H of Reference 1 provides examples of comments and responses for previous ASP analyses.

Criteria for Evaluating Comments

Modifications to the event analysis may be made based on the comments that you provide. Specific documentation will be required to consider modifications to the event analysis. References should be made to portions of the LER, AIT, or other event documentation concerning the sequence of events. System and component capabilities should be supported by references to the FSAR, IPE, plant procedures, or analyses. Comments related to operator response times and capabilities should reference plant procedures, the FSAR, the IPE, or applicable operator response models. Assumptions used in determining failure probabilities should be clearly stated.

Criteria for Evaluating Additional Recovery Measures

Additional systems, equipment, or specific recovery actions may be considered for incorporation into the analysis. However, to assess the viability and effectiveness of the equipment and methods, the appropriate documentation must be included in your response. This includes:

- normal or emergency operating procedures.*
- piping and instrumentation diagrams (P&IDs),*
- electrical one-line diagrams,*
- results of thermal-hydraulic analyses, and
- operator training (both procedures and simulator),* etc.

Systems, equipment, or specific recovery actions that were not in place at the time of the event will not be considered. Also, the documentation should address the impact (both positive and negative) of the use of the specific recovery measure on:

- the sequence of events,
- the timing of events,
- the probability of operator error in using the system or equipment, and
- other systems/processes already modeled in the analysis (including operator actions).

For example, Plant A (a PWR) experiences a reactor trip, and during the subsequent recovery, it is discovered that one train of the auxiliary feedwater (AFW) system is unavailable. Absent any further information regrading this event, the ASP Program would analyze it as a reactor trip with one train of AFW unavailable. The AFW modeling would be patterned after information gathered either from the plant FSAR or the IPE. However, if information is received about the use of an additional system (such as a standby steam generator feedwater system) in recovering from this event, the transient would be modeled as a reactor trip with one train of AFW unavailable, but this unavailability would be

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* Revision or practices at the time the event occurred.

mitigated by the use of the standby feedwater system. The mitigation effect for the standby feedwater system would be credited in the analysis provided that the following material was available:

- standby feedwater system characteristics are documented in the FSAR or accounted for in the IPE,
- procedures for using the system during recovery existed at the time of the event,
- the plant operators had been trained in the use of the system prior to the event,
- a clear diagram of the system is available (either in the FSAR, IPE, or supplied by the licensee),
- previous analyses have indicated that there would be sufficient time available to implement the procedure successfully under the circumstances of the event under analysis,
- the effects of using the standby feedwater system on the operation and recovery of systems or procedures that are already included in the event modeling. In this case, use of the standby feedwater system may reduce the likelihood of recovering failed AFW equipment or initiating feed-and-bleed due to time and personnel constraints.

Materials Provided for Review

The following materials have been provided in the package to facilitate your review of the preliminary analysis of the operational event.

- The specific LER, augmented inspection team (AIT) report, or other pertinent reports.

- A summary of the calculation results. An event tree with the dominant sequence(s) highlighted. Four tables in the analysis indicate: (1) a summary of the relevant basic events, including modifications to the probabilities to reflect the circumstances of the event, (2) the dominant core damage sequences, (3) the system names for the systems cited in the dominant core damage sequences, and (4) cut sets for the dominant core damage sequences.

Schedule

Please refer to the transmittal letter for schedules and procedures for submitting your comments.

References

1. L. N. Vanden Heuvel et al., Precursors to Potential Severe Core Damage Accidents: 1994, A Status Report, USNRC Report NUREG/CR-4674 (ORNL/NOAC-232) Volumes 21 and 22, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory and Science Applications International Corp., December 1995.