Station Support Department

10CFR50.54(f)

PECO NUCLEAR

A UNIT OF PECO ENERGY

PECO Energy Company
Nuclear Group Headquarters
965 Chesterbrook Boulevard
Wayne, PA 19087-5691

June 28, 1996

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: Limerick Generating Station, Units 1 and 2
Response to Request for Additional Information Regarding Review of Individual Plant Examination of External Events

REFERENCE: Letter, G. A. Hunger, Jr (PECO Energy) to USNRC
Dated January 31, 1996

Dear Sir:

Attached is our response to your Request for Additional Information dated December 22, 1995, regarding review of the Limerick Generating Station (LGS), Units 1 and 2, Individual Plant Examination of External Events. The attachment to this letter provides a restatement of the questions, followed by our response.

Additionally, the remaining actions cited in the referenced letter (i.e., several housekeeping and maintenance concerns and administrative control of additional doors as "fire" doors) have been completed.

If you have any questions, please contact us.

Very truly yours,

G. A. Hunger, Jr.

G. A. Hunger, Jr., Director
Licensing

JLP/bgr

030089

Attachment

cc: T. T. Martin, Administrator, Region I, USNRC
N. S. Perry, USNRC Senior Resident Inspector, LGS
R. R. Janati, Commonwealth of Pennsylvania

COMMONWEALTH OF PENNSYLVANIA :

: SS

COUNTY OF CHESTER :

D. B. Fetters, being first duly sworn, deposes and says:

That he is Vice President of PECO Energy Company; the Applicant herein; that he has read the enclosed response to the NRC request for additional information dated December 22, 1995, concerning the Limerick Generating Station Individual Plant Examination of External Events, and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.

Vice President

Subscribed and sworn to
before me this 26th day
of June 1996.

Notary Public

Notarial Seal
Mary Lou Skrocki, Notary Public
Tredyffrin Twp., Chester County
My Commission Expires May 17, 1999
Member, Pennsylvania Association of Notaries

Docket Nos. 50-352
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Response to Request for Additional Information
Limerick Generating Station, Units 1 and 2
Individual Plant Examination of External Events (IPEEE)

Seismic Analysis

Question

1. Limerick has been identified in NUREG-1407 as a plant belonging to the 0.3g focused-scope seismic margin assessment bin; hence, the reduced-scope evaluation at 0.15g, as performed in the LGS seismic IPEEE, does not conform to the review guidance in NUREG-1407 and Supplement 4 to Generic Letter (GL) 88-20. Accordingly:

   a. Provide a list of structures, systems, and components (including Safe Shutdown Equipment List (SSEL) items and containment systems equipment) that did not screen at 0.3g.

   b. Provide the basis for disposition of each such item at 0.3g. Indicate if the Severe Accident Risk Assessment (SARA) capacity calculations continue to be valid; discuss any other basis that has been used for component disposition, including any results of new calculations.

   c. Provide an evaluation of masonry/block walls that may influence the performance of success path components.

   d. Provide an evaluation of flat-bottomed tanks, as requested in NUREG-1407 and GL 88-20 for focused-scope plants.

Response

1. In a letter dated July 28, 1994, PECO Energy notified the NRC that in light of the revised Lawrence Livermore National Laboratory's seismic hazard curves, a reduced scope seismic margins evaluation would be performed at LGS. We believe our reduced-scope evaluation meets the intent of Supplement 4 to GL 88-20 for increased understanding of seismic severe accident behavior and identification of seismic severe accident vulnerabilities.

Additionally, Supplement 5 to GL 88-20 states that "Licensees who previously submitted their requests to modify their seismic IPEEEs may choose not to submit any response to this generic letter supplement; should that be the case, NRC will respond separately to their previous requests." As such, we await NRC's response to our July 28, 1994 letter.

Question

2. Provide a list of "bad actor" relays which are installed in the preferred and alternate safe shutdown (SSD) paths for Limerick, including in your response all of the safe shutdown (SSD) frontline systems in Section 3.1.2.5.1 of your submittal, and SSD support systems identified in Section 3.1.2.5.2 of your submittal. For each "bad actor" relay identified, discuss the impact of malfunctions of the relay on integrity of the preferred and alternate shutdown paths.

Response

2. Identification of "bad actor" relays is not required in a reduced scope seismic evaluation. However, in 1989, prior to the initiation of the LGS IPEEE project, PECO Energy concluded a settlement with Limerick Ecology Action (LEA) to evaluate relays and circuit breakers in those systems that would be used to achieve and maintain safe shutdown following a seismic event. The objective of this agreement was to identify those relays and circuit breakers that would be susceptible to relay chatter during a seismic event beyond and outside the applicable regulatory requirements.

The agreement with LEA was to perform this evaluation independently of the Nuclear Regulatory Commission's request for nuclear power plant licensees to perform an IPEEE as requested in Supplement 4 to GL 88-20. This evaluation was completed in 1991. The evaluation was consistent with the methodology in EPRI NP-6041-SL. Five chatter-prone relay types were identified as being used in risk significant systems. They are listed below with their resolution.

General Electric PVD21-B

The normally open 87H contact of this relay may chatter when it is energized or in the closed position. LGS uses this contact in parallel with the 87L contact to pick up the 4 kV bus lock-out relay in order to trip and lock-out all circuit breakers on the associated 4 kV bus. Opening of the 87H contact will not affect the relay function since the 87L contact has a lower setpoint and will remain closed to assure that the trip and lock-out function is accomplished. Therefore, contact chatter is acceptable at LGS for this relay type.

Westinghouse SV

This protective relay type is used as the diesel generator "ready to load" voltage permissive relay. One contact to the SV relay is in series with the "ready to load" frequency permissive relay to pick up the diesel generator "ready to load" (RL) relay. The RL relay supplies a close permissive for the diesel generator output circuit breaker. If the SV contact inadvertently closed, there would be no adverse effect since the frequency permissive or a circuit breaker close signal is not present. If the SV contact inadvertently opened, there would be no adverse effect because once the circuit breaker is closed, no action by the RL relay can cause the circuit breaker to open. Therefore, contact chatter is acceptable at LGS with this relay type.

General Electric HFA

PECO Energy identified these relays during the documentation review to be chatter-prone. The documentation review identified that if the HFA relay was field converted to a NC contact configuration, and was not subsequently adjusted for proper contact and wipe, it might experience a problem.

On the HFA-51/151 relay type, only the normally closed relay contact is a concern. A review of LGS plant drawings verified that the normally closed relay contact for this type relay is not used in a safety-related application. Therefore, since all risk significant systems analyzed are safety-related, this relay has no effect on the identified risk significant systems.

General Electric HMA

The GE HMA relay is an auxiliary type relay that was identified from the PECO Energy documentation research as a relay that might be prone to chatter. The documentation search identified that on this relay type, only the de-energized, normally closed, relay contact is prone to chatter. A review of LGS plant drawings determined that the de-energized, normally closed, relay contact is used in only one application. The normally closed relay contact is only used for indication of RCIC turbine trip at the Remote Shutdown Panel. This usage will not adversely affect the RCIC system.

Westinghouse Type-W Contactor

A review of LGS plant drawings verified that only one contactor model type from the PECO Energy documentation research is used at LGS in a risk significant system. This contactor is a Westinghouse Type-W contactor. The Type-W contactors are used in safety-related DC MCCs 10D201, 10D202, 10D203, 20D201, 20D202, and 20D203 and were identified as a potential chatter problem as a result of auxiliary contact chatter during original MCC shake tests. Review of manufacturer documentation

determined that the contactor performance is acceptable. The original seismic testing by ANCO, where the contactor exhibited contact chatter, had test inconsistencies that may have adversely affected the test results. Alternate seismic testing, proprietary to Westinghouse, confirms that the contactors are qualified for use in LGS without any contact chatter.

Question

3. The alternate shutdown success path uses Low Pressure Coolant Injection/Residual Heat Removal (LPCI/RHR) "C" and "D" loops for inventory control and the "B" loop for suppression pool cooling. Identify and explain how the LPCI/RHR system is used in the alternate shutdown path (indicating what trains of the system must operate in order for the alternate shutdown path to succeed), and explain how non-seismic failures were accounted for in this regard.

Response

3. The Limerick RHR system for each unit has four trains. Each train has its own injection path (four injection points total per unit). In the alternate success path, the "C" and "D" RHR trains are used for dedicated LPCI mode, taking suction from the suppression pool and injecting into the reactor vessel. The "B" RHR train takes suction from the suppression pool and passes it through the "B" RHR heat exchanger (suppression pool cooling function) to the reactor. The water returns to the suppression pool via an open SRV (reactor heat removal function). The complete pathway from the suppression pool through the RHR heat exchanger to the reactor vessel and through the SRV to the suppression pool is referred to as alternate shutdown cooling and is shown in Figure S3.1 (attached).

Non-seismic failures were accounted for in the use of both "C" and "D" trains of RHR for level control. If necessary, the "D" RHR pump may be aligned in non-LPCI modes using the B RHR Heat Exchanger. Failure of an SRV is not significant because of the number (5 ADS SRVs and an additional 9 non-ADS SRVs) available.

Question

4. Provide a copy of the "Success Path Logic Diagram" (SPLD) which is referred to in Section 3.1.2.5.4.1 of the IPEEE submittal report.

Response

4. Figure S4.1 (attached) is the SPLD for Limerick.

Question

5. List all shutdown-path-related non-seismic failures and human actions, together with their failure rates, noting any lack of redundancies. Also provide a discussion concerning the anticipated effects of the seismic margin earthquake on rates of operator errors which may impact the integrity of the preferred and alternate success paths. Identify the locations at which operator actions must be performed.

Response

5. The Limerick seismic analysis is a seismic margins analysis (SMA) and as such does not explicitly itemize either non-seismic failures or human actions as a seismic PSA would.

Non-seismic failures were accounted for in the analysis by providing alternates for single train/low reliability systems (e.g., providing RCIC as an alternate to HPCI) per EPRI SMA criteria. Also, system diversity was maximized between the primary and alternate paths so that an equipment failure in one path would leave the other path available.

As noted in the IPEEE report (Section 3.1.2.5.4.2), the actions called for in the SMA are nearly identical to those in the Limerick IPE. The IPE human reliability analysis takes into account operators and other personnel performing actions required for safe plant shutdown under the stress of plant transients ranging from a manual shutdown to an ATWS or large break LOCA. The rate of operator error should not vary greatly from the IPE values due to the trigger event being a seismic event. The actions involved are proceduralized and trained actions and are performed primarily from the control room or locations in other seismically qualified buildings; and thus, access to non-control room locations is judged to be similar to other loss of offsite power scenarios.

Question

6. Indicate to what extent the cabinet internals were checked for adequate installation, and provide the results of these checks.

Response

6. Cabinet internals were checked for adequate installation in accordance with guidance provided in EPRI NP-6041. The implementation of this guideline has been discussed in the submittal report in the following sections:

3.1.4.1.4.2 **Functional Capability**

"...To address the functional capability of the equipment, certain equipment caveats based on earthquake experience data and Appendices A and F of EPRI NP-6041 were reviewed. As a minimum, the caveats noted in Appendix F of EPRI NP-6041 and Part B of the SEWS sheets and under Section "i" of the walkdown checklists were reviewed during the walkdown."

3.1.4.1.4.3 **Anchorage Adequacy**

"During the walkdown, the equipment anchorage (type, number, size, etc.) was reviewed for conformance with the design documents and qualification reports..."

3.1.4.1.4.5 **Sampling**

"A detailed review of at least one component for each equipment type in an equipment class was performed... However, all accessible components were 'walked by.' The 'walk by' considered the three parts of equipment assessment (functional capability, anchorage, and seismic interaction) but emphasized a confirmation that the construction pattern was typical and looked for unique seismic interaction concerns for each equipment item..."

Results of these checks have been documented on the Screening and Evaluation Work Sheets (SEWS) that were prepared for this project. A summary of each equipment category evaluation is presented in Table 3.1.4-2 of the submittal report, and Section (18) of this table discusses Control Panels and Cabinets. Additionally, the 'walk by' that was performed for these cabinets included opening the doors for an internals 'walk by' for all cabinets that could be opened, estimated to be at least 80% of the total population.

Question

7. Section 3.1.5.1 of the submittal references EPRI NP-7498 as providing the technical approach used for containment evaluation in the LGS seismic IPEEE. Please provide a copy of EPRI NP-7498.

Response

7. Attached is a copy of EPRI NP-7498.

Question

8. NUREG-1407 requests an evaluation of seismic-fire interactions to consider: (i) seismic-induced fires, (ii) seismic actuation of fire suppression systems, and (iii) seismic degradation/failure of fire suppression systems. Examples of items found in past studies include (but are not limited to):

- Unanchored CO_2 tanks or bottles
- Sprinkler standoffs penetrating suspended ceilings
- Fire pumps unanchored or on vibration isolation mounts
- Mercury or "bad actor" relays in fire protection system (FPS) actuation circuitry
- Weak or unanchored 480V or 600V (non-safety related) electrical cabinets in close proximity to essential safety equipment (i.e., as potential fire sources)
- Use of cast iron fire mains to provide fire water to fire pumps

NUREG-1407 suggests a walkdown as a means of identifying any such items.

Please provide the related results of your seismic-fire interaction study. Provide guidelines given to walkdown personnel for evaluating these issues (if they exist).

Response

8. Walkdowns were performed as suggested above to evaluate: unanchored CO_2/ Halon tanks, Mercury or "bad actor" relays, fire pump mounts, cast iron fire mains, electrical cabinet mounting, unanchored O_2 or H_2 bottles, and sprinkler system interactions. Other than those concerns documented in Section 4.8 of the IPEEE submittal, there were no other concerns identified during the walkdown process.

Question

9. Failure of room cooling has been identified as an important failure mode in past probabilistic risk assessment studies. However, in Table 3.1.2-1 ("Preferred and Alternate Shutdown Paths"), pump room cooling is not mentioned. Discuss the need for pump room cooling for High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), and RHR for achieving and maintaining safe shutdown conditions for 72 hours, and discuss the extent to which pump room cooling considerations were addressed during the walkdowns.

Response

9. Room cooling for ECCS systems is listed as a support system in Section 3.1.2.5.2 but was not included in the referenced table. This was a clerical oversight. Room cooling is provided by unit coolers supported by ESW as a heat sink. Since the time of the data cutoff for the IPEEE analysis, the requirement for room cooling for HPCI and RCIC has been eliminated through analysis and modification. The RHR system still requires room cooling and the required components were included on the SPCL and were reviewed during the walkdown.

Question

10. Discuss the performance of containment cooling and hydrogen control systems at the 0.3g Peak Ground Acceleration (PGA) review level earthquake.

Response

10. Heat removal from containment is accomplished via suppression pool cooling as indicated in the success path discussion in the IPEEE report. These suppression pool tooling components were walked down with the results as described in Table 3.1.4-2. Both containment cooling in the form of the drywell unit coolers and hydrogen control (post-LOCA recombiners at LGS) are not required to prevent early containment failure and thus were not credited in the Limerick IPEEE containment evaluation. Hydrogen control is only required if there is fuel damage which the success paths are designed to prevent. See also response to questions above.

Question

11. Discuss the ability of the preferred and alternate shutdown paths to respond to medium and large Loss of Coolant Accident (LOCAs) that may result from stuck-open safety-relief valves.

Response

11. Medium and large LOCAs are not required to be considered in the seismic margins analysis per EPRI NP-6041-L, Section 3. However, if an SRV would stick open, reactor depressurization would result. Reactor depressurization is required for the alternate path to allow LPCI injection. Thus, a stuck open SRV only moves the plant from the preferred to the alternate success path.

Fire Analysis

Questions

1. The submittal (Section 4.0) states that "quantification of fire induced safe shutdown system unavailability was obtained by propagating fire induced system failures through a modified Probalistic Safety Analysis (PSA) plant model." Identify which plant model was used (e.g., was it the LGS IPE plant model or some other?), and explain how the model was modified. In addition, discuss how this model was verified as accurately representing the plant configuration and its response to fire initiating events.

Response

1. As identified in Section 4.6, the quantification was performed using the 1993 LGS PSA model. The 1993 LGS PSA model updated the IPE model with plant equipment and procedure changes that occurred after the freeze date of the IPE. Updates of the LGS PSA model have occurred regularly to assure equipment and procedure changes are reflected in the risk profile. Specific changes associated with this update involved inclusion of feedwater deep bed demineralizer system, update of initiating event frequencies and maintenance terms based on plant data, revision of instrument

miscalibration events to better reflect plant practice and restructuring to increase ease of model solution and application of models. Verification of the modelling and database changes was performed independent of the initiator. Sensitivity analyses were performed to assure the model and the new results correctly reflected the current plant configuration. The changes to the PSA model for the quantification of safe shutdown system unavailability are provided in Section 4.6 of the IPEEE. See also response to Question F5 below.

Question

2. The submittal states (Section 4.0.2), "Fire-induced disabling of the control room Heating Ventilating and Air Conditioning (HVAC) is not assumed to result in loss of control room habitability. The control room is constantly manned, and a heating or cooling failure would be corrected in a timely manner according to the applicable procedure." Identify the fire areas from which a fire-induced disabling of the control room HVAC could occur and, comparing these scenarios with the applicable procedure, verify that the procedure steps would result in recovery of the control room HVAC system in time to prevent loss of habitability. Specify the criteria used to judge whether loss of habitability has occurred (e.g., a room temperature criterion). Further, demonstrate that no system or component failures would result from fire-induced loss of control room HVAC prior to loss of habitability. If such failures are possible prior to loss of habitability, demonstrate that the failures are recoverable or that their consequences can be adequately controlled by existing procedures.

Response

2. A fire induced loss of control room HVAC not resulting in the loss of control room habitability is an analyzed condition for an existing safe shutdown operator manual action. Safe shutdown components located in the control room were determined to be able to function continuously at a temperature of 120°F. A room heat-up analysis was performed for a period of 9 hours, at which time the temperature in the control room rises to 115.3°F. The procedure to maintain the control room below 120°F is a safe shutdown operator manual action that opens doors at entrances to the control room from a stairway and the turbine building then places portable fans with 20 foot flexible duct (to place the fans outside the control room to minimize the noise) in a suction/exhaust configuration. The fans are powered by a portable generator. All equipment is staged and maintained as required for safe shutdown operator manual actions. The operator manual action is deemed achievable within the time allowable to maintain the control room temperature below 120°F, thereby mitigating the potential for safe shutdown component failures.

Question

3. The submittal states (Section 4.0.2), "Fire brigade response time is assumed to be equal to the manual fire suppression time." This assumption is not considered an acceptable approach. An assessment of manual suppression times must include: (a) time to detection, (b) brigade response time, and (c) extinguishment time. Provide the effect on the screening analysis by considering all of these components of fire suppression time.

Response

3. The fire brigade response time as analyzed in the submittal included: (a) time to detection and (b) brigade response time; however, fire extinguishment was assumed to be concurrent with the arrival of the fire brigade. Fire brigade response and credit for manual suppression has been re-evaluated with 20 minutes allotted for fire extinguishment. This allows for 30 minutes from fire detection to fire extinguishment. Due to the conservatism included in the original fire brigade response and manual suppression calculations, the results of the calculations were unaffected by the additional time for fire extinguishment.

   Manual suppression was credited in the screening analysis of the following plant compartments: 44, 45, 47, 64, 67, 68, and 70. The results of the screening analysis for these compartments remains as stated in Section 4.4.1.3 of the submittal.

Question

4. The submittal states (Section 4.0.2), "For any analyzed fire only one worst-case spurious actuation or signal is postulated (with the exception of Hi-Low pressure interfaces). Operator actions and repairs may be available to correct the actuation or signal or redundant equipment may be utilized in order to provide the required safe shutdown function. The analysis of spurious operations is identical to that performed for Appendix R analyses." Explain how the "one worst-case spurious actuation or signal" is postulated (e.g., Is it based on failure modes and effects analysis, on expert judgment, or on some criteria?). Justify the implicit assumption that multiple failures are not possible or are unimportant, and explain the basis for any related evaluations.

Response

4. The "worst-case spurious actuation or signal" postulated is the evaluation of a control or power circuit for the effects of single conductor fire induced faults (i.e., short, open or ground) that have the potential to render a required safe shutdown component inoperable or in an undesired position. Safe shutdown components that support trains of equipment not required to be operational to effect safe shutdown in the area of concern are not analyzed for spurious operation. The evaluation of Hi-Low pressure interfaces includes simultaneous multiple conductor faults (e.g., simultaneous 3-phase shorts in the proper phase rotation). The conductor faults that have the ability to affect the operation of required safe shutdown equipment are mitigated. That is, for a valve that is required to remain in its normal position, an open or ground in the control circuit would not change the state of the valve; however, if a short could change the state of the valve, the short being worst case would require mitigation by an operator manual action, repair or by the use of redundant equipment. Combinations of, or multiple shorts, opens, and grounds required to occur simultaneously or within a specific time frame are not postulated to occur with the exception of the Hi-Low pressure interface. The evaluation of potential fire induced circuit failures is consistent with the guidance provided in Generic Letter 86-10 which identifies the specific case where analysis of multiple failures is required.

Question

5. The IPEEE submittal notes that a generic event tree was developed to represent the potential shutdown systems available and was used as a template for individual fire areas. The event trees were then modified to specifically model each unique set of systems categorized as successful and failed for each particular fire compartment. Provide a copy of the event tree (including definitions of all event tree top events), a listing of the conditional probability of all events in the tree, and a discussion of the bases for the quantification values used.

   Explain how initiating events other than an automatic or manual reactor trip (e.g., fire-induced loss of offsite power) were considered, including specifically how they were modeled.

Response

5. A copy of the event tree(s) is provided. The event tree top events and the conditional probabilities are represented directly on the tree(s). The bases for the quantification values used are those typically described in the LGS IPE submittal. The one exception to this is the quantification of the vent path failure. Only one vent path was considered in the fire analysis because of the number of cables that would need to be identified and tracked from a fire perspective if the multiple vent paths were modeled similar to the IPE.

Question

6. The submittal states (Section 4.1.2), "Transient ignition sources were identified by calculating a generic number (see Section 4.4.1.2) which was used for all fire compartments at Limerick." This methodology is not consistent with the "FIVE" computer code, and is also not considered to be an acceptable Probabilistic Risk Assessment (PRA) practice. The generic number used in such an analysis must be shown to bound the probability of transient combustible fires in each compartment throughout the plant. Provide either a FIVE-consistent analysis or demonstrate that the generic number used in the IPEEE is bounding.

Response

6. The generic transient ignition source factor was calculated following the information in the FIVE Methodology. The generic number was calculated to bound all possible transient ignition sources in all plant compartments.

   Per plant Administrative Controls, no cigarette smoking or use of candles is permitted in plant structures; therefore, they are not included in the transient calculation. The remaining transients, Extension Cord, Heater, Overheating, and Hot Pipe were assumed to be allowed in all areas. Using this information, the "generic" transient ignition source factor was calculated using the following equation taken from the FIVE Methodology Section 6.3.1.2.

$$\text{Transient Factor} = \frac{\text{Total of factors allowed in compartment}}{\text{number of compartments}}$$

Total of Factors:

Extension Cord	4
Heater	3
Overheating	2
Hot Pipe	1
	10

$$\text{Transient Factor} = \frac{10}{\text{(number of fire compartments)}}$$

$$= \frac{10}{127}$$

$$= 7.87\ E\text{-}2$$

Question

7. Provide the results of the walkdowns. In addition, address how the walkdowns ensured that cable routing information used in the fire IPEEE represents as-built information, and how the walkdowns evaluated possible dependence between the remote shutdown and control room circuitry (as provided for in NUREG-1407, Appendix C, Section C.3).

Response

7. As outlined in Section 4.2.4 of the submittal, the following walkdowns were performed to confirm information taken from plant drawings and documentation:

- Fire Ignition Source
- Fire Source
- Fire Compartment Boundary
- CFZ Boundary

- Sphere of Damage (SOD)
- Combustible Review
- Sensitive Electrical Equipment

Cable/raceway location and fire/seismic interactions walkdowns were also performed.

These walkdowns were performed by qualified individuals and were performed in accordance with the project walkdown procedure. General results of the walkdowns are as follows.

Fire Ignition Source - These walkdowns were used to confirm the number of ignition sources in each plant compartment which was taken from PIMS and plant drawings. The walkdowns confirmed the majority of information. Some compartments required revision to the number of electrical cabinet compartments; however, revised F_1 numbers did not vary significantly from the original calculated values.

Fire Source Locations and Quantities - These walkdowns were used to confirm the location of each fire source and the relative amount of combustibles, and to provide characteristics about the fire source. All equipment locations were as shown on the plant layout drawings. Relative quantities of combustibles were in agreement with plant data. Characteristics gathered included information on:

- Cabinet size and height from floor level
- Existence of cabinet vents and equipment locations
- EQ status of equipment
- Number of compartments within electrical cabinets

This information was used during the fire modeling of significant compartments.

Fire Compartment Boundary Verification - This walkdown was conducted to confirm the requirements for barriers as stated in the FIVE Methodology Section 5.3.6, "Perform Fire Compartment Interaction Analysis." As a result of this walkdown, several fire areas which had been subdivided into compartments were recombined due to pathways between the compartments.

CFZ Boundary Verification - This walkdown was performed to confirm the CFZ boundaries were properly identified in the plant. Results of the walkdown show that all CFZ boundaries were identified in the plant per the design basis documentation.

SOD Walkdowns - SOD walkdowns were conducted to verify intervening combustibles and targets within each calculated SOD. The results indicated that information on the location of intervening combustibles was accurate. Some target information required revision for proper location for fire modeling calculations.

Transient and Fixed Combustible Review - These walkdowns gathered the information on the location and quantity of fixed combustibles and expected transient combustibles. These walkdowns were performed by member(s) of the plant fire protection staff. This information was used as the basis for fire modeling in the compartments. Information was based on actual plant conditions, historical records, and the experience of the plant fire protection personnel.

Sensitive Electrical Equipment Verification - To support the evaluation of sensitive electrical equipment, walkdowns were performed to locate the equipment with respect to the SODs. Walkdowns were performed for all critical compartments. The equipment was then evaluated for damage using the fire modeling techniques outlined within the FIVE Methodology.

Cable/Raceway Location - Section 4.2.1 discusses PECO Energy's management of cable location data and Section 4.2.2 discusses Control Room/Remote Shutdown Circuit Dependencies. To ensure adequate separation between the Control Room and the Remote Shutdown Room, raceway/cable locations were 100 percent verified by walkdown. No anomalies were identified. For the remaining plant areas, when required, raceway/cable locations identified by drawing review were verified by walkdown prior to updating the cable management system.

Fire/Seismic Interactions - To address the issue of fire/seismic interaction, proceduralized walkdowns were performed. The issue of fire/seismic interactions including the results of the walkdowns are in Section 4.8.2.1.

Question

8. The study assumes that passive fire-barrier elements (e.g., walls, floors, ceilings, and penetration seals) are 100% reliable. Such an analysis is not valid unless the assumption is adequately justified and it can be demonstrated that there are no paths through the barrier for the spread of damage. Provide such justification and demonstration for high-hazard fire areas, such as: the turbine building, diesel generator rooms, cable spreading rooms, switchgear rooms, and lube oil storage areas.

Response

8. Section 5.2.1 of the FIVE Methodology states:

> "The Phase I Screen takes credit for fire area boundaries (see Definitions 2.1 and 2.2) being effective in controlling a fire from spreading to the other side of a fire barrier. This is based on an assumption that the plant can demonstrate that the fire barriers and their components (i.e., fire doors, fire dampers, and fire penetration seal assemblies) are being inspected and maintained on a regular basis in accordance with established plant surveillance procedures and that appropriate compensatory measures are being taken when discrepancies in the barriers are found. This plant fire barrier surveillance program should be able to satisfy the intent of the guidelines in Item II of the Sandia Fire Risk Scoping Study Evaluation (Attachment 10.5).
>
> Fire barriers reviewed as part of the plant's Appendix R Safe Shutdown Analysis are assumed to be designed and installed correctly in accordance with good fire protection engineering practice and nationally recognized fire protection standards."

Fire compartments used in the analysis were identified by first dividing the plant into fire areas as defined by the LGS fire safe shutdown analysis. These fire areas are bounded by rated fire barriers as defined in the fire safe shutdown analysis and the FIVE Methodology. These fire areas were then subdivided as applicable into fire compartments using the methodology outlined in the FIVE procedure Section 5.3.6. Compartment barriers that were screened using the Fire Compartment Interaction Analysis (FCIA) had walkdowns completed by two qualified fire protection engineers to verify the ability of the barrier(s) to withstand the expected fire and prevent fire spread to the adjacent compartment(s).

Fire barrier qualifications as addressed in the Sandia Fire Risk Scoping Study Item II are discussed in Section 4.8.2.2 of the submittal. The fire barrier control and inspection program at LGS in the Technical Requirements Manual was evaluated as being adequate to assure the proper design basis and function of fire barriers and any active components (fire dampers, fire doors) within these barriers. The fire barrier inspection and control program includes those areas ascribed in the question.

Question

9. The fire compartment interaction analysis (FCIA) is based on the assumption that fire barriers are effective as rated. For active fire barriers (e.g., a normally open fire door that gets closed by fusible link), the failure probability can be significantly high. Provide a list of compartments with active fire barriers, a description of the active barriers, and a discussion regarding qualitative screening of these (and their adjacent) compartments.

Response

9. See response to Question F8 above. Additionally, it is believed that a detailed listing of fire compartments with active components or additional discussion of their screening (other than that provided in the submittal) would not be beneficial based on the above discussion.

Question

10. It is not considered technically justifiable that open hatchways in the reactor building are capable of containing hot gas and smoke spread. Provide an analysis of the effect on fire area multi-zone screening of considering the potential for hot gas and smoke spread.

Response

10. The Units 1 and 2 reactor buildings at LGS are provided with an equipment hatchway which extends from the grade elevation (217') to elevation 332'. The hatchway is provided with a suppression system at each elevation which is designed to prevent the spread of smoke and hot gases from one elevation to another. For the purposes of the LGS fire risk analysis, an evaluation was performed to analyze the effects of a fire on one elevation to the elevation(s) above. The results of these evaluations showed that with the expected fixed and transient combustibles in each reactor building, a fire in one elevation would have no effect on cabling and equipment on upper elevations due to the cooling of the fire plume as it rose up the hatchway. This evaluation did not credit the cooling effect of the hatchway sprinkler systems.

Due to the results of these evaluations, it was assumed in the reactor building analyses that the smoke and hot gases from a fire would be contained to the elevation of fire origin. This assumption provides a conservative approach as it causes the fire plume to remain on the elevation of origin, thereby producing elevated plume and hot gas layer temperatures for the area under consideration. This is consistent with the guidance provided in the FIVE Methodology.

Question

11. Provide the details concerning the screening analyses of the following fire compartments (including the relative separation between potential combustible sources and critical equipment, as well as whether or not any non-IEEE 383 rated cabling is utilized):

- Fire Compartment 1E - Recombiner Access Area
- Fire Compartment 07 - 4kV Switchgear Corridor
- Fire Compartment 22 - Unit 1 Cable Spreading Room
- Fire Compartment 23 - Unit 2 Cable Spreading Room
- Fire Compartment 44 - Unit 1 Safeguard System Access Area
- Fire Compartment 45 - Unit 1 Control Rod Drive (CRD) Hydraulic Equipment Area
- Fire Compartment 47 - Unit 1 Isolation Valve Compartment Areas
- Fire Compartment 64 - Unit 2 Reactor Enclosure Cooling Water Equipment Area
- Fire Compartment 67 - Unit 2 Safeguard System Access Area
- Fire Compartment 68 - Unit 2 CRD Hydraulic Equipment Area
- Fire Compartment 70 - Unit 2 Isolation Valvem Compartment Area
- Fire Compartment 87 A/B/C - Condensate Pump Rooms, Generator Equipment Areas, Operating Floor

Response

11. All 12 of the compartments were first analyzed using the qualitative (Phase 1) screening process as outlined in the FIVE Methodology Section 5.0. All of the compartments are fire safe shutdown analysis fire areas bounded by rated barriers except for Compartment 1E which was separated from other compartments in fire area 1 by following the compartment interaction analysis as outlined in the FIVE Methodology Section 5.3.6. Since all twelve compartments did not meet the screening criteria of the Phase 1 analysis (F_2 < 1E-6), they were subjected to the quantitative (Phase 2) screening process as outlined in Section 6.0 of the FIVE Methodology. Details of this screening process are described in the following paragraphs and summarized in Table F11.1 below.

Compartment 1E

The compartment was analyzed for the probability of fixed and transient combustibles exposure damage. The evaluation of fixed combustible loading determined that the only combustible present in the compartment was electrical cable insulation, which due to plant installation requirements is IEEE 383 rated. The methodology states that "Self-ignited cable fires for IEEE 383 qualified cable will not be considered fire source initiations in this methodology, consistent with past PRA methods." In accordance with the assumption in the methodology, the cable is not considered an ignition source. No other material or equipment exists within the compartment.

The transient combustible evaluation showed that the "typical" transient combustible that was used could damage redundant equipment in the area. Therefore, administrative controls could be established for the compartment which restricted transient combustibles from the area unless they are constantly attended. As stated in the FIVE Methodology, transient combustibles controlled in this manner do not need to be considered "exposed" combustibles. Therefore, no exposure due to transient combustibles is expected in the compartment.

Due to the lack of possible exposure from both fixed and transient sources, no fire damage to critical equipment is postulated for this compartment. Following the guidance of the FIVE Methodology with P_f = 0 and P_{tc} = 0, the compartment screens as F_3 < 1E-6.

Compartment 07

This compartment was analyzed following the same methodology as Fire Compartment 1E.

Compartment 22

This compartment was analyzed following the same methodology as Fire Compartment 1E.

Compartment 23

This compartment was analyzed following the same methodology as Fire Compartment 1E.

Compartment 44

Fire Compartment 44 consists of the Unit 1 Safeguard System Access Area on the 217' elevation of the reactor building. To meet the intent of 10CFR50 Appendix R, Section III.G.2.b, which is equivalent to CMEB BTP9.5-1 Section C.5.b(2), to which Limerick is committed, a 20 ft. separation zone is established in the area to separate redundant safe shutdown methods. Although this separation zone is established, it was not initially credited in the IPEEE analysis, and thus served as a bounding analysis verifying the need for the separation zone.

Following the guidance provided in the FIVE Methodology, the probability of damage to targets (required safe shutdown equipment) was performed for both fixed and transient combustibles. Fixed combustibles consisted of electrical cabinets. An evaluation of the intervening combustibles for all fixed sources was performed, and the heat release rate (HRR) from these was added to the fixed source HRR. Each target was analyzed using the calculation sheets within the methodology. Due to the size of the fixed fire sources in the compartment and the relative positions to the targets, no damage from the fixed fire sources was expected; therefore, the probability of fixed combustible damage (P_f) was assigned a value of 0.0.

As described in Section 4.3.2.3 of the submittal, a bounding transient combustible was analyzed for each critical compartment in the plant. The probability of damage due to this transient was analyzed using the calculation sheets within the FIVE Methodology. Transient combustible damage was precluded by administrative controls on spatial separation and the response of the site fire brigade; therefore, the probability of fire suppression unavailability from transient combustible exposure (P_{fst}) was assigned a value of 0.1. Other transient combustible considerations were calculated as follows:

w: w is defined as the probability of having critical amounts of transient combustibles between periodic inspections. Due to the administrative controls on combustibles and to allow conservatism within the calculation, a value of 1.0 was assumed.

u: u is defined as the probability of the transient combustible being located where it can cause damage, and is a ratio of the target areas to net floor area. This value was calculated as outlined in the FIVE Methodology Section 6.3.7.2, Step 3.5, to be 0.159.

p: p is defined as the probability of having the transient combustible exposed. Due to the administrative controls on the use and storage of combustibles in the plant, p was assigned a value of 0.1 as defined in the FIVE Methodology, Section 6.3.7.2, Step 3.6.

Using these values, the probability of transient combustibles exposure damage (P_{tc}) was calculated by:

$$P_{tc} = P_{fst} \cdot u \cdot p \cdot w$$

$$= 0.1(0.159)(0.1)(1.0)$$

$$= 1.59E\text{-}3.$$

The probability of damage due to fixed and transient combustibles exposure (P_3) was calculated by:

$$P_3 = P_f + P_{tc}$$

$$= 0.0 + 1.59E\text{-}3$$

$$= 1.59E\text{-}3.$$

The overall frequency of a fire occurring and damaging safe shutdown components in the compartment (F_3) was calculated by:

$$F_3 = P_3 \cdot F_2$$

$$= 1.59E\text{-}3(6.9E\text{-}3)$$

$$= 1.1E\text{-}5.$$

This is the value as stated in Section 4.4.1.3 of the submittal. Because this value was above the screening criteria of 1.0E-6, an analysis on the availability of safe shutdown equipment was performed.

Subsequent to the calculation of F_3, this compartment was separated into its respective Appendix R East/West compartment designations by confirming the adequacy of the separation zone consistent with the FIVE Methodology. Given this confirmation, Compartment 44E and Compartment 44W were re-analyzed for fire-induced equipment failures.

The re-analysis involved the identification of the cables and components (targets) with respect to the known sources of fixed and potential transient combustibles. It was performed to confirm that credit could be given to specific equipment within this compartment. When it was determined that a system or train would survive, the PSA models were used to calculate the unavailability of the surviving systems from non-fire-induced causes and again compared to the 1.0E-6/yr. screening criteria. Because the calculated P_4 includes credit for systems outside the compartment, the final probability associated with a fire in this compartment is F_3 divided by P_2 and then multiplied by P_4.

Compartment 45

Fire compartment 45 consists of the Unit 1 CRD Hydraulic Equipment Area on the 253' elevation of the reactor building. To meet the separation intent of 10CFR50, Appendix R, Section III.G.2.b, which is equivalent to CMEB BTP9.5-1 Section C.5.b(2), to which Limerick is committed, a 20 ft. separation zone is established in the area to separate the redundant safe shutdown methods.

The compartment was analyzed following the same methodology as Fire Compartment 44.

Compartment 47

Fire Compartment 47 consists of the Unit 1 Isolation Valve Access area on the 283' elevation of the reactor building. To meet the separation intent of 10CFR50, Appendix R, Section III.G.2.b, which is equivalent to CMEB BTP9.5-1 Section C.5.b(2), to which Limerick is committed, a 20 ft. separation zone is established in the area to separate the redundant safe shutdown methods.

The compartment was analyzed following the same methodology as Fire Compartment 44.

Compartment 64

Fire Compartment 64 consists of the Unit 2 RECW Equipment Area on the 201' elevation of the reactor building. The area was analyzed following the same methodology as Fire Compartment 44.

Compartment 67

Fire Compartment 67 consists of the Unit 2 Safeguard Systems Access Area on the 217' elevation of the reactor building. To meet the separation intent of 10CFR50, Appendix R, Section III.G.2.b, which is equivalent to CMEB BTP9.5-1 Section C.5.b(2), to which Limerick is committed, a 20 ft. separation zone is established in the area to separate the redundant safe shutdown methods.

The compartment was analyzed following the same methodology as Fire Compartment 44.

Compartment 68

Fire Compartment 68 consists of the Unit 2 CRD Hydraulic Equipment Area on the 253' elevation of the reactor building. To meet the separation intent of 10CFR50, Appendix R, Section III.G.2.b, which is equivalent to CMEB BTP9.5-1 Section C.5.b(2), to which Limerick is committed, a 20 ft. separation zone is established in the area to separate the redundant safe shutdown methods.

The compartment was analyzed following the same methodology as Fire Compartment 44.

Compartment 70

Fire Compartment 70 consists of the Unit 2 Isolation Valve Access Area on the 283' elevation of the reactor building. To meet the separation intent of 10CFR50, Appendix R, Section III.G.2.b, which is equivalent to CMEB BTP9.5-1 Section C.5.b(2), to which Limerick is committed, a 20 ft. separation zone is established in the area to separate the redundant safe shutdown methods.

The compartment was analyzed following the same methodology as Fire Compartment 44.

Compartment 87

Fire Compartment 87 includes the condensate pump rooms, generator equipment areas, and operating floor of the turbine building. This fire compartment consisted of fire areas 87, 100, 113, and 114 due to the inability to separate the areas by following the Fire Compartment Interaction Analysis (FCIA). Although the barriers separating the areas were fire rated, the structural steel supporting the ceiling assembly in the condensate pump rooms was not protected and had not been evaluated per the NRC approved LGS structural steel analysis as being acceptable.

As part of the LGS fire risk analysis, a structural steel analysis of the ceiling assemblies was performed. The results of the analysis showed that the structural steel could withstand the expected fire hazard; therefore, the barriers were capable of providing separation of the areas. Following the FCIA methodology, the areas were compartmentalized into Compartments 87A (Fire Area 87), Compartment 87B (Fire Area 100), and Compartment 87C (Fire Areas 113 and 114). These three compartments screened in the Phase 1 analysis due to F_2 < 1E-6.

TABLE F11.1

FIRE COMPARTMENT	F_2	P_f	U	P_{tc}	P_3	F_3
1E	7.7E-4	0.0	---	0.0	---	0.0
7	7.7E-4	0.0	---	0.0	---	0.0
22	4.0E-3	0.0	---	0.0	---	0.0
23	7.7E-4	0.0	---	0.0	---	0.0
44	6.9E-3	0.0	0.159	1.59E-3	1.59E-3	1.1E-5
45	5.1E-3	0.0	0.0915	9.15E-3	9.15E-3	4.7E-6
47	6.6E-2	0.0	0.0811	8.11E-4	8.11E-4	5.3E-5
64	1.1E-4	0.0	0.0114	1.14E-4	1.14E-4	1.2E-8
67	7.1E-3	0.0	0.3737	3.74E-3	3.73E-3	2.7E-5
68	4.7E-3	0.0	0.0642	6.42E-3	6.42E-3	3.0E-6
70	6.6E-2	0.0	0.0889	8.89E-4	8.89E-4	5.9E-5
87A	9.5E-8					
87B	1.1E-7					
87C	1.0E-6					

Question

12. With regards to the analysis described in Section 4.3.3 in your submittal, have any combustible fire barrier materials been used as the basis for establishing 20-ft-separation combustible-free zones? If so, has the analysis considered propagation of fire via combustion of these fire barrier materials? If not, please provide such an assessment for fire spread.

Response

12. Thermo-Lag 330-1 fire barrier material is presently used on exposed electrical cabling in combustible-free zones. The material is used as the basis for removing the cabling as a combustible within the separation area. PECO Energy's Thermo-Lag Reduction Project is evaluating alternate methods of controlling combustibles within combustible-free zones. As a result of this effort the Thermo-Lag will be removed or fire propagation path eliminated.

    Per the information contained in NRC Information Notice 95-32, "Thermo-Lag 330-1 Flame Spread Test Results," the maximum distance of flame travel was approximately 8 ft. during the 10 minute test period. As a result of these test results, the hazards present, protection available and administrative controls which include hourly firewatches implemented as a GL 92-08 compensatory measure, it is not expected that the Thermo-Lag as installed in the combustible-free zones will affect the integrity of the combustible-free zones.

Question

13. The submittal states, "Operator effectiveness in performing manual safe shutdown actions is not considered to be affected by areas which contain smoke and hot gases." This assumption is not considered to be acceptable. Please provide a description of any sequences for which credit has been taken for operator actions in the affected fire areas. Provide an assessment of the impact on area screening if no credit is given for operator recovery actions in an affected fire area.

Response

13. Operator effectiveness in performing manual safe shutdown actions has been analyzed for the effects of smoke and hot gases. Of the manual actions credited for fires at LGS, 15 actions occur in the same fire area where the fire occurs.

Five of the 15 actions occur in the same area as the fire but are separated from the fire by a 20 ft. separation zone. As stated in Section 4.3.3 of the submittal, these zones have been evaluated as providing adequate separation to prevent smoke and hot gases from affecting equipment on the opposite side.

The ten remaining manual actions are required to be performed 2 to 3 hours into the fire/safe shutdown event. As evaluated, the plant fire brigade is credited with providing manual extinguishment of fires at 0.5 hours into the fire event. It is assumed that operator actions will be unaffected by smoke and hot gases 1.5 to 2.5 hours after fire extinguishment.

Question

14. Section 4.6.0 of the submittal states that "pre-cursor" events (such as miscalibration of sensors) from the IPE models were used to derive the fire IPEEE PRA model. It is also assumed that all systems are available at the time of fire initiation (i.e., no test and maintenance unavailabilities were included). This practice could distort or mask important risk contributors. Provide an assessment of the impact on area screening if these factors are included in the analysis.

Response

14. As stated in 4.6.0(5), pre-cursor events such as miscalibration were not used in the fire analysis. The total initiator frequency in the IPE model (the addition of all transient, LOCA, ATWS, LOOP initiators) is several orders of magnitude greater than the total of all fire initiators. Pre-cursor events, as modeled in the IPE, did not significantly contribute to the risk as calculated in the IPE. Given the previous IPE results and the initiator probabilities, the pre-cursor events would not contribute to the calculated results in the fire analysis.

A quantitative assessment of the impact of removing the test and maintenance unavailabilities was performed. The dominant contributors to the unavailability of systems not damaged by the fire were, as in the IPE, human actions. The contribution of the maintenance terms was sufficiently small that the screening of fire compartments did not change any compartment from one that screened to one that did not (i.e., migrate past the 1E-06 screening criteria).

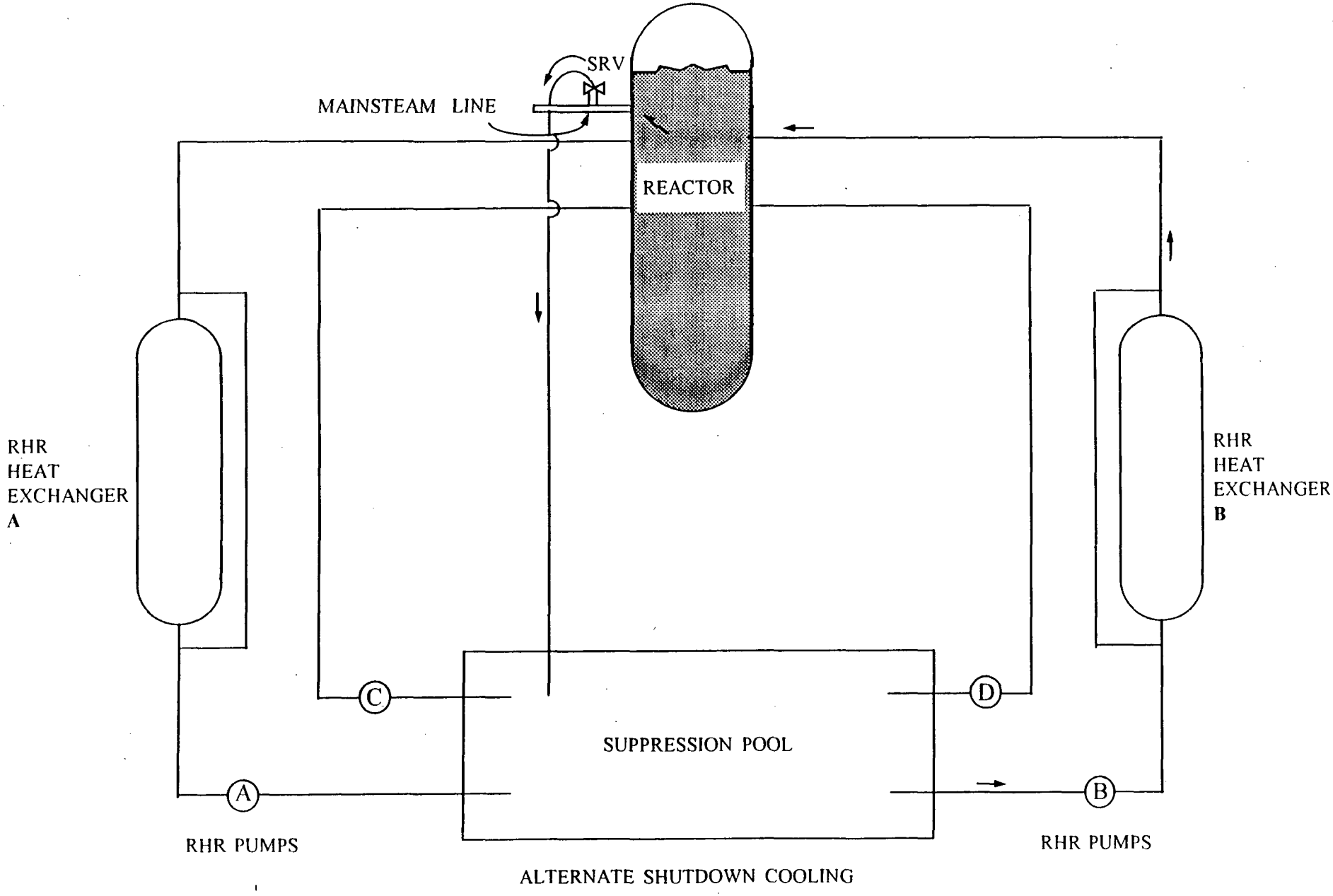
Question

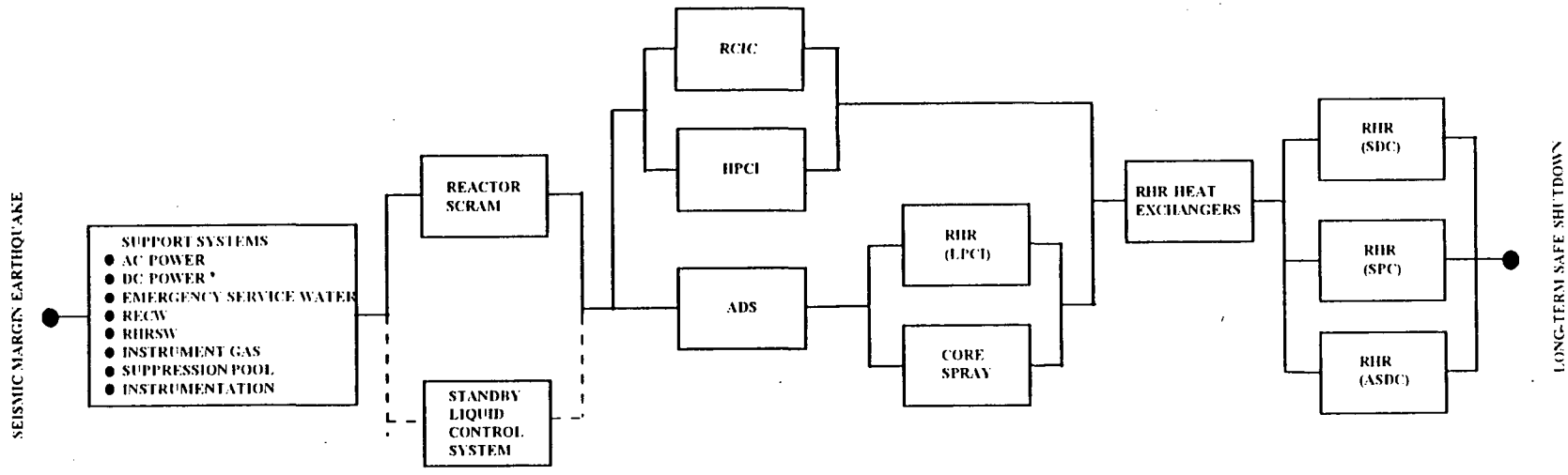
15. A listing of shared systems is not documented in the submittal. Provide a listing of shared systems (if any) and an analysis of dual-unit fire-induced core damage scenarios, including a discussion of whether or not additional fire compartments survive the screening analysis.

Response

15. The simultaneous impact of a fire in a particular compartment on both Limerick units was analyzed. Systems shared between the LGS units were reflected in the fire-induced failures and the quantification. Table 3.2.3-1 in the LGS IPE submittal indicates that portions of the Emergency Service Water (ESW) and the RHR Service Water systems are shared between the units. These systems were explicitly assessed on an individual train basis from fire impact and PSA quantification perspectives.

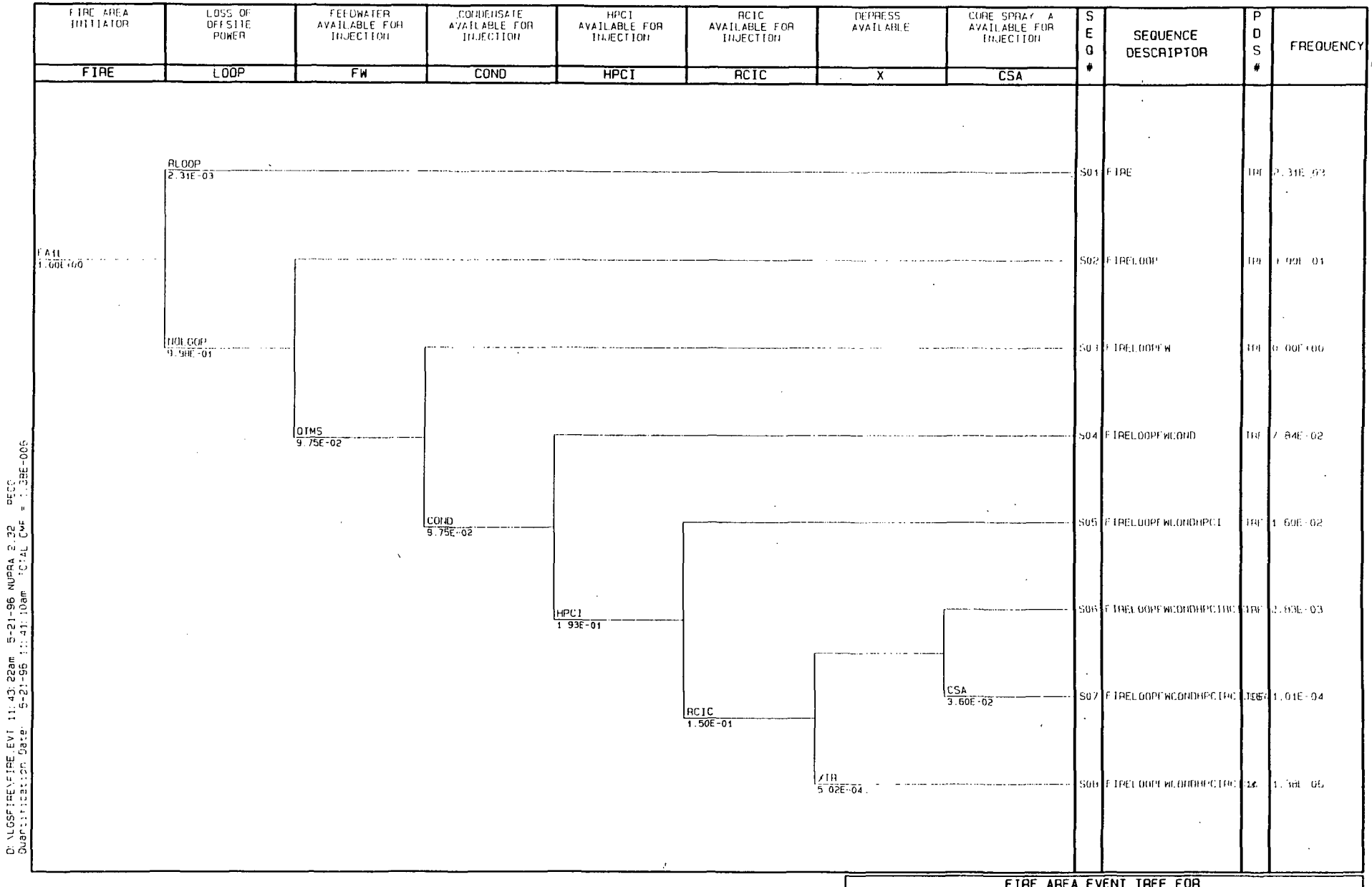
Figure S3.1


SRV
MAINSTEAM LINE
REACTOR
RHR
HEAT
EXCHANGER
A
RHR
HEAT
EXCHANGER
B
C
D
SUPPRESSION POOL
A
B
RHR PUMPS
RHR PUMPS
ALTERNATE SHUTDOWN COOLING



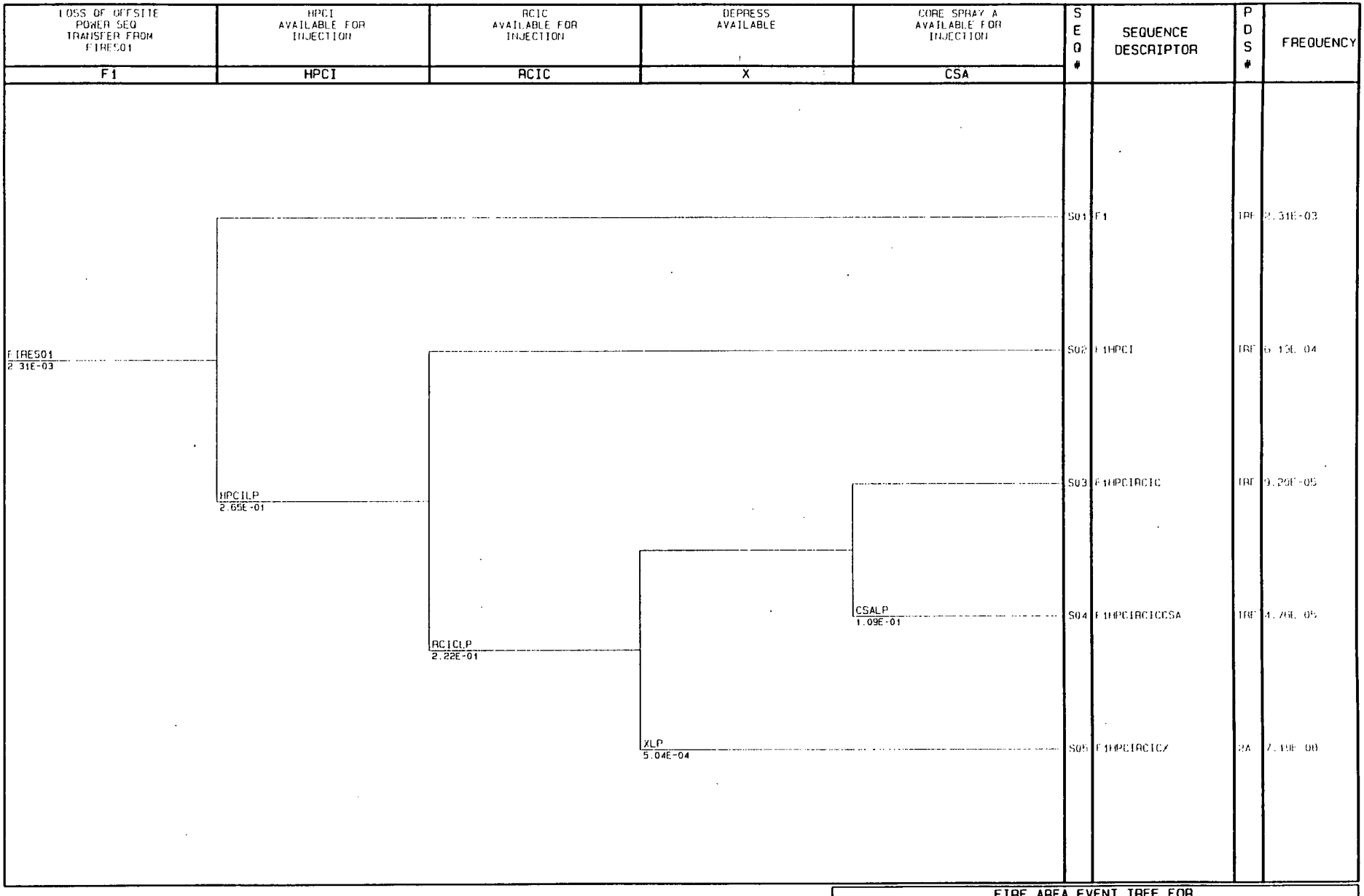
Limerick Success Path Logic Diagram

Figure S4.1



FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

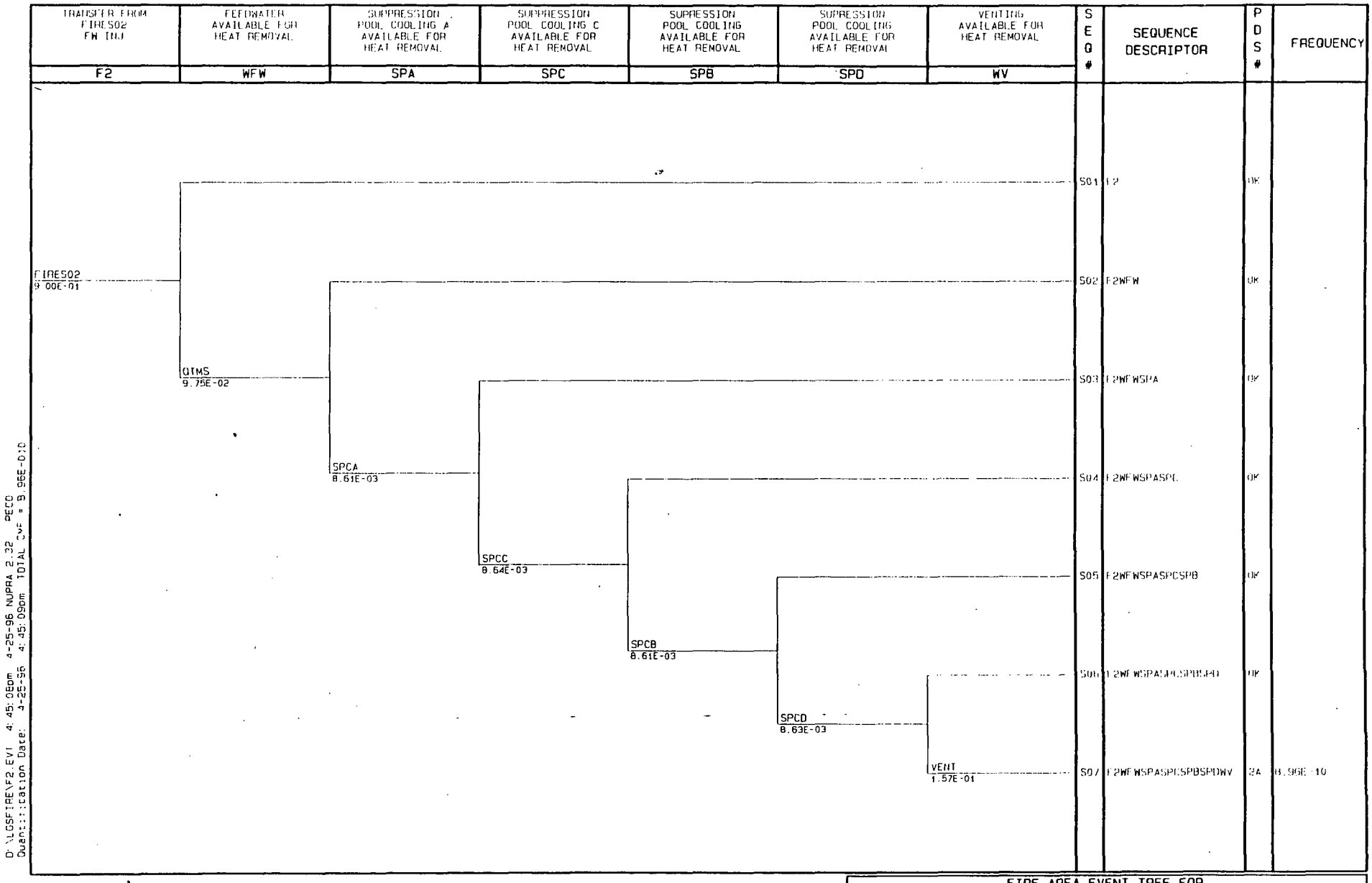
INITIAL FIRE EVENT TREE



FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

LOSS OF OFFSITE POWER SEQUENCES
TRANSFER FROM FIRES01

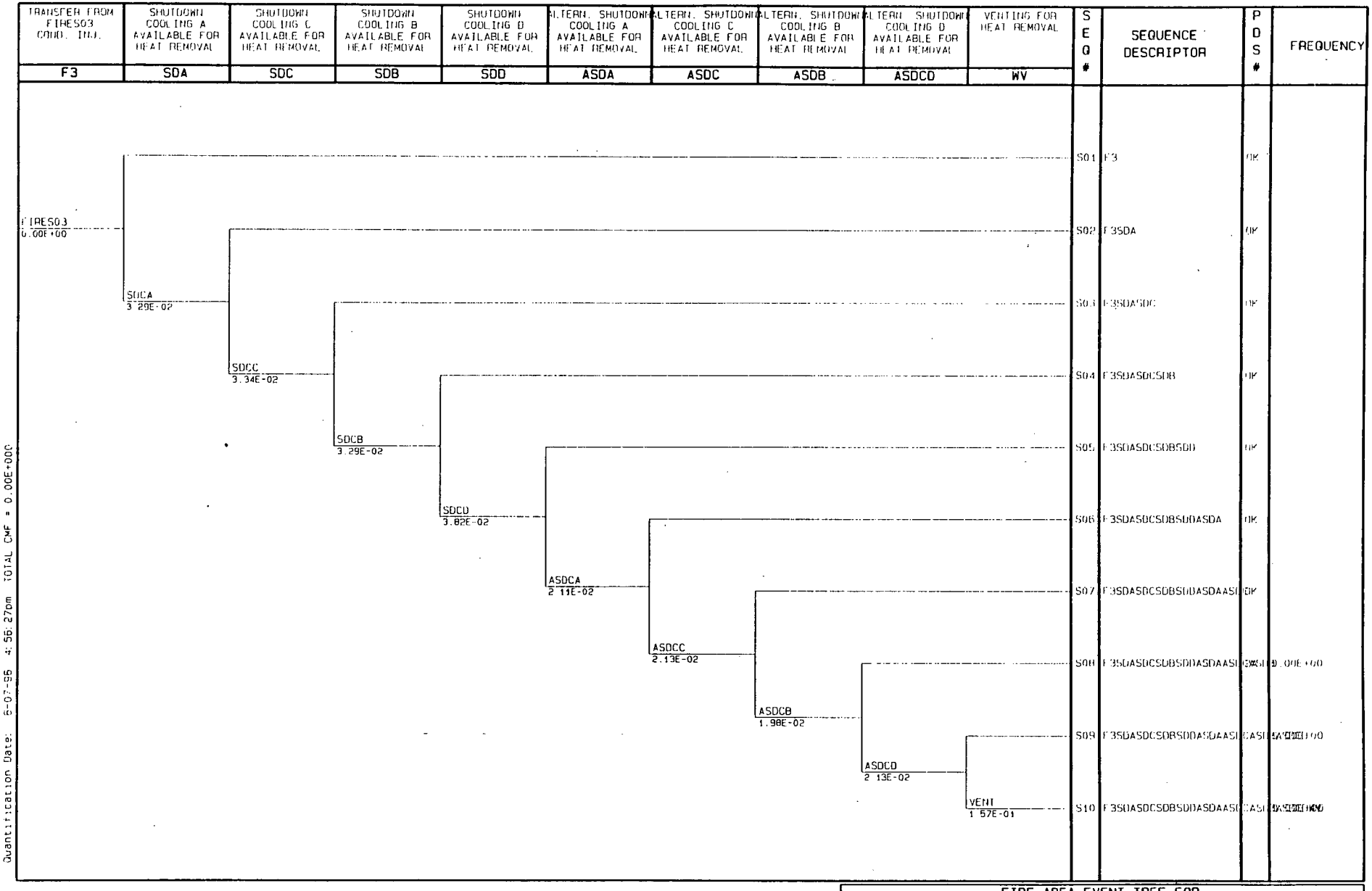
D:\LGSFIRE\F1.EVT 5:11:38pm 4-25-96 NUPRA 2.32 PECO
Quantification Date: 4-25-96 5:11:32pm TOTAL CMF = 7.19E-002



D:\LGSFIRE\F2.EV1 4:45:06pm 4-25-96 NUPRA 2.32 PECO
Quantification Date: 4-25-96 4:45:09pm TOTAL CVF = 8.96E-010

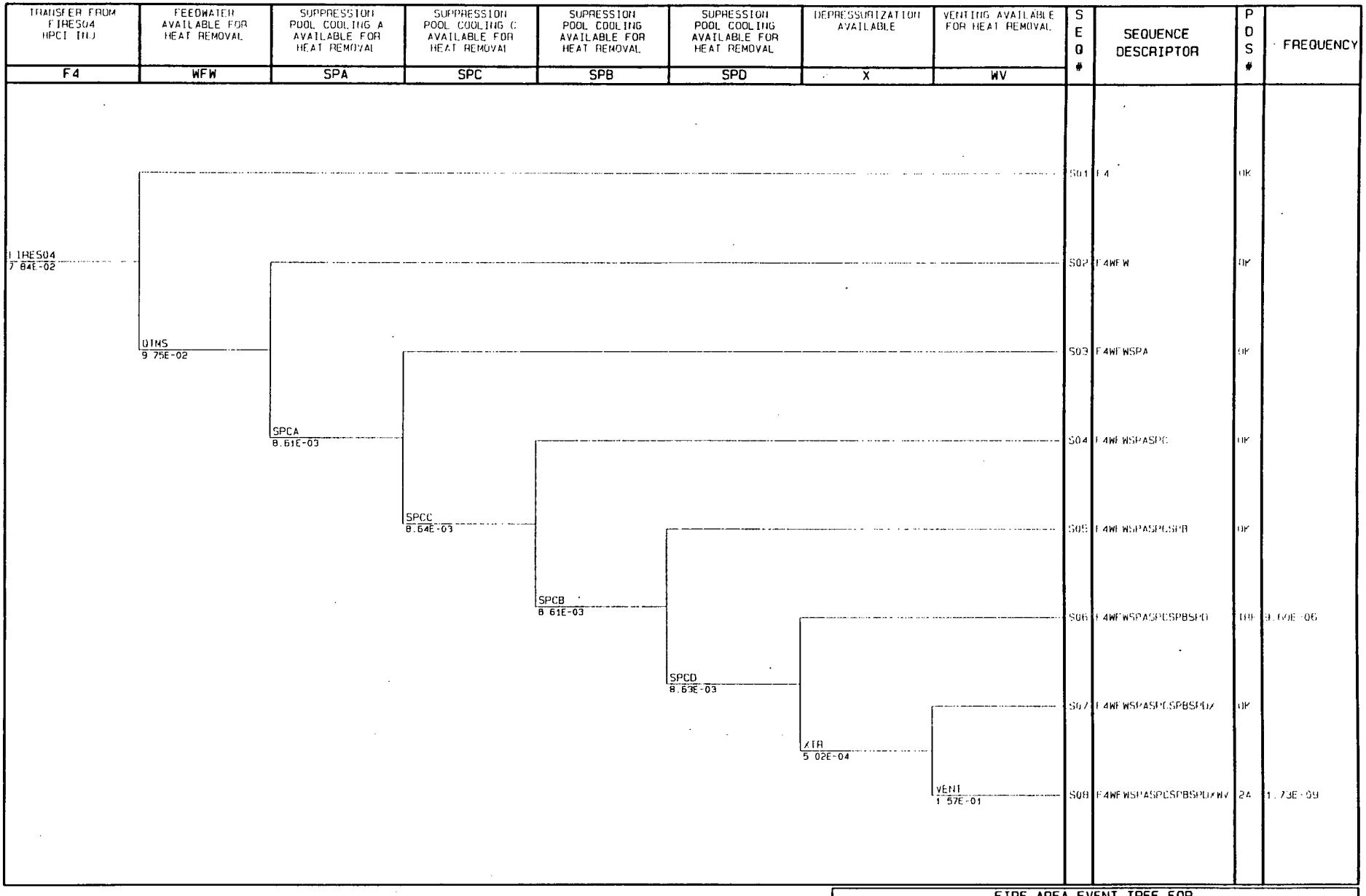
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM FIRES02
FEEDWATER INJECTION SUCESSFUL



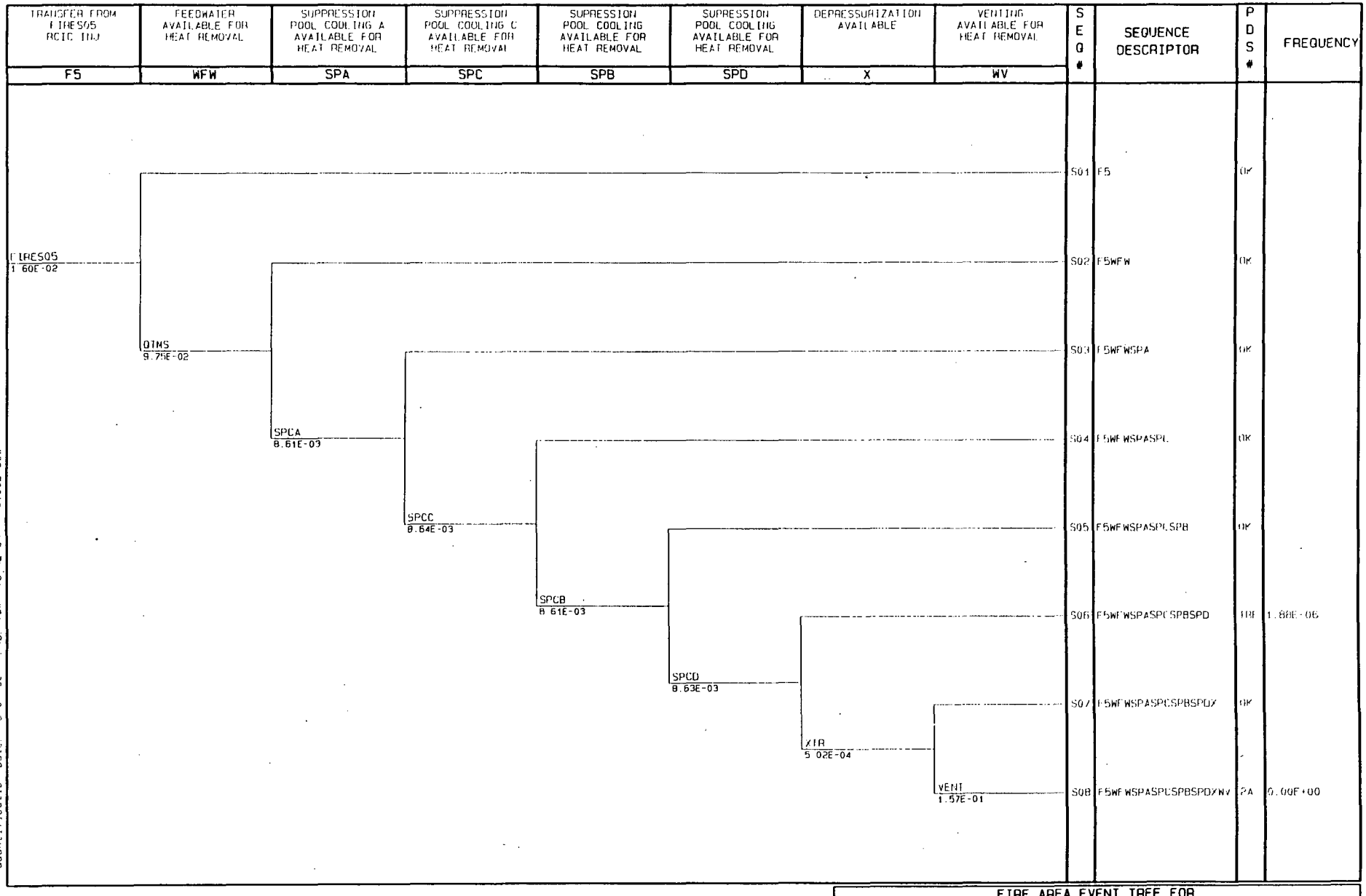
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM FIRES03
CONENSATE FOR INJECTION



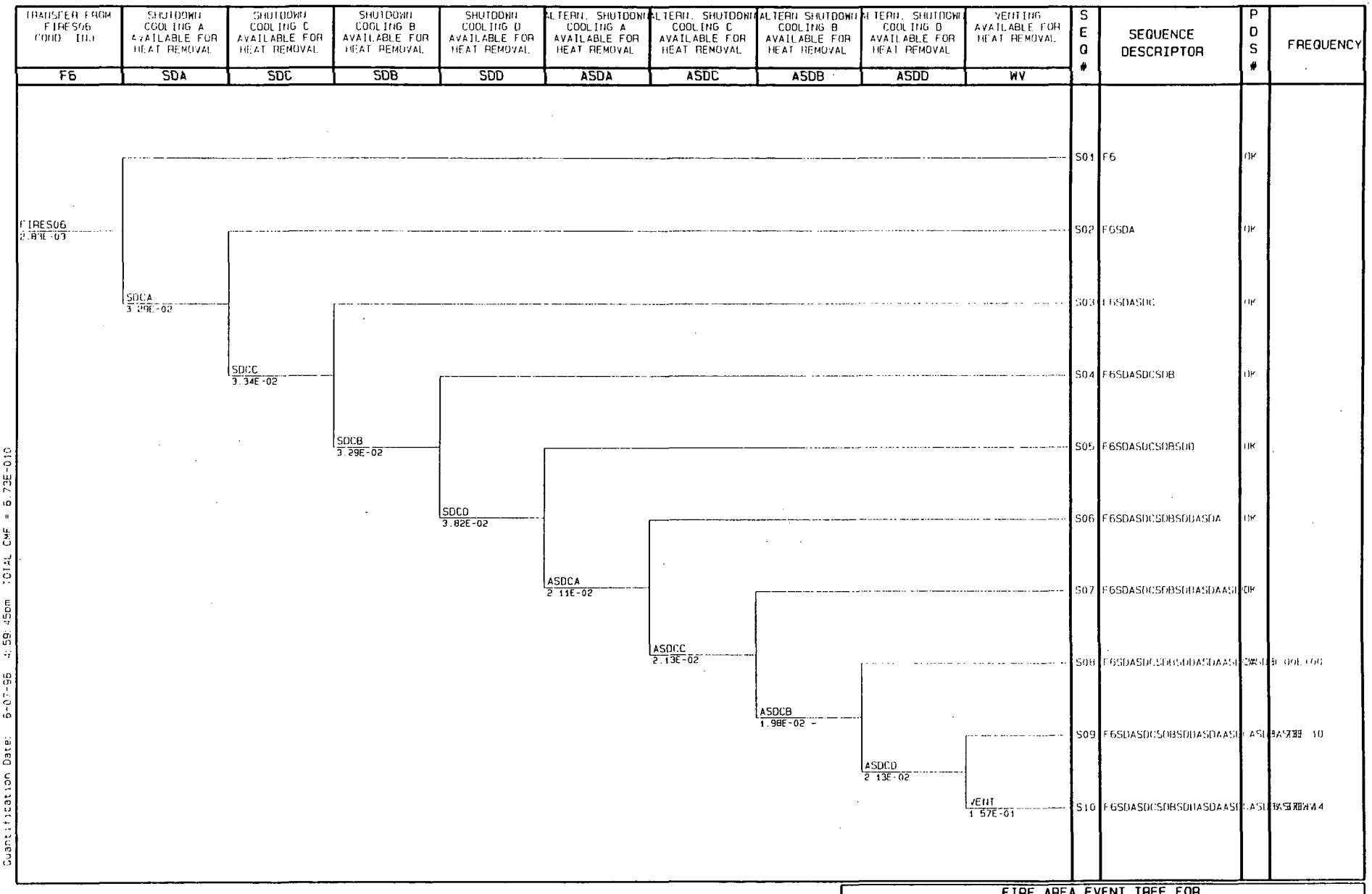
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

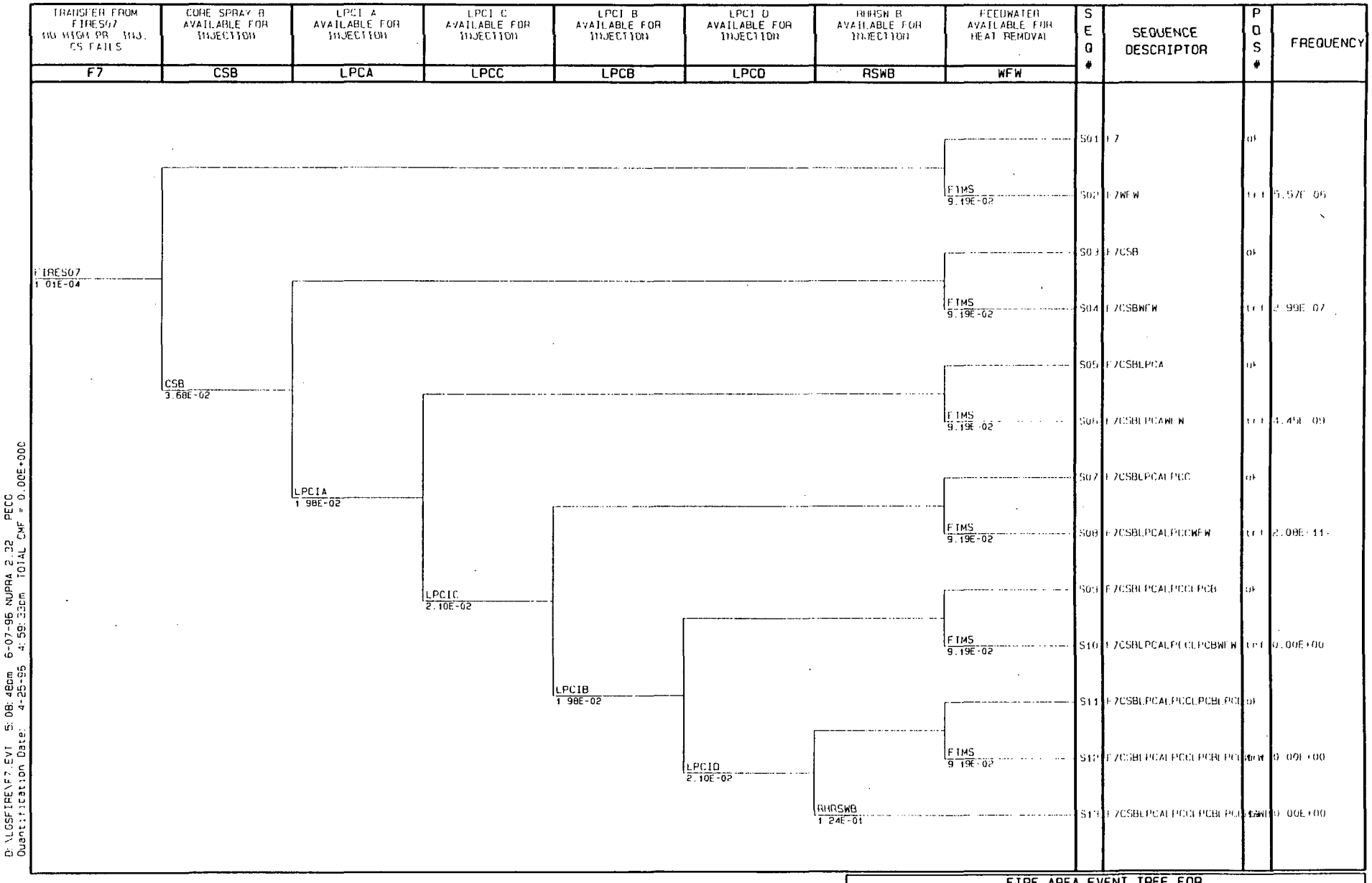
TRANSFER FROM FIRES04
FEEDWATER INJECTION SUCESSFUL



FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

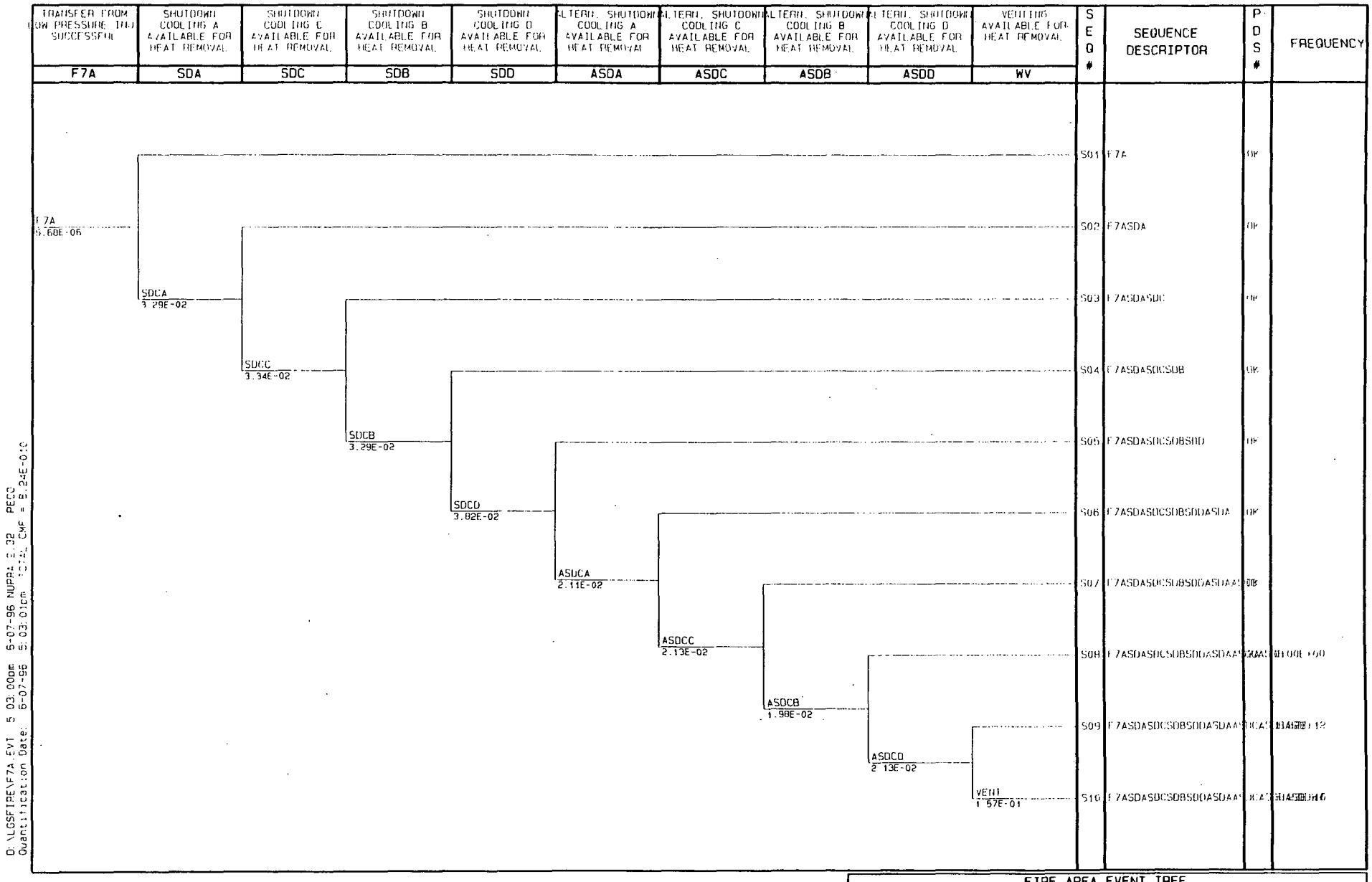
TRANSFER FROM FIRES05
FEEDWATER INJECTION SUCESSFUL


TRANSFER FROM FIRES06 COND. INJ.
SHUTDOWN COOLING A AVAILABLE FOR HEAT REMOVAL
SHUTDOWN COOLING C AVAILABLE FOR HEAT REMOVAL
SHUTDOWN COOLING B AVAILABLE FOR HEAT REMOVAL
SHUTDOWN COOLING D AVAILABLE FOR HEAT REMOVAL
ALTERN. SHUTDOWN COOLING A AVAILABLE FOR HEAT REMOVAL
ALTERN. SHUTDOWN COOLING C AVAILABLE FOR HEAT REMOVAL
ALTERN. SHUTDOWN COOLING B AVAILABLE FOR HEAT REMOVAL
ALTERN. SHUTDOWN COOLING D AVAILABLE FOR HEAT REMOVAL
VENTING AVAILABLE FOR HEAT REMOVAL
SEQ #
SEQUENCE DESCRIPTOR
PDS #
FREQUENCY
F6
SDA
SDC
SDB
SDD
ASDA
ASDC
ASDB
ASDD
WV
FIRES06
2.88E-03
SDCA
3.29E-02
SDCC
3.34E-02
SDCB
3.29E-02
SDCD
3.82E-02
ASDCA
2.11E-02
ASDCC
2.13E-02
ASDCB
1.98E-02
ASDCD
2.13E-02
VENT
1.57E-01
S01 F6 OK
S02 F6SDA OK
S03 F6SDASDC OK
S04 F6SDASDCSDB OK
S05 F6SDASDCSDBSDD OK
S06 F6SDASDCSDBSDDASDA OK
S07 F6SDASDCSDBSDDASDAASD OK
S08 F6SDASDCSDBSDDASDAASD
S09 F6SDASDCSDBSDDASDAASD
S10 F6SDASDCSDBSDDASDAASD
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION
TRANSFER FROM FIRES06
CONDENSATE FOR INJECTION



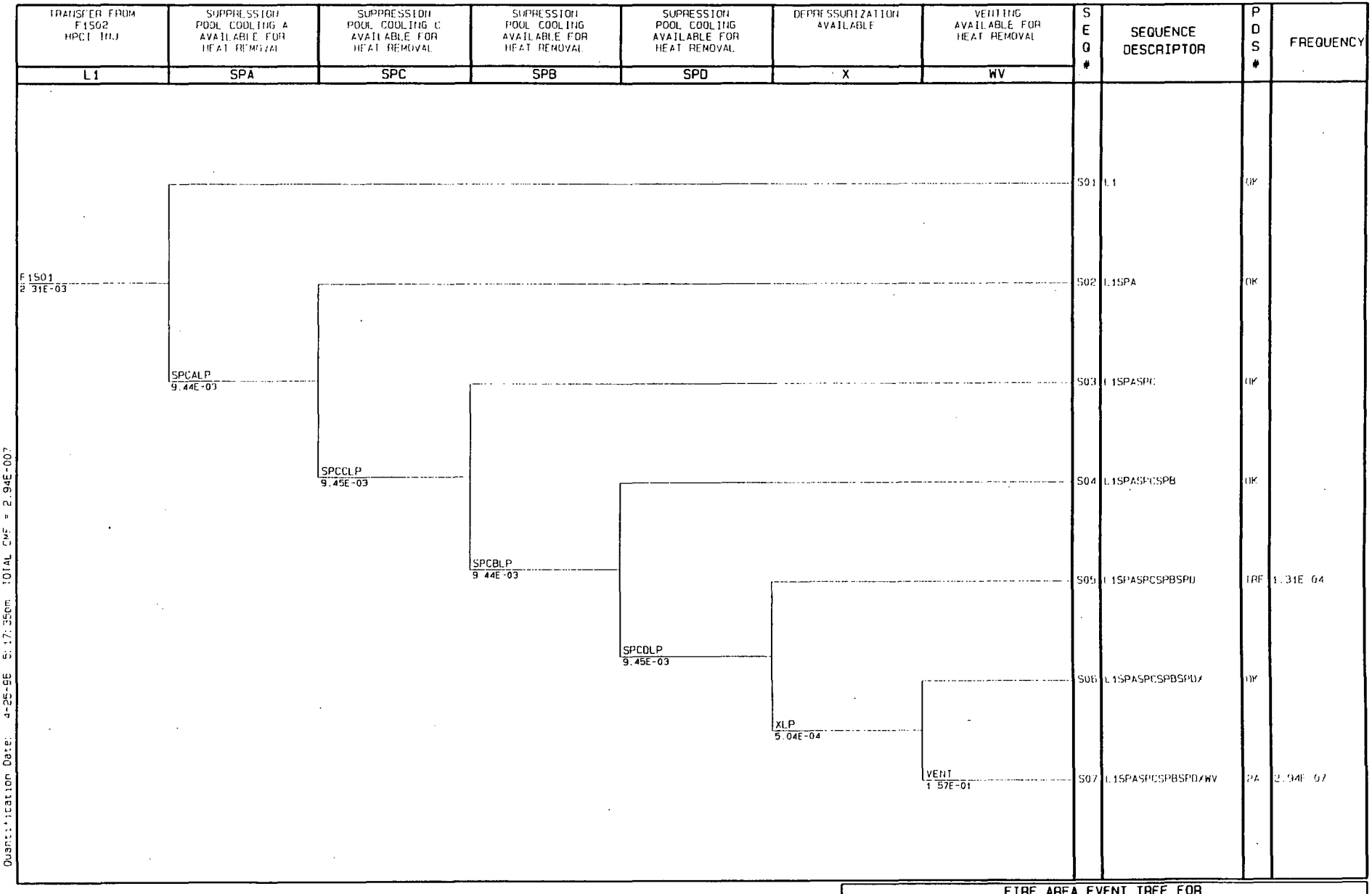
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM FIRES07
NO HIGH PR. INJ. CSA FAILS



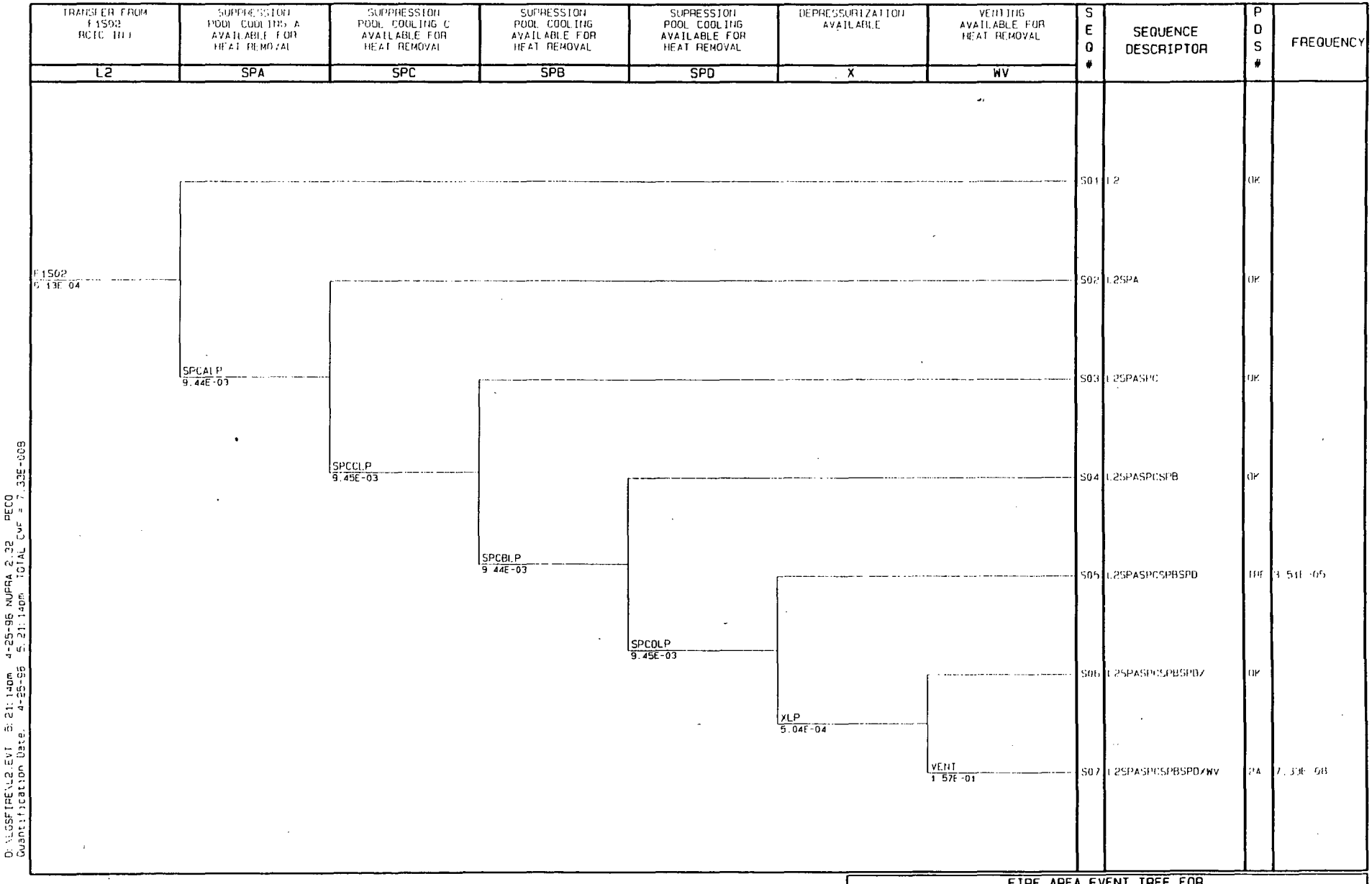
FIRE AREA EVENT TREE
LIMERICK GENERATING STATION

LOW PRESSURE INJECTION SUCCESSFUL



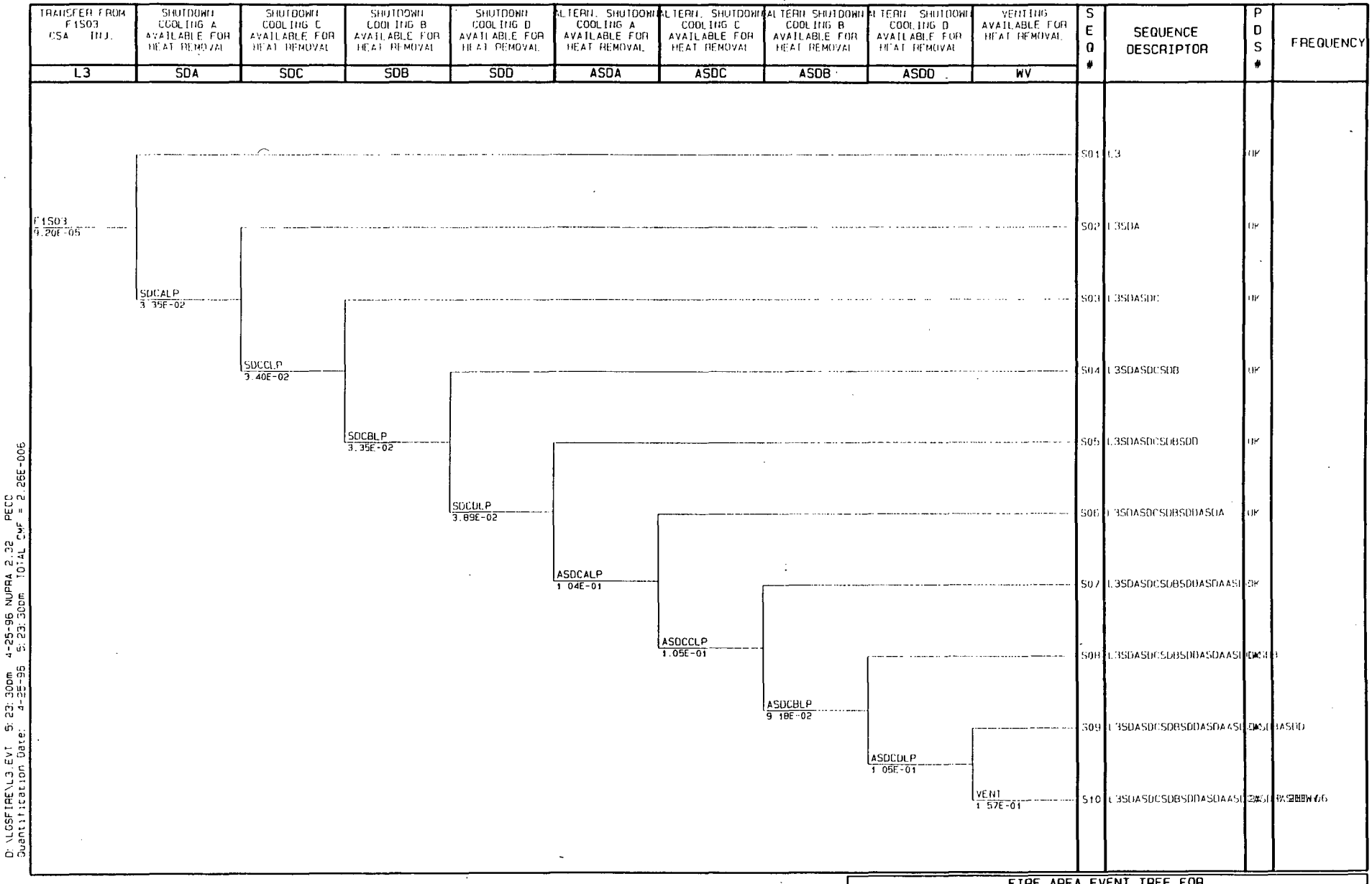
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM F1S01
HPCI INJECTION SUCCESSFUL



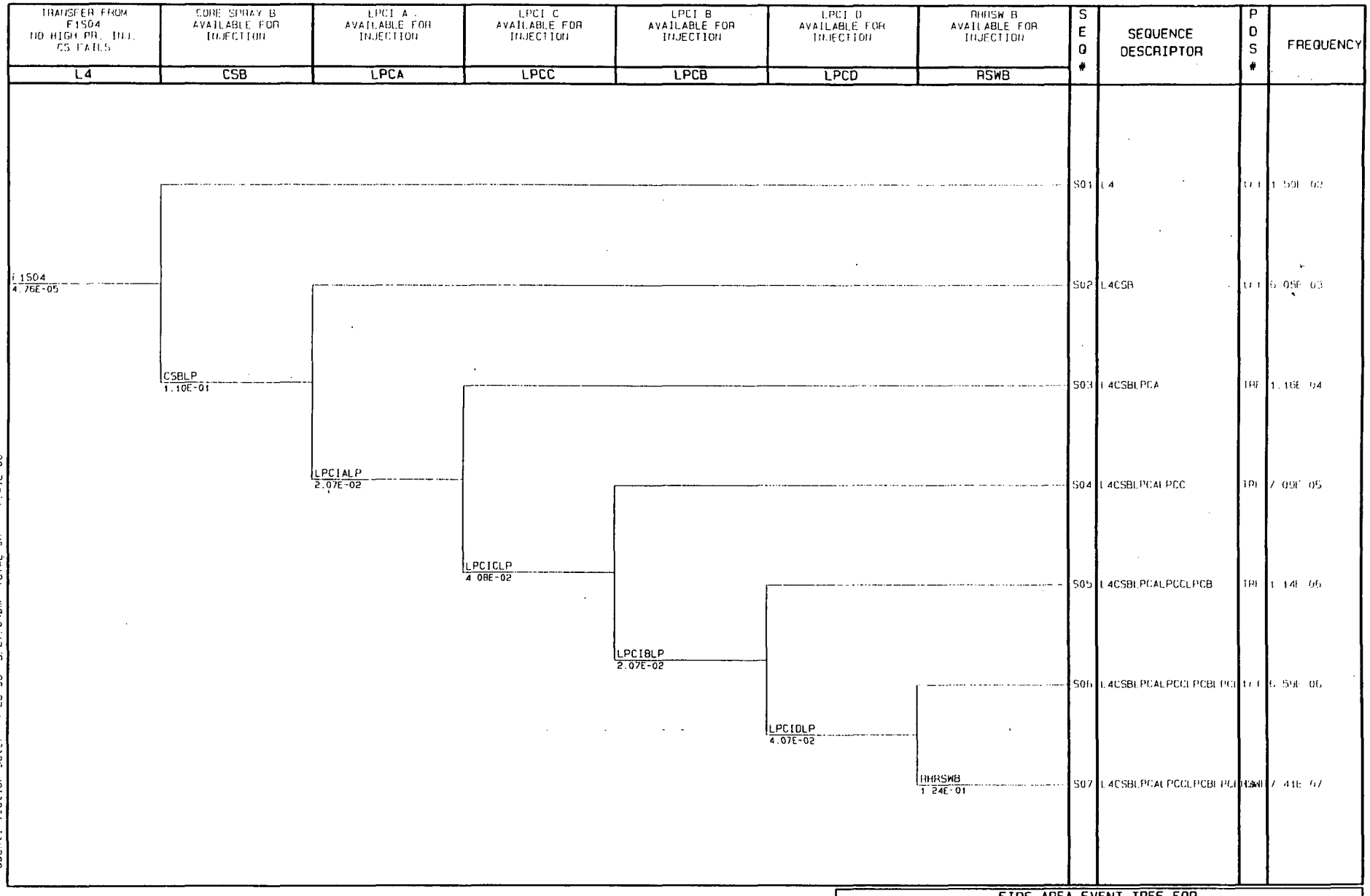
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM F1S02
RCIC INJECTION SUCCESSFUL



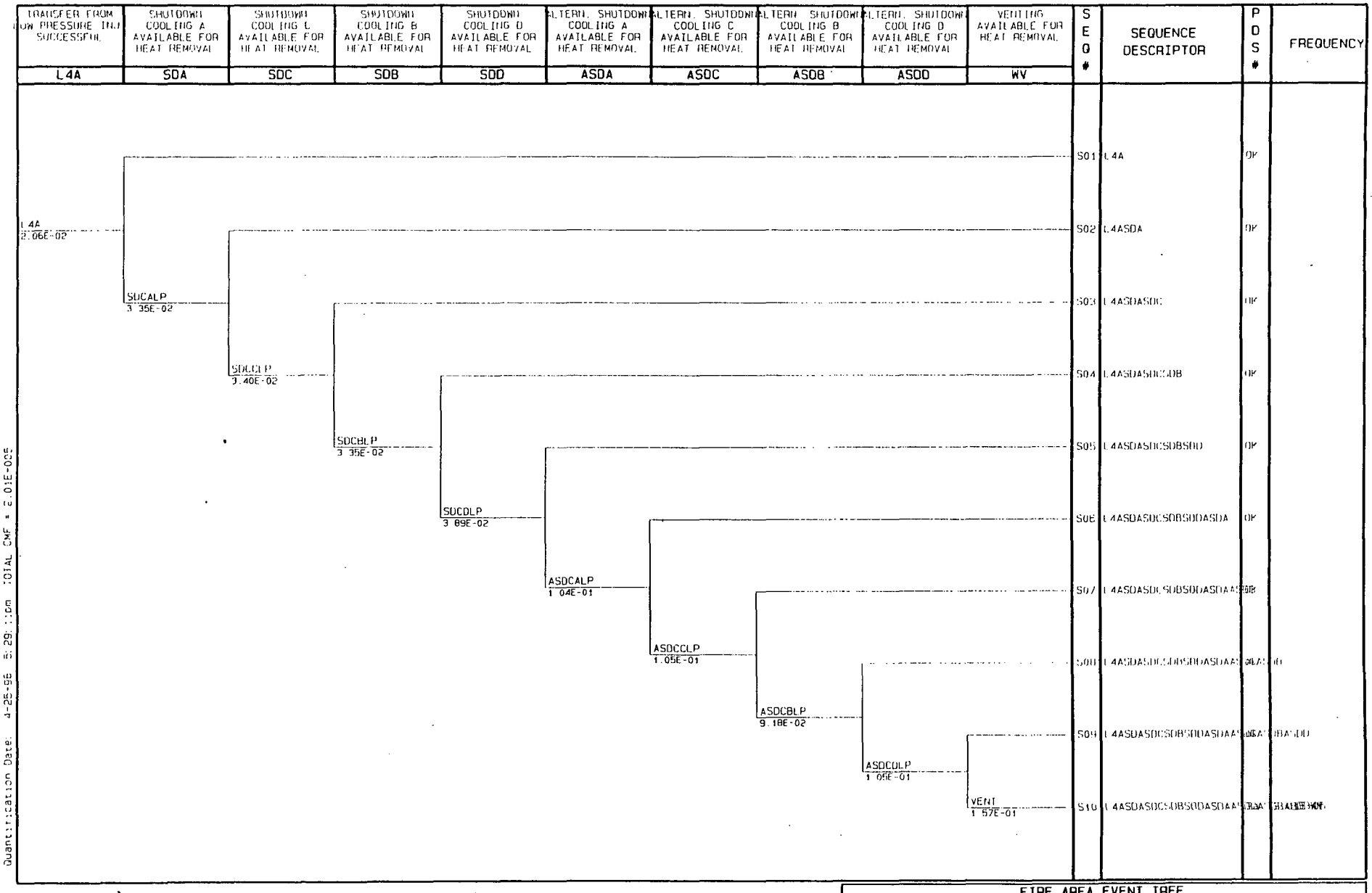
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM F1S03
CORES SPRAY A FOR INJECTION



FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM F1S04
NO HIGH PR. INJ. CSA FAILS



FIRE AREA EVENT TREE
LIMERICK GENERATING STATION

LOW PRESSURE INJECTION SUCCESSFUL
LOSS OF OFFSITE POWER

EPRI
Electric Power
Research Institute

Keywords:
Earthquakes
Seismic effects
Seismic qualification
Electrical equipment
Mechanical equipment
Equipment anchorage

EPRI NP-7498
Project 2722-23
Final Report
November 1991

Industry Approach to Seismic Severe Accident Policy Implementation

Prepared by
Jack R. Benjamin & Associates, Inc.
Yankee Atomic Electric Company
Risk Engineering, Inc.
Pickard, Lowe and Garrick
and
MPR Associates, Inc.

REPORT SUMMARY

Industry Approach to Seismic Severe Accident Policy Implementation

This report provides utilities with industry recommended guidelines for cost-effective seismic evaluation of nuclear power plants in response to NRC Generic Letter 88-20. Guidance is provided on application of seismic probabilistic risk assessment and seismic margin methods for full-, focused-, and reduced-scope evaluations. It provides strategies for coordinating these evaluations with similar reviews needed for resolution of Unresolved Safety Issue (USI) A-46.

INTEREST CATEGORIES

Nuclear seismic risk, design, and qualification
Nuclear component reliability
Risk analysis, management, and assessment

KEYWORDS

Earthquakes
Seismic effects
Seismic qualification
Electrical equipment
Mechanical equipment
Equipment anchorage

BACKGROUND NRC issued a severe accident policy for existing plants in August 1985. It describes the basis for NRC to use in resolving issues related to the potential for severe reactor accidents due to internal and external initiators. Individual Plant Examination for External Events (IPEEE), which includes earthquakes, is addressed in Supplement 4 of Generic Letter 88-20 and NUREG 1407. In these documents, NRC accepts two methods for evaluating plants for seismic events, seismic probabilistic risk assessment, and seismic margin assessment (SMA). NRC further classifies each U.S. nuclear plant into one of three categories for application of the SMA method: full-, focused-, or reduced-scope, indicating the level of review effort to be performed.

OBJECTIVE To provide the nuclear power industry with guidance for performing IPEEEs.

APPROACH The authors developed the basis for the guidelines through interactions with both NRC and utility industry representatives during the formulation of the Generic Letter. The interaction was coordinated by the Nuclear Utility Management and Resources Council. The iterative approach employed by the team allowed for resolution of issues before the issuance of the Generic Letter. The team developed cost-effective evaluation approaches based on seismic hazard relative to the plant design basis using results from EPRI report NP-6395-D. They also used the lessons learned from the three trial plant evaluations already performed by industry (Catawba, EPRI report NP-6359, and E. I. Hatch, report NP-7217) and NRC (Maine Yankee, NUREG/CR-4334). The guidelines direct emphasis and resources to those areas that have been shown to be the most cost-effective.

RESULTS The key features of the report are guidance on the use of the two methods acceptable to NRC for seismic IPEEE, including criteria for choice of methods, effective application to IPEEE, strategies for coordinating and combining the reviews with those to be performed to address USI A-46, and closure procedures and criteria for addressing potential vulnerabilities. The application of the margins method includes details on the level of effort appropriate for plants in each of the three NRC categories and provides procedures for limited evaluations of containment systems and components, an area not included in the original scope of seismic margins review in EPRI report NP-6041 or NUREG/CR-4334.

EPRI PERSPECTIVE The development of the guidelines in this report had a major influence on the requirements of the NRC Generic Letter in two areas. The first is the grouping of plants into the full-, focused-, and reduced-scope categories. The categorization is based not on absolute seismic hazard, as was originally envisioned, but on hazard relative to the design basis (safe shutdown earthquake) which allows accounting for the inherent margin in the plant's design basis in assigning a review category. This resulted in most plants being assigned to the focused-scope rather than the full-scope category. The second area was relay evaluation. Both of the EPRI trial plant evaluations (Catawba and E. I. Hatch) clearly showed that extensive relay review is a major cost element in the IPEEE and yields little or no increase in plant seismic safety. A simple check for certain predefined, seismically sensitive relays was determined to be much more cost-effective. The Generic Letter provides that focused-scope plants may limit the relay review to just such an approach within the constraints imposed by USI A-46. The interaction with NRC on these issues allowed common understanding with the industry. As a result, these methods are, with some minor exceptions, compatible with the Generic Letter.

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Industry Approach to Seismic Severe Accident Policy Implementation

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ABSTRACT

The Nuclear Regulatory Commission (NRC) issued a severe reactor accident policy for existing plants on August 8, 1985 which describes the formal basis by which the NRC intends to resolve issues related to potential severe reactor accidents. Examination of plant-specific vulnerabilities due to seismic and other externally initiated events was considered on a later schedule and is addressed in Supplement 4 of the NRC Generic Letter No. 88-20 and a NRC guidance document, NUREG-1407, issued in June 1991. This report was prepared to provide a coherent and effective approach for seismic severe accident review which meets the intent of Generic Letter No. 88-20, Supplement 4.

The recommendations in this report provide guidance on plant review types and review implementations which is consistent with the "limited-scope" intent of systematic evaluations as described in the NRC's Severe Accident Policy Statement. In addition, to assist in implementing cost-effective modifications that reduce vulnerabilities, this report also presents specific guidelines for identification and treatment of vulnerabilities that may be used as a basis for defining closure of earthquake-related severe-accident issues.

In line with the severe-accident policy statement, the approach proposed in this report for treatment of seismic issues focuses on the objectives of completing high-quality systematic plant evaluations, effectively identifying plant-specific vulnerabilities, and implementing improvements that are cost-effective in mitigating the risk impact of the vulnerabilities. The intent is to achieve an optimum seismic-IPE program, where severe-accident policy concerns are completely satisfied, yet industry-wide effort is not wasted on identifying potential modifications that are not cost-beneficial. The elements of the proposed approach are:

- Development of guidance on the type of systematic seismic evaluation to perform.
- Delineating effective review procedures for deterministic systematic evaluations.
- Delineating a scope of seismic review for probabilistic systematic evaluations.

- Development of procedures for effective integration of the deterministic seismic IPE and other unresolved seismic issues.

- Development of procedures and closure criteria to delineate potential vulnerabilities and resolve their treatment within cost-benefit guidelines.

This report provides procedural instructions and guidance to support resolution of earthquake-related severe accident issues. More detailed background and technical justifications for the methods are documented elsewhere, and are referenced throughout this report as appropriate.

ACKNOWLEDGEMENTS

The preparation of this report has benefitted from extensive reviews and interactions with members of the Nuclear Management and Resources Council, Seismic Issues Working Group (SIWG). We want in particular to recognize the value of extensive discussions with Dr. Orhan Gurbuz.

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INTRODUCTION

BACKGROUND ON SEVERE-ACCIDENT ISSUES FOR SEISMIC EVENTS

On August 8, 1985, the Nuclear Regulatory Commission (NRC) issued the document *Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants*, published in 50 FR 32138 (1). That statement describes the formal policy to be followed by the NRC to resolve issues related to potential severe reactor accidents; key highlights of the Commission's statement are noted as follows:

- Based on currently available information, the Commission concludes that existing nuclear power plants pose no undue risk to public health and safety;
- Based on NRC and industry experience with plant-specific PRAs (Probabilistic Risk Assessments), however, systematic plant examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents for which safety improvements may be justified;
- Each existing plant should, therefore, perform a systematic examination to identify any plant-specific vulnerabilities, and report the results to the Commission.

In response to a request by the Commissioners, the NRC Staff developed an implementation program for integrated closure of severe accident issues. This integration plan (2), consists of the following six major elements or sub-programs:

1. Individual Plant Examination (IPE) Program. A systematic examination of existing plants for severe accident vulnerabilities.
2. Containment Performance Improvements (CPI) Program. Development of generic containment performance improvements with respect to severe accidents to be implemented, if necessary, for the major containment types.
3. Improved Plant Operations Program. Development of improved NRC and utility programs for plant operations.
4. Severe Accident Research Program. Investigation of a variety of topics related to severe accident phenomena and progression.

5. External Events Program. Research to identify external events requiring severe accident examination, and development of procedures to conduct Individual Plant Examinations for External Events (IPEEEs).

6. Accident Management Program. Utility development and implementation of plant-specific, severe accident management plans.

Internal Events

NRC and industry programs addressing the above elements of the severe accident policy (SAP) integration plan are well underway. To commence with execution of the IPE program, on November 23, 1988 the NRC Staff issued a Generic Letter No. 88-20 (3) to licensees of existing plants, requesting them to perform an IPE for severe accident vulnerabilities that may be uncovered due to internally initiated events, and to report the results to the Commission.

The specific objectives of the IPE are, for each utility in charge of operating an existing plant, to (2):

- "Develop an overall appreciation of severe accident behavior."
- "Understand the most likely severe accident sequences that could occur at its plant."
- "Gain a more quantitative understanding of the overall probability of core damage and fission product releases."
- "Reduce the overall probability of core damage and fission product releases, if necessary, by appropriate modifications to procedures and hardware that would help prevent or mitigate severe accidents. (It is expected that achievement of these goals will ensure that the severe core damage and large radioactive release probabilities for U.S. nuclear power plants are generally consistent with the Commission's safety goal policy)."

A Level 1 PRA (4) (including containment-performance considerations) has been identified as an appropriate and recommended procedure by which internal-event IPEs may be conducted. A number of utilities are thus conducting or planning to conduct a Level 1 PRA for their plant(s).

Seismic Events

As noted in Generic Letter No. 88-20, examination of plant-specific vulnerabilities due to externally initiated events (e.g., earthquakes, internal fires, external floods and tornadoes) would be expected, but could proceed

separately and on a later schedule. This time lag was introduced to allow the NRC and industry to develop procedures to identify those external events requiring examination; to develop simplified, systematic examination procedures; and to integrate ongoing NRC programs [for example, unresolved safety issues (USI) A-45, A-46 and the Seismic Margins program] dealing with external events with the IPE program, to ensure efficient, non-redundant allocation of industry efforts in these programs. Utilities were encouraged to retain documentation and plant-specific data (e.g., as derived from plant walkdowns) from the internal-events IPE to facilitate the later conduct of their IPEEEs.

On November 5, 1989, the NRC Staff issued draft Supplement 4 to Generic Letter No. 88-20 concerning IPEEEs (5), following in March 1990 with a draft guidance for conducting the IPEEE (6). Earthquakes were identified as a major class of events requiring consideration. Another Draft Supplement 4 of the generic letter (7) and a revised draft of the guidance document (8) were issued in July 1990. The final NRC Generic Letter and NUREG-1407 were issued in June 1991 (9 and 10).

As outlined in the NRC documents, the specific objectives of the IPEEE are, for each utility in charge of operating an existing plant, to (2):

- "Develop an appreciation of severe accident behavior."
- "Understand the most likely severe accident sequences that could occur at the licensee's plant."
- "Gain a qualitative understanding of core damage and fission product releases."
- "If necessary, to reduce the potential of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents."

It is significant that the objectives of the IPEEE de-emphasize quantitative assessment of risks and use of safety goals. This de-emphasis clearly encourages the use of examination procedures other than PRA for conducting IPEEEs. For instance, the Seismic Margin Assessment (SMA) procedures and a reduced-scope implementation (Section 3) of SMA-type screening procedures have been identified by the NRC Staff as being appropriate for conducting the Individual Plant Examination for Seismic Events (seismic IPE).

It is generally recognized that the SMA-type walkdown procedures (11) result in thorough and efficient identification of "outliers" (or "weak-links") when

directed by a well-qualified Seismic Review Team (SRT). In fact, most knowledgeable engineers believe that such a well-focused, well-directed, thorough plant walkdown is the single-most important aspect of plant examination for identifying severe-accident vulnerabilities. For these reasons, application of PRA methods as an alternative to SMA-type procedures in conducting the Seismic IPE have been endorsed by the NRC Staff (9, 10) only if certain enhancements are undertaken; among these enhancements is the requirement that SMA-type walkdown procedures be implemented in the PRA walkdown.

Both the severe accident policy statement and the generic letter encouraged the nuclear utilities, through NUMARC (Nuclear Management And Resources Council), to propose a methodology for the IPEEE that meets the intent of severe accident policy (i.e., to efficiently find and correct, as justified, plant-specific vulnerabilities to external hazards). The development of such methodology for earthquake-related issues, as presented in this report, has relied on the understanding and appropriate use of both seismic hazard results and information on the seismic capacity of nuclear power plant structures and equipment.

BACKGROUND ON SEISMIC HAZARD AND REVIEW METHOD SELECTION

During the past 12 years extensive effort has been devoted to developing probabilistic seismic hazard procedures as a tool to assess the low-probabilities of exceeding seismic design bases for nuclear power plants. This work originally focused on early vintage plants which had design bases derived deterministically prior to implementation of the current siting guidelines contained in 10 CFR 100, Appendix A (12). More recently, the scope of this effort has been expanded to address issues of large earthquakes as well, in particular the so-called Charleston earthquake issue.

The Charleston Issue, briefly stated, is the hypothesis that large earthquakes may occur in the eastern United States (EUS) at locations where supporting tectonic conditions exist, even though such events have not been observed historically. The possibility that earthquakes of magnitude similar to that of the 1886 Charleston, South Carolina event may occur in regions throughout the EUS was formally raised in a U.S. Geological Survey (USGS) letter (13) which recommended to the NRC that "probabilistic evaluations of seismic hazard should be made for individual sites in the eastern seaboard to establish the seismic engineering parameters for critical facilities."

Responding to NRC concerns surrounding the Charleston Issue and its impact (if

any) on seismic safety, the nuclear utilities, working through the Electric Power Research Institute (EPRI), developed a methodology for seismic hazard assessment that specifically considered the Charleston Issue (see 14 and 15 for further background). The Seismicity Owners Group (SOG), an assembly of 42 nuclear utilities in the central and eastern United States was formed to finance, oversee, and advise the development and use of this methodology.

The EPRI-SOG methodology, consists of systematic procedures to determine the probabilistic seismic hazard at any site. The methodology accepts multiple input interpretations by earth scientists, and uncertainties resulting from these alternative interpretations are quantified by use of logic trees and propagated through the hazard results (16). Seismic sources were developed by six Earth Science Teams specifically to model the possible locations at which severe earthquakes might occur in the EUS, and to estimate the probabilities associated with those occurrences. In development of these interpretations, the six Earth Science Teams considered all proposed hypotheses on earthquake causes and characteristics in the EUS, and weighted those based on available data and evidence.

Using this state-of-the-art, EPRI-SOG approach, seismic hazard results (i.e., estimates of the probabilities of exceedance of ground-motion amplitudes) have been obtained for 58 nuclear power plant sites in the EUS (15). Uniform hazard (i.e., equal exceedance probability) spectra, obtained for each of the 58 sites, provide a complete description of the site-specific ground motion threat, including the effect of local soil conditions. These uniform hazard spectra results specifically consider the possibility of severe earthquakes and their influence on site ground motion characteristics; the EPRI-SOG uniform hazard spectra are, therefore, directly applicable to the present development of procedures for resolution of seismic severe-accident issues.

A separate hazard methodology has also been developed under the sponsorship of the NRC by Lawrence Livermore National Laboratories (LLNL). The LLNL-NRC methodology considers multiple expert opinion and accounts for the Charleston Earthquake Issue; it has been applied to obtain hazard estimates for 69 EUS nuclear power plant sites (17). Development of the LLNL-NRC methodology had proceeded on a somewhat earlier schedule than the EPRI-SOG program, but both programs produced hazard results for EUS plants at about the same time.

The NRC has expressed the position that both EPRI-SOG and LLNL-NRC hazard results

should be considered in decision making on seismic issues (18); in particular, for developing seismic IPE guidance, the NRC has required (informally) that both hazard results be used in grouping (binning) plants for selection of seismic IPE review levels. This report develops a seismic IPE approach that meets this NRC requirement. The primary objective has been to search for consistency among the two hazard results on a relative, plant-to-plant basis for guiding the process of grouping plants for similar review.

BACKGROUND ON SEISMIC CAPACITY OF NUCLEAR POWER PLANT STRUCTURES AND EQUIPMENT

In the past 12 years there have been at least three major programs that have addressed the issue of nuclear power plant seismic margin. The first program was Seismic Probabilistic Risk Assessment (SPRA) which has been performed for over 20 plants. Considerable resources have been expended on these analyses which have demonstrated that nuclear power plants are generally rugged for earthquakes. A few weak components have been identified; however, very few modifications have been required for seismic upgrading of nuclear power plants (19). The modifications that have been made were primarily the result of observations and findings made during the plant walkdowns.

A second program is currently addressing capacity of equipment in older nuclear power plants. In response to NRC unresolved safety issue (USI) A-46, the nuclear industry formed the Seismic Qualification Utility Group (SQUG) to address the seismic capability of equipment in these older plants. The purpose of the SQUG program is to demonstrate that older plants have adequate seismic ruggedness for safety equipment not seismically designed to current criteria (i.e., post-1973) to withstand the plant design Safe Shutdown Earthquake (SSE). The Seismic Qualification Utility Group has gathered an extensive data base on earthquake experience in fossil fuel power plants and heavy industrial facilities. This data base has been reviewed by both the NRC and a five-member Senior Seismic Review and Advisory Panel (SSRAP), who were jointly selected by the NRC and SQUG (20). With certain caveats and exclusions SSRAP has concluded that 20 classes of equipment in nuclear power plants are at least as rugged as similar equipment in the data base plants. Currently, detailed seismic walkdowns of the older plants are planned to address the adequacy of seismic anchorage, confirm compliance of the equipment with the SSRAP exclusions and caveats, and to look for certain seismic spatial-systems-interaction concerns (21).

In the third program, a Seismic Margin Assessment (SMA) was developed to address plant capability to reach safe shutdown for seismic motions beyond the SSE design

level (22). In response to this need a methodology was developed to determine whether high confidence of a low probability of failure (HCLPF) exists for a specified ground motion input. If a plant HCLPF does not exist for this level, the methodology provides a procedure for determining at what level a HCLPF can be stated. It is believed that SMA is more cost effective than SPRA, easier to use by the practitioners in the nuclear industry, and results in a better understanding of the plant's seismic performance.

While SPRA provides risk estimates, the results are highly uncertain. Seismic Margin Assessment does not provide risk values, but the HCLPF capacities are more certain since they are in the range of experience of most seismic engineers. The NRC methodology (22) and the procedures developed by EPRI (11) both provide alternate approaches to SMA for assessing seismic margin of nuclear power plants and to identify outliers, if any.

Based on the findings from these three programs there is a broad consensus that most safety-related equipment and structures necessary for shutdown and for containment of radioactive materials generally are inherently rugged. This consensus is reflected in the following documents which have broad authorship:

- American Society of Civil Engineers, Uncertainty and Conservatism in the Seismic Analysis and Design of Nuclear Facilities, 1986 (23).
- Expert Panel on the Quantification of Seismic Margin, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," August 1985 (22).
- Senior Seismic Review and Advisory Panel (SSRAP), "Use of Seismic Experience and Test Data to Show Ruggedness of Equipment in Nuclear Power Plants," February 28, 1991 (20).

The consensus among experienced practitioners is that equipment and structures at nuclear power plants have substantial seismic ruggedness. However, certain weaker elements have occasionally been found. Therefore, the principal seismic concerns are not with the vast majority of safety-related equipment and structures in operating plants, but with the potential for a limited number of seismically weaker elements (e.g., inadequate equipment anchorage) to affect plant safety. It is also believed that the plant walkdown phase of a seismic review is the most important step in finding these problems and any other potential vulnerabilities to earthquakes.

OUTLINE OF PROPOSED APPROACH

NRC's Severe Accident Policy requires that systematic evaluations be performed with the purpose of finding and correcting (within the guidelines of NRC Backfit Policy) severe-accident vulnerabilities. A fundamental objective of the Severe-Accident Policy is to verify the widely held belief that plants pose no "undue risk" and that "all reasonable steps are taken to reduce the chances of occurrence of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident should one occur." The Individual Plant Examination (IPE) program was developed by the NRC Staff to address particular facets of severe accident issues.

In its Severe Accident Policy Statement the NRC clarifies the level of effort the IPEs should involve: "licensees of each operating reactor will be expected to perform a limited-scope, accident safety analysis designed to discover instances (i.e., outliers) of particular vulnerability to core melt or to unusually poor containment performance, given core-melt accidents." Hence, a limited-scope plant investigation that makes effective use of insights gained through past detailed investigations, aimed at effectively identifying cost-effective mitigation of plant-specific vulnerabilities, is the course intended by the NRC Commissioners for implementation of IPEs in severe accident policy resolution.

The recommendations in this report provide guidance on plant review types and review implementation which is consistent with the "limited-scope" intent of systematic evaluations as described in the NRC's Severe Accident Policy Statement. In addition, to assist in implementing cost-effective modifications that reduce vulnerabilities, this report also presents specific guidelines for identification and treatment of vulnerabilities, that may be used as a basis for defining closure of earthquake-related severe-accident issues.

In line with the severe-accident policy statement, the approach proposed in this report for treatment of seismic issues focuses on the objectives of completing high-quality systematic plant evaluations, effectively identifying plant-specific vulnerabilities, and implementing improvements that are cost-effective in mitigating the risk impact of the vulnerabilities. The intent is to achieve an optimum seismic-IPE program, where severe-accident policy concerns are completely satisfied, yet little effort industry-wide is wasted on identifying potential modifications that are not cost-beneficial. The elements of the proposed approach, and the overall organization of this report, are outlined below:

- Development of preliminary guidance on the type of systematic seismic evaluation to perform (Section 2).
- Delineating effective review procedures for deterministic systematic evaluations (Section 3).
- Delineating a scope of seismic review for probabilistic systematic evaluations (Section 4).
- Development of procedures for effective integration of the deterministic seismic IPE and other unresolved seismic issues (Section 5).
- Development of procedures and closure criteria to delineate potential vulnerabilities and resolve their treatment within cost-benefit guidelines (Section 6).

This report provides procedural instructions and guidance to support resolution of earthquake-related severe accident issues. More-detailed background and technical justifications for the methods are documented elsewhere, and are referenced throughout this report as appropriate. In particular, Reference (24) describes bases for using seismic hazard results in rational decisionmaking related to the treatment of seismic issues, and Reference (25) describes bases for the treatment of high-frequency seismic ground-motion effects in nuclear power plants.

Section 2

SEISMIC REVIEW-METHOD DETERMINATION

BACKGROUND AND OUTLINE OF APPROACH

This section provides guidance to establish an efficient and effective basis for selecting the type of plant systematic evaluation to perform for the seismic IPE and for determining the ground-motion level at which the seismic IPE should be conducted (see Reference (26) for background). Plants are differentiated and grouped based on the combined use of seismic hazard results and seismic design bases, in the manner discussed in Appendix A. Delineation of binning categories is ambiguous in light of imperfect information on plant hazards and on the unclear relationship between plant margin and seismic design. Before deciding on a systematic evaluation approach to implement therefore, a utility is advised to factor its own knowledge of plant design, maintenance, and backfit history into the process of deciding how to proceed with the seismic IPE.

Based on judgment consistent with results of past studies, the binning proposed in this section results in a very efficient (industry-wide) seismic-IPE program that will identify cost-effective modifications which best enhance plant seismic safety. The desirability of conducting the IPE program in an efficient manner has been stressed by the NRC in the severe accident policy statement (1). Key points from the severe-accident policy statement in this regard pertain to the intent and desired implications of the seismic IPE, i.e., that: (1) systematic evaluations should be of limited scope; (2) they should efficiently reveal low-cost modifications of the type found in past PRAs; (3) they should serve as a basis for verifying conclusions developed from more intensive studies (e.g., NUREG 1150 27); and (4) only those modifications found to be justified within the cost-benefit criteria of the NRC backfit policy should be implemented.

In line with these points, the recommended approach emphasizes the use of well-focused, systematic approaches for the efficient identification of outliers. Procedures and criteria discussed in Section 6 provide guidance on closure of the seismic IPE, and can be used by utilities regardless of which systematic evaluation procedure (reduced-scope assessment, focused-scope SMA, full-scope SMA,

or PRA) they decide to implement. The format of these resolution criteria help to ensure that the major set of cost-effective and safety-effective modifications will be found under whatever format a utility chooses to undertake for its seismic IPE.

As developed in Section 3 and discussed in Appendix A, four systematic evaluation procedures are available for conducting the seismic IPE: (1) the reduced-scope assessment, (2) the focused-scope SMA, (3) the full-scope SMA, and (4) the seismic PRA (SPRA). The first three of these methods are deterministic implementations, and the last is probabilistic. Roughly speaking, in the context of the seismic IPE, the reduced-scope SMA is considered appropriate where the seismic hazard is low; the full-scope SMA is considered appropriate where the seismic hazard is comparatively high relative to the design basis; the SPRA, an alternative to the full-scope SMA, is considered appropriate in special situations where risk results may be anticipated to facilitate decisionmaking; and the focused-scope SMA is considered appropriate for the remaining bulk of plants that have comparatively moderate seismic hazard relative to design basis.

PLANT SEISMIC IPE GROUPS

Figure 2-1 shows results of ordered design-basis hazards (i.e., probabilities of exceeding plant design-motion levels) for eastern U.S. nuclear power plants, based on EPRI median hazard results. Figure 2-2 presents similar data based on LLNL median hazards. Using the results in these figures, one can differentiate plants based on the level of composite design-basis hazard. For instance, in each of the two plots, one can clearly distinguish the following two groups: (1) a small group of comparatively high design-basis hazard plants, and (2) a large group (comprising the bulk of the population) of plants with comparatively moderate-to-low design-basis hazard. The six plants with the highest design-basis hazard from the EPRI median results are the same six plants with the highest design-basis hazard from the LLNL median results. This consistency among EPRI and LLNL results lends confidence to immediately differentiating two groups of plants: a small group of plants with comparatively high design-basis hazard, and the remaining population of plants.

It is appropriate, however, to differentiate a third group of plants (as mentioned above) on the basis of having negligibly low seismic hazard, regardless of design basis. For instance, certain areas in the deep south and in the northern-midwest portions of the eastern U.S. (EUS) are known to be substantially quiet tectonically; because the seismic threat is so low, plants in these regions would

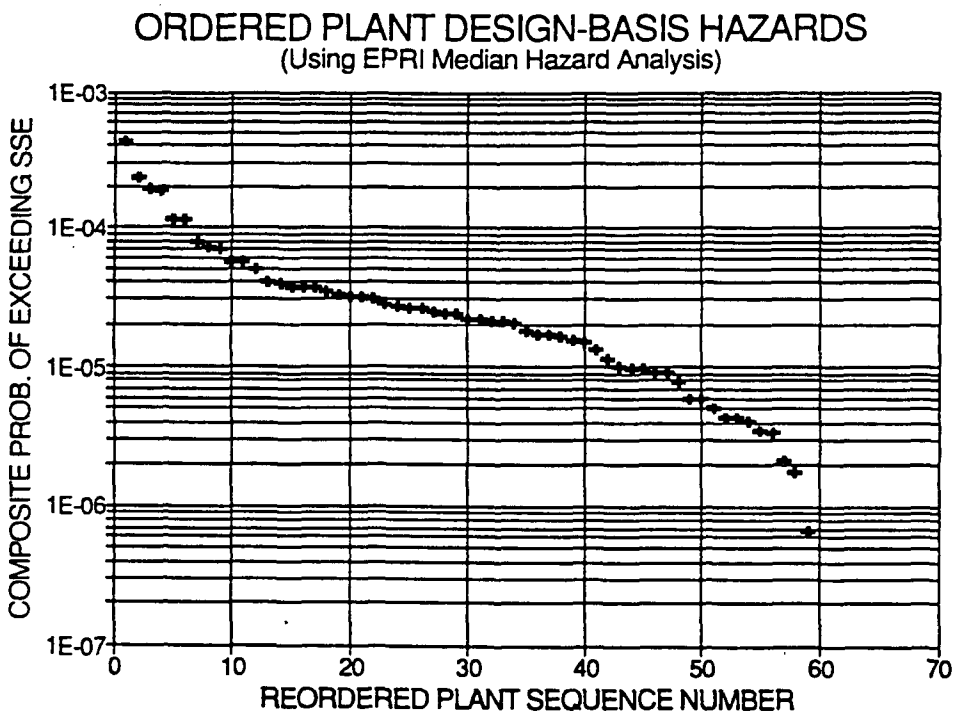


Figure 2-1. Ordered, composite design-basis hazards for EUS nuclear power plants, based on EPRI median results.

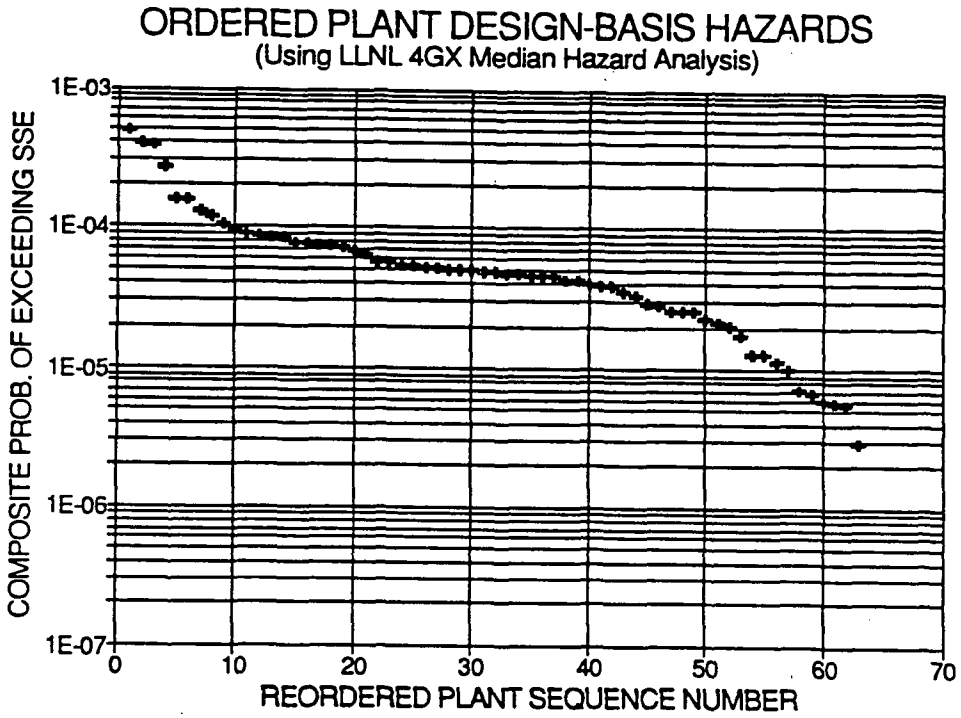


Figure 2-2. Ordered, composite design-basis hazards for EUS nuclear power plants, based on LLNL median results.

not warrant the same rigor in severe-accident evaluation as would plants in other areas of the U.S. The NRC has performed a plant ranking analysis (8), considering LLNL and EPRI hazard results, that identified such low-hazard plants. The analysis is based on a clustering methodology (developed by LLNL) applied to site hazard (as opposed to design-basis hazard) results. In their analysis, the NRC identified ten plants as belonging to a low-hazard group, where a reduced-scope severe-accident evaluation is appropriate. (Because these ten plants belong to a distinct group, they are not included in the plots of Figures 2-1 and 2-2.)

Based on Figures 2-1 and 2-2, and on NRC's identification of low-hazard plants, therefore, the following groups of plants are delineated:

- A small number of plants for which the design-hazard (i.e., probability of exceeding the SSE) is high compared to other plants (full-scope SMA category).
- Ten plants for which the site hazard is negligibly low (reduced-scope assessment category).
- The remaining (bulk of) the plant population, where the design-hazard is comparatively moderate (focused-scope SMA category).

The use of design-basis hazard and these three groupings is intended to provide guidance, not fixed rules or criteria, to plant licensees on the type of seismic IPE review they may wish to perform. For its own internal purposes, a licensee may wish to implement a more detailed review than would be recommended here. Morever, a licensee may use its discretion and judgment (based on its extensive experience and understanding of its plant) in deciding that a less detailed study than that identified by the procedure described here may be more suitable for implementation at its plant. In some instances, it may be useful for a licensee to undertake additional study to assist in its decision. In addition to the guidelines discussed here, the NRC also identifies, in its final revision to Supplement 4 of Generic Letter 88-20, the specific plants which it deems appropriate as belonging to each of these plant groups. Licensees are encouraged to also review that final document to assist in selecting seismic-IPE review types appropriate for their plants.

For the ten plants that have negligibly low seismic hazard, the need for any seismic severe-accident evaluation at all is questionable; for such plants, a reduced-scope set of procedures that emphasize a thorough plant walkdown has been developed (28). For the ten EUS plants identified above as falling in a

low-hazard group, therefore, a reduced-scope assessment is recommended for implementation of seismic IPEs.

A focused-scope SMA will find all design-independent potential vulnerabilities and also those design-dependent potential vulnerabilities that may compromise plant resistance to severe seismic accidents, the same potential design-dependent and design-independent vulnerabilities as would be found in a full-scope SMA or SPRA. The focused-scope SMA, therefore, is considered an appropriate procedure for severe-accident review of any EUS plant. For some plants where the seismic hazard or the probability of exceeding the design basis (i.e., SSE spectrum) is markedly larger than the major remaining plant population, it may be of benefit to licensees and to the NRC to have very realistic capacity calculations for certain components to assist in decisionmaking and in seismic-IPE resolution. In addition, in cases of higher seismic hazard, it may be prudent to perform (as discussed in greater detail in Section 3) a relay chatter evaluation, with the specific purpose of identifying and replacing known vulnerable relays and contacts. For these higher hazard plants, implementing the set of full-scope (vis-à-vis focused-scope) SMA procedures is considered potentially useful, although not clearly warranted.

For the set of EUS plants having comparatively high design-basis hazard, therefore, a full-scope SMA is recommended for implementation of seismic IPEs. (An SPRA would, however, generally be recommended for those plants where an SPRA has been performed previously). For the remainder of EUS plants that do not belong to either the reduced-scope or full-scope SMA class, a focused-scope SMA is recommended.

In all cases for EUS plants, SMA screening against the 0.3g PGA (0.8g spectral acceleration) screening tables is considered sufficient to identify severe-accident outliers. For some Western U.S. (WUS) plants, the severity of the seismic environment might lead one to conclude that a full-scope SMA at 0.5g PGA (1.2g spectral acceleration) or an SPRA may be appropriate or required for severe-accident review. Without seismic hazard results for these plants, a seismic IPE recommendation cannot be made. However, given the generally higher seismic design levels for WUS plants and given the geographic diversity of the WUS seismic threat, it is reasonable to suggest, based on the approaches in this document, that some WUS plants may be differentiated into a classification where less than full SPRA or full-scope SMA procedures are justifiably appropriate.

This approach for review-method selection of plant seismic-IPEs focuses on the clear merit of the following objectives: (1) emphasizing the value of performing a thorough plant walkdown, an accomplishment that will identify potential safety enhancements that past experience indicates are important or cost-beneficial to fix; (2) de-emphasizing the tasks of performing calculations and producing a paper product, items that clearly will not enhance plant safety; (3) de-emphasizing the task of performing evaluations (e.g., of relays) that past experience says will have a very low probability of identifying or achieving any significant, cost-effective safety enhancement; and (4) helping to insure that the work of available qualified technical professionals will be focused in such a manner that encourages a high-quality end product and the development of optimal safety-enhancement solutions. Each licensee is encouraged to use the guidelines presented in this section, factored with plant-specific information, to decide on the review method most appropriate for seismic-IPE implementation at its plant.

This process of determining seismic review method, by itself, does not assess earthquake review levels. The review method, consistent with conventional SMA procedures, may specify a standard (default) review-level ground motion. For full-scope and focused-scope SMAs, the review motion is taken as a NUREG/CR-0098 median spectrum (29) anchored to a PGA of 0.3g (9). For the reduced-scope assessment, because the seismic hazard is negligibly low, the severe-accident review earthquake is taken as the SSE spectrum (or licensing commitment) itself. Further guidelines on seismic-IPE implementation procedures and on ground-motion input are provided in the next section.

Section 3

DETERMINISTIC SEISMIC REVIEW IMPLEMENTATION

The requirements for deterministic seismic review are given in this section and are provided for the following three types of review:

- Full-Scope SMA
- Focused-Scope SMA
- Reduced-Scope Assessment

Depending on the review bin selected for a plant (i.e., see Section 2) one of the three types will be performed.

Seismic margins assessment (SMA) determines whether a plant has a high confidence of a low probability of failure (HCLPF) capacity for core damage and release from containment, for a selected review level earthquake (RLE). For these cases it is stated that the plant HCLPF equals or exceeds the RLE. When it is found that the HCLPF is less than the RLE, then the actual HCLPF for the plant is calculated.

The SMA is a walkdown-based seismic assessment (similar in approach to the A-46 seismic review). It focuses on structures and equipment in two independent safe shutdown paths and is based largely on available seismic experience, test and fragility data. In this approach, experienced engineers perform screening walkdowns to assess the capacity of identified safe shutdown piping, equipment and structures to withstand a review level earthquake which is well above the plant's licensing basis Safe Shutdown Earthquake (SSE). The SMA review level earthquakes for U.S. plants have been specified by the NRC and are given in Supplement 4 to NRC Generic Letter 88-20. For most U.S. plants, the review level is 0.3g. Four western U.S. plants in high seismic hazard areas have a review level of 0.5g; ten plants in very low seismic hazard areas in the South, Southwest and Northern Midwest will have a review level earthquake equal to their design basis SSE.

Equipment and structures which do not pass the deterministic screening criteria at the review level are identified as outliers and are subsequently evaluated in more detail to determine the earthquake level at which there is a "high confidence of a

low probability of failure" (HCLPF). The lowest of these HCLPF levels sets the HCLPF for the plant. These outliers are evaluated on a judgmental cost/benefit basis to determine the merits of various corrective actions to raise the plant's HCLPF level. Relay chatter is addressed in a somewhat simplified manner based on the A-46 approach.

Two approaches for performing an SMA are recognized by the NRC Generic Letter; the EPRI method, and the NRC method.

The EPRI SMA methodology has been used in two trial plant reviews, is documented in Reference (11), and the original version has been accepted by the NRC. The NRC method is documented in NUREG/CR-4334 and has been used in one trial plant. The HCLPF is obtained for core damage and is based on either event and fault trees (NRC method) or on components in a deterministically selected success path (EPRI method). For the latter approach, the plant HCLPF is the lowest HCLPF of all components in the success path selected for review.

In the EPRI methodology, the HCLPF is calculated using deterministic procedures that are generally familiar to the design engineer. The factors of safety are liberalized in SMA compared to plant design procedures to reflect the philosophy that a HCLPF has approximately a 95% confidence of about a 5% probability of failure at the Seismic Margin Earthquake (SME) level. A HCLPF can be determined in either a deterministic or probabilistic manner. The probabilistic approach is based on the same procedure as used in fragility analysis in a SPRA.

The procedures for performing a SMA are very similar to a A-46 review. Selection of review components, plant walkdown, screening, and evaluation of outliers are common steps to both methods. Both approaches rely heavily on engineering judgement by experienced, trained, seismic engineers and seek to eliminate unnecessary work by focusing efforts on the most significant concerns.

IPEEE plants which the NRC has designated as 0.3g RLE plants will either perform a full-scope review that follows NP-6041, Reference (11), or a focused-scope review which requires less effort. Plants selected for a reduced-scope review follow only selected portions of the SMA.

Table 3-1 summarizes the three types of reviews. The recommended requirements for the three types are given below. The discussion parallels the summary which is given in Table 3-1.

Table 3-1

SUMMARY OF RECOMMENDATIONS FOR DETERMINISTIC PLANT SEISMIC REVIEW[1]

Type of Review	Success[2] Path Elements	Containment	Relays	Soil Failure	Screening Criteria	Outliers	Input	Documentation
Full-Scope SMA	NP-6041 and Containment Systems[3]	Yes[3]	Yes[4]	Yes[6]	NP-6041[8]	HCLPF Calculations	NUREG-0098	NP-6041 (CH 8)
Focused-Scope SMA	NP-6041 and Containment Systems[3]	Yes[3]	Yes/No[5]	Yes[7]	NP-6041[8]	HCLPF[9] Calculations	NUREG-0098	NP-6041 (CH 8)
Reduced-Scope	NP-6041	No	No	No	NP-6041[8]	FSAR (or GIP for A-46)	SSE	Concise

[1] Utilities may propose procedures different than those shown in this table
[2] Elements identified should be consistent with type of review and IPEEE enhancements to be included.
[3] Functions required for containment integrity, isolation, prevention of bypass.
[4] Search for low seismic ruggedness relays only.
[5] Review for low seismic ruggedness relays. For A-46 plants, if low seismic ruggedness relays are found, expand to IPEEE review.
[6] Review potential soil failure modes (i.e., instability, settlement, and liquefaction).
[7] Review based on existing soils analyses, soils test reports, and design and construction records only.
[8] A-46 GIP may also be used to screen components.
[9] Order outliers, perform bounding calculations for lowest fragility components only.

It is important to note for all deterministic reviews that a plant walkdown will be conducted by a Seismic Review Team (SRT). The steps in the walkdown will be the same in all cases. Experience in past seismic SPRAs and SMAs indicates that significant seismic concerns have been identified in the walkdown process. The only exception might be the refueling water storage tank (RWST) which was modified as result of the SMA conducted for the Maine Yankee Nuclear Power Plant. However, flat-bottom tanks are a special concern which must always be considered in any seismic capability assessment. It is the widely-held opinion of many seismic capability engineers that the plant walkdown is the most important step in a seismic review, which will address the same structures and equipment independent of the type of review. It is anticipated that most, if not all, seismic performance issues will be discovered during the walkdown.

IDENTIFICATION OF SUCCESS PATH ELEMENTS

For all three types of deterministic review, procedures for identifying structures and equipment to be reviewed are the same and are based on the recommendations in EPRI Report NP-6041 (11). Depending on the system functions suggested for review type as discussed below, primary and alternate success paths for achieving shutdown should be selected as outlined in EPRI Report NP-6041. The approach for identification of systems and components needed to prevent early containment failure are discussed in Appendix D of this report. Note that the steps and investigations leading to the selection of structures and components are essentially identical and independent of which type of deterministic review is performed.

Identification of low seismic ruggedness relays in the systems analysis is discussed below.

CONTAINMENT REVIEW

The EPRI Seismic Margin Assessment approach includes evaluation of systems and components that are included in the "success paths" that will assure safe shutdown of the plant and maintain it in a safe condition for 72 hours. The EPRI SMA methodology does not include consequence mitigation (i.e., "containment") systems in the scope of evaluations. Considering the "defense-in-depth" principle, the draft Generic Letter 88-20, Supplement 4 also requests evaluation of containment performance.

It is recommend that the full- and focused-scope SMA reviews be limited to

evaluation of only those functions that are necessary to insure containment isolation and to prevent containment bypass and early containment failure. For essentially all containment designs, this includes: (1) successful containment isolation; (2) maintaining containment structural integrity (including penetrations and closures); and (3) prevention of containment bypass. For pressure suppression containment designs, the equipment necessary for the pressure suppression function (e.g., the suppression pool and the vent system for BWRs or the ice buckets, ice chamber and inlet/outlet "doors" for ice condenser designs) would be included in the scope of seismic reviews. In addition, drywell sprays (for Mark I) or hydrogen control (for uninerted Mark II, Mark III or ice condenser containments) may be required to prevent early containment failure. Because there is a significant variation in containment designs, input and guidance should be obtained from the IPE internal events containment evaluation team to determine if any other systems are required to prevent early containment failure. The success path logic diagrams (SPLDs) described in EPRI Report NP-6041 (11) denote those systems which are necessary to provide a long term safe shutdown condition. The SPLDs should be extended to include containment functions necessary to prevent early containment failure assuming that severe core damage has occurred.

Evaluation of systems and equipment whose functionality is required to prevent long term containment failure is not considered necessary because previous PRAs indicate that risk to the public due to severe accident sequences involving failure of long term containment integrity is low. This implies, for example, that for large, dry containments, review of the spray systems and fan coolers should be excluded from the scope of the IPEEE. Moreover, physical examination of some of this equipment may not be practical due to access requirements for entering radiation areas.

RELAY EVALUATION

The SMA conducted for the Hatch Nuclear Plant clearly demonstrated that relay review is not cost effective (30). Relay evaluations at three other plants have shown that the only relays that were found to be not sufficiently rugged were those on the low seismic ruggedness relay list or those eliminated by operator actions. Furthermore, relay chatter has not been a significant issue at non-nuclear facilities and industrial sites that were subject to ground motions on the order of 0.3 g. On this basis, the scope of relay chatter review should be limited as follows:

Full-Scope SMA

It is recommended that plants that perform a full-scope SMA only conduct a review for low-ruggedness relays. However, plants performing an A-46 review should conduct a relay review according to the procedures in the GIP (21) for all relays within the scope of the A-46 review. Investigation of relays outside the A-46 review but within the scope of the IPEEE review should address only the issue of low-ruggedness relays.

Focused-Scope SMA

A full relay review will be performed for A-46 plants in any case. For non A-46 plants a search for low-ruggedness relays as described in the GIP will be conducted. For A-46 plants further review of relays outside the scope of A-46 is generally not required. This is justified because there will be a review for the relays included in the A-46 scope of work and extending the list to include the IPEEE scope is unlikely to be cost-effective. However, if a low seismic ruggedness relay is discovered during the A-46 program, then the scope of plant relay investigations for that plant should be expanded to check for low-ruggedness relays outside the A-46 scope but within the IPEEE and to assess the systems implications of their chatter.

Reduced-Scope SMA

No relay chatter review is needed for the plants in this category since it is unlikely that any cost-effective seismic risk reduction opportunities will be found.

The relay chatter review for IPE of seismic events should consist of the following steps:

- Identify relays that are part of the selected success paths which are considered to have low seismic ruggedness (e.g., based on Reference (11))
- Determine if the consequence of contact chatter for the identified low seismic ruggedness relays is unacceptable.
- Ascertain whether systems affected by susceptible relays can be reset in a timely manner after ground motion has ceased, and well before their function is needed.

Relays which fail to pass the ruggedness, consequence or recovery tests are considered to be outliers. These relays should be replaced, or a more detailed analysis should be conducted to determine their disposition.

SOIL FAILURE INVESTIGATION

Soil failure has not been found to be a significant issue in past SMAs and SPRAs. However, for some plants on relatively soft-soil sites, the margin for soil failures may not be as high as it is for typical nuclear power plant structures and equipment. As a matter of prudence, it is recommended in a full-scope review that potential soil failure modes (i.e., instability, settlement and liquefaction) be reviewed as required in EPRI Report NP-6041. It is anticipated that existing soil test data will be adequate. A review of plant site conditions, using state-of-the-art approaches, will quickly determine whether soil failure is a significant issue. For plants in the focused-scope SMA category, a review based on existing soils analyses, soils test reports, and design and construction records is considered adequate. A review of soil failure should not be required for plants in the reduced-scope bin.

SCREENING CRITERIA

The SRT must perform the walkdown themselves and take complete responsibility for all elements screened out. At the conclusion of the walkdown concise documentation will be prepared which records the basis for screening out each component and which is signed by all members of the SRT.

There are two basic parts to the inspection of each component. First, the structural integrity and/or functionality of a component (exclusive of the anchorage) is considered. The guidance given in the EPRI Report NP-6041, Tables 2-3 for structures and 2-4 for equipment can be used to screen components (11). For the full- and focused-scope review the first column criteria in these tables, corresponding to 0.8 g spectral acceleration (i.e., which replaces the 0.3 g peak ground acceleration limit, as discussed in Appendix B) should be used. Note that plants in the Western U.S. (i.e., with a review level earthquake of 0.5 g pga) would use the 0.8 g to 1.2 g S_a column.

For older plants which are also being reviewed for the A-46 program, as well as other plants, the owner utility can use the walkdown and screening requirements in the GIP in addition to the SMA screening tables as discussed below. Note that the Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment (GIP) (21) procedures are generally more conservative than the SMA screening tables. However, there may be some cost savings to using the GIP for plants already performing A-46 reviews (e.g., see Appendix C). If the provisions of the GIP are used, all caveats given in that report must be followed for the

components screened by that approach (e.g., the bounding ground response spectrum in the GIP must exceed the review level earthquake response spectrum).

The second part of the walkdown inspection is concerned with anchorage adequacy. The screening tables given in the EPRI Report NP-6041 are primarily for the capacities of elements. Anchorage must be considered in addition to the guidance given in the screening tables, which generally only considers structure and component integrity and functionality. It should be noted that anchorage for all cabinets which contain relays included in the selected success paths must be reviewed, even though the relays per se are not evaluated. Guidance for anchorage review is given in EPRI Report NP-6041.

Ultimately, the SRT must certify that the anchorage capacity exceeds the seismic demand based on the input as defined above. It is recommended that prior to the plant walkdown the SRT review the construction drawings and specifications, and develop generic capacities for as many of the anchorage configurations as practical. Hopefully, existing design calculations can be used as a direct means for determining anchorage capacities. If design computations are not immediately available then specific calculations will have to be performed for each group of similar components, or for individual components, if they are each significantly different. By becoming calibrated for the plant-specific anchorage details and seismic input the SRT can more efficiently screen anchorage during their walkdown. The product of the plant walkdown by the SRT is concise documentation listing components which were not screened out in the walkdown (i.e., outliers) and the basis for screening out all other components identified in the success paths.

EVALUATION OF OUTLIERS

For both full- and focused-scope SMA reviews, HCLPFs should be determined for elements not screened out during the walkdown. A principal difference between full- and focused-scope reviews is in the number of components for which HCLPFs should be calculated or estimated. For elements in the focused-scope review, it is recommended that judgement be used to rank the capacities of the outlier structures and equipment from the lowest to the highest. HCLPF capacities should be calculated as necessary (only for approximately the lowest one-third of the ranked components). The remaining components should be assigned a conservative HCLPF based on the highest calculated HCLPFs. This will reduce the analysis for plants in the focused-scope bin.

For some components where it may be difficult to calculate a HCLPF for the "as-is"

condition (e.g., batteries with no spacers or unanchored equipment), the HCLPF can be computed for the modified upgraded configuration. The input is the same as used in the screening analysis discussed below. Guidance for calculating HCLPF values is given in EPRI report NP-6041 (11).

For plants in the reduced-scope bin, which are also in the A-46 program, the outliers should be evaluated for the requirements in the GIP. For elements outside the scope of the A-46 review (e.g., structures and piping), the requirements of the plant FSAR should be used in the evaluation. For plants in the reduced-scope bin which are not being evaluated for the A-46 program, the requirements of the plant FSAR should be used in the evaluation for all elements in the success paths. All elements which do not meet the acceptance criteria should be addressed by using the normal plant procedures to resolve safety issues.

Structures and components which are screened out based on the plant walkdown and review of drawings and specifications are not considered further. By intention, the SMA philosophy dictates that the SRT has high confidence of a low probability of failure (HCLPF) for these elements. Outliers identified in this process are reviewed further to determine their HCLPF values. The term "outlier" does not imply that a plant procedure and/or physical modification is required.

It is recommended that when simple modifications will increase its margin, a component be strengthened. For potentially expensive changes, more realistic detailed calculations or component testing can be used to justify that the outliers are not truly safety-related deficiencies relative to the plant design basis. Also, alternate equipment or procedural changes should be considered.

SEISMIC INPUT

For full- and focused-scope SMA reviews it is recommended that the ground response spectra be the NUREG/CR-0098 median curve anchored to the review level peak ground acceleration for the plant (i.e., 0.3 g or 0.5 g). The guidance provided in EPRI Report NP-6041 should be used to develop in-structure response spectra for the evaluation, when necessary. The calculated spectra should be median centered. As discussed in Section 6 site-specific response spectra may be used in closure evaluations. For plants performing a focused-scope review, the use of simplified scaling procedures is encouraged in order to reduce the analysis cost. However, this may lead to development of conservative HCLPFs in these cases.

The input for the reduced-scope review should be the in-structure response spectra

developed for the SSE ground response spectrum. New floor spectra can be developed which reflect state-of-the-art, soil-structure interaction models and building response analyses. In this case, however, to be consistent with the conservatism in the design input, mean plus one standard deviation level in-structure response spectra should be developed.

REVIEW DOCUMENTATION

The documentation of the IPEEE for the full- and focused-scope SMA reviews should follow the guidance outlined in Chapter 8 of EPRI Report NP-6041. A list of HCLPF values are to be provided for elements not screened out or found to have HCLPF values less than the review level earthquake. However, in a focused-scope review HCLPFs are calculated for only about one third of the screened-in elements. Therefore, HCLPFs for the remaining elements can be conservatively assigned based on the highest calculated HCLPFs. Note that in both the full- and focused-scope reviews HCLPFs for some equipment may be estimated by conservative comparison.

At the conclusion of the review the SMA will be documented as required in EPRI report NP-6041. A list of all structures and equipment identified in the systems analysis is divided into two groups. The first group contains all elements which were either screened out during the walkdown or have calculated HCLPF values equal to or greater than the review level earthquake. The SRT will document the basis for their decisions. The second group will consist of a list of elements where the calculated HCLPF values, which are less than the review level earthquake will be provided.

The report for the reduced-scope review should be concise and should include the systems and elements identified in the success paths, the procedure for the walkdown and findings and the resolution of all outliers.

Section 4

SCOPE OF SEISMIC REVIEW USING SPRA APPROACH

Draft Generic Letter 88-20, Supplement 4, identifies seismic PRA (SPRA) methods as being acceptable for use in conducting the IPE of seismic events. Seismic probabilistic risk assessment produces a mean core damage frequency (CDF) and ranking of accident sequences, systems, and structures/equipment that are potentially significant contributors to core damage risk due to seismic events. A SPRA should be a Level 1 PRA with a partial Level II analysis to address containment performance. Seismic probabilistic risk assessments may be performed using the procedures described in NUREG/CR-2300 (31), NUREG/CR-2815 (32) or NUREG/CR-4840 (33).

In performing a SPRA for IPEEE, a mean seismic hazard curve, event and fault trees, and structure and equipment mean fragility curves are combined. Hazard curves, based on the studies performed at EPRI and Lawrence Livermore National Laboratory (LLNL), have been developed for most Eastern U. S. nuclear power plant sites. It is the industry's position that the EPRI curves are more realistic and should be the ones used. The NRC has consistently indicated that site-specific seismic hazard curves developed by both LLNL for the NRC, and by EPRI for the Seismicity Owner's Group (SOG) (or the higher of the two) should be used in the SPRA. However, it is the industry position that use of both sets of curves, or the highest of the two, is not needed or justified since the dominant sequences will be the same, and since only relative CDF values are important. The absolute values are to be deemphasized due to the large uncertainties.

For Western U. S. nuclear power plant sites, hazard curves applicable to each site will have to be obtained directly by the plant owners. These site hazard results present ground motion parameters, such as peak ground acceleration (PGA) and response spectra, for different annual probabilities of exceedance.

The SPRA approach results in identification of seismic-induced accident sequences, systems and specific equipment/structures which are significant contributors to the calculated risk. Potential chatter of relays and other contact devices as a result of earthquake excitation is not normally covered in the SPRA methodology,

but has been addressed in recent SPRA evolutions, and is required for IPEEE. Plant walkdowns similar in scope to SMA are required for SPRA. The information to be gathered during the walkdown is essentially the same as in a SMA review. Seismic probabilistic risk assessments conducted in the past may require new walkdowns to bring them up to the same level of thoroughness as currently requested by the NRC. The EPRI screening criteria in EPRI NP-6041 can be used to screen elements in a SPRA. It is assumed that screened out elements are not significant contributors to core damage risk. This can be verified easily by bounding analysis.

Elements which are not screened out are analyzed to develop fragility curves, which are characterized by median capacities and logarithmic standard deviations for variability.

A detailed guidance document for performing the SPRA that is comparable to the GIP for an A-46 review or NP-6041 for a SMA review does not now exist. However, References (31), (32), and (33) provide procedures and general information that can be used to assist in performing a SPRA. The systems analysis for SPRA involves developing event and fault trees that logically relate the plant structures and equipment which are significant to preventing core damage. Non-seismic failures, human errors, and dependencies between elements are considered. For each structure and component, fragility curves are developed that typically have been expressed in terms of a lognormal model with a median capacity and a logarithmic standard deviation for uncertainty and randomness. Fragility curves relate probability of failure or malfunction of a specific type of equipment to the intensity of loading (e.g., level of shaking). For IPEEE, only mean fragility curves are required; hence, variability can be combined into a single parameter for each element, which simplifies analysis.

Next, all the fragility curves for the elements in the event and fault trees are combined in a probabilistic manner to produce a mean core damage fragility curve. Intermediate output, including mean fragility curves for each of the accident sequences (or combination of sequences making up a particular plant damage state) that contribute to the probability of core melt can be obtained. Finally, the mean core damage fragility curve for each plant is integrated with the mean site hazard curve to obtain the mean frequency of core damage due to seismic events. The mean frequency of core damage is normally expressed as an annual probability.

The procedures in SPRA can be used to obtain a mean containment release frequency.

However, the NRC has only requested that vulnerabilities be identified for containment-related systems/functions that could lead to early containment failure and result in high consequences. Recommendations for performing a SPRA are summarized below.

PLANT WALKDOWN AND DOCUMENTATION

As part of performing a SPRA, a plant walkdown should be performed which is consistent with the procedures and guidelines used in a walkdown conducted for a SMA review (11). The documentation of the walkdown and PRA should be consistent in detail with the requirements for a SMA review.

USE OF SEISMIC HAZARD RESULTS

In a SPRA, the emphasis is placed on relative ranking of dominant accident sequences that contribute to overall risk; bottom line numbers should be deemphasized because of the large uncertainties. Since the EPRI and LLNL seismic hazard curves result in only minor variations, if any, in sequence ranking, use of only one set of hazard curves is adequate. Thus, it is recommended that the hazard results presented in the EPRI Report NP-6395-D (15) be used in performing the SPRA.

At sites where only LLNL hazard results are available, use of those results in the SPRA may be appropriate. However, licensees may wish to perform computations using the EPRI seismic hazard procedures in order to develop new seismic hazard data.

FRAGILITY CALCULATIONS

Mean fragility curves are considered adequate for use in a SPRA rather than a family of curves. Fragility curves should reflect the data obtained during the walkdowns. Potential for soil failure (e.g., slope stability, settlement and liquefaction) should be considered similar in scope to the requirements for SMA given in Section 3.

RELAY CHATTER

Relay chatter evaluation has not been generally addressed in past SPRAs; although, some recent SPRAs have addressed the relay chatter in detail. As noted in Section 3, evaluations performed as part of a SMA have been shown not to be cost effective. The USI A-46 program has identified the most common low seismic ruggedness relays; and the relay evaluations performed so far have shown that all

suspect relays are either on this low seismic ruggedness list or have been screened out by operator actions or systems considerations. Therefore, consideration of relays in a SPRA should parallel the effort for a deterministic review. For a plant at a full-scope site low seismic ruggedness relays can be explicitly considered in the SPRA in the fault trees. Although it appears that a SMA-type review for relays could be performed in lieu of embedding the relay chatter investigation in the SPRA. At a focused-scope site only a search for low seismic ruggedness relays should be conducted similar to a full-scope PRA. Finally, if a SPRA is conducted for a plant at a reduced-scope site no relay review is necessary.

HCLPF CALCULATIONS

Although the final Generic Letter 88-20, Supplement 4 makes calculation of HCLPF values for components, sequences, and the plant optional, SPRA without HCLPF values fully satisfies the IPEEE objectives stated in the Generic Letter 88-20, Supplement 4. Therefore, it is recommended that HCLPF values not be provided.

Section 5

INTEGRATION OF IPEEE AND A-46 REVIEWS

CONSIDERATIONS IN SELECTING THE IPEEE APPROACH

There are a number of considerations in selecting the best method of complying with the IPEEE requirements at a given A-46 site. These include the utilities overall objectives for use of the IPEEE results, cost, availability of prior seismic review work and data, availability of trained, experienced seismic engineers, and compatibility/overlap with the A-46 seismic review program. These factors are discussed briefly below in the form of advantages and disadvantages of the two approaches.

Seismic Margins Assessment

Advantages:

- The SMA is based largely on the same approach as an A-46 review and is proceduralized in a manner similar to the Generic Implementation Procedure (GIP). As a result, the SMA can be more easily integrated into the A-46 walkdown by utility engineers so as to minimize duplication of effort. The feasibility of this integration was demonstrated to the NRC's satisfaction at Plant Hatch.

- Equipment and structures to be reviewed in the SMA will normally be based on deterministic selection of safe shutdown paths. Significant potential exists for selecting shutdown paths such that the majority of equipment reviewed is the same equipment as that reviewed for resolution of A-46. Also, the option exists to select paths with higher equipment HCLPFs, so as to raise the plant HCLPF value.

- If components are screened out or have calculated HCLPFs that exceed the review level earthquake (RLE) then the components are adequate and do not need to be considered further. If all of the components exceed the RLE then the closure procedures (i.e., Section 6) do not have to be used.

- A combined SMA/A-46 review has been successfully completed (30).

Disadvantages:

- The Seismic Margins Assessment does not provide numerical risk

estimates as does the SPRA; thus, making cost/benefit analyses more difficult. Also, SMA does not provide insight into system dependencies and reliability.

- Seismic Margin Assessment was developed primarily for low-hazard sites (i.e., less than or equal to 0.3g pga). The methodology is not as applicable to higher-hazard sites.

- SMA has limited applicability for addressing safety issues in the future.

- Closure guidelines for SMA have been developed using the EPRI hazard curves which are generally lower than the LLNL curves.

Seismic PRA

Advantages:

- Method is compatible with the PRA which is being performed by the IPE for evaluation of internal accident initiators.

- May be economically attractive if fault trees developed in IPE can be used for equipment selection, although costs associated with relay evaluation and supplemental plant walkdowns need to be considered. Although it is expected that costs for relay review should be similar for both SMA and SPRA.

- Results in numerical risk assessment values which are relatable to safety goals and which can be used in cost/benefit analyses.

- SPRA is amenable to use on future safety issues; although, depending on extent of modification or evaluation, "pruned" trees may require extensive reconstruction.

Disadvantages:

- The results of the SPRA for most plants are strongly dependent on the seismic ground motion hazard estimates developed by EPRI and the NRC. These hazard estimates are significantly different (the NRC/LLNL estimates are higher) and are a source of ongoing debate and controversy.

- Beyond the plant walkdown and relay evaluations, there is less potential for integrating the A-46 seismic review with a SPRA.

- To date no attempt has been made to date to integrate the two reviews.

- No proceduralized document exists for SPRA, as does the GIP for A-46, and NP-6041 for SMA. Many utilities will have PRA-technical skill developed from the IPE; however, if this is not the case, utility seismic capability engineers cannot be practically trained to perform a PRA in the same time frame as with A-46 or SMA. Also, there are few in-house practitioners of fragility analysis; in most cases, an outside contractor will be needed.

- The SPRA has a potentially greater scope of equipment if the event/fault trees are not carefully "pruned".

On the basis of preliminary data and estimates, it appears that on balance the SMA approach is more compatible with A-46 than SPRA and, when relay evaluation and the need for plant walkdowns are considered, it appears the SMA would be comparable in total cost to A-46 if done separately. If the A-46 and SMA or SPRA reviews are integrated (e.g., an effort is made to select the same equipment and a single walkdown is performed for both the A-46 and SMA or SPRA), the savings will be substantial. It is expected that in general the SMA approach will be somewhat more cost effective (5 to 20%) than the SPRA, although plant-specific factors (such as availability of existing SPRA work, experience personnel, etc.) could increase the PRA cost.

Differences Between A-46, SMA, and SPRA Reviews

The intent of both the SMA and SPRA review programs is to identify seismic outliers which can be improved in a practical and cost effective manner with a significant safety benefit.

The principal differences between A-46/SQUG, SMA, and SPRA reviews are summarized below. Table 5-1 summarizes the technical differences, and Table 5-2 summarizes some of the managerial issues.

SMA Reviews - General. The A-46 and SMA review approaches are similar in that they are walkdown-based and make use of experienced, trained engineers who are expected to use screening techniques and engineering judgment to assess seismic capacities and identify seismic outliers. A key difference is that the SMA review is made for a higher ground motion level than the plant licensing basis. Items which do not pass the review level earthquake in the SMA are identified as outliers for further detailed review, but there is no commitment, nor necessarily a need, to implement corrective action. The SMA is less prescriptive and requires less documentation than the SQUG methodology, but after the initial screening walkdown, the SMA requires more extensive evaluation of outliers to establish the capacity of equipment and structures with a high confidence of a low probability of failure (HCLPF). In addition, the scope of equipment and structures to be evaluated for the SMA is broader than the A-46 review.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Scope of Plants Covered		
An A-46 review applies to plants designed for seismic equipment qualification requirements prior to IEEE 344-75.	Seismic IPE applies to all nuclear power plants. For SMA, plants are divided for analysis into three groups according to relative seismic hazard (i.e., full-scope, focused-scope, and reduced scope reviews).	Seismic IPE applies to all nuclear power plants. Site-specific hazard curves will be used for SPRA reviews.
Seismic Input		
Review based on licensing-basis SSE ground response spectrum. In-structure response spectra may be licensing-basis spectra or new spectra developed and used per GIP.	For focused- and full-scope reviews, median NUREG/CR-0098 response spectrum shape anchored to 0.3 g, or 0.5 g for plants in the western U.S. is used. Development of new in-structure response spectra, including effects of SSI, is encouraged. SMA seismic input is higher than A-46 ground motion. For Reduced-Scope reviews, SSE input based on GIP or licensing basis is used.	EPRI site-specific hazard curves for peak ground acceleration and response spectra should be used. If LLNL hazard curves are used, the absolute risk values will be significantly higher than using EPRI curves. The intent of IPEEE is met by relative results whether the EPRI or LLNL curves are used. The NRC GL 88-02 requests the use of both curves, but does allow use of one set, if the higher set is used. This issue requires resolution if SPRA is used. Ground motion input higher than both A-46 and SMA reviews is considered in SPRA, although the ground motions are weighted by their probability of occurrence.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Scope of Review		
Review limited to safe-shutdown equipment, including active electrical and mechanical equipment, tanks, heat exchangers, and cable trays and conduit raceways.	Review includes same types of safe shutdown equipment as for A-46 review, as well as passive mechanical and NSSS equipment, piping, containment, and structures. In addition, a small break LOCA is postulated to occur and soil failure modes are considered. Potential for earthquake-induced flooding/fire is also considered, as well as nonseismic failures and human actions.	Event trees and fault trees are developed for the seismic PRA using the IPE internal event/fault trees. Structures and elements whose failure could impact and fail safety-related elements are added to the trees. Generic letter requests evaluation of nonseismic failures and human actions.
Selection of Equipment		
Safe shutdown equipment may be in single safe shutdown path with redundancy for all active components, or in two separate safe shutdown paths. Single active failures are assumed.	Two separate and independent shutdown paths are selected using EPRI NP-6041 approach. Elements whose failure could lead to core damage are considered initially in NRC NUREG/CR-4334 approach. (Fault trees are "pruned" based on systems and fragility considerations.) Overlap with A-46 equipment list can be achieved by coordinating reviews.	Elements whose failure could lead to core damage are considered. (Fault trees are "pruned" based on systems and fragility considerations.) Overlap with A-46 equipment list can be achieved by coordinating reviews.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Required Experience and Training of Engineers		
A team of trained experienced seismic capability and systems engineers will perform A-46 reviews. A training course is required as part of SQUG program.	SMA will be performed by trained experienced seismic capability and systems engineers. An add-on training course will be given to cover areas of different between A-46 and SMA reviews.	SPRA will be performed by experienced systems and seismic capability engineers. PRA skills from IPE will be transferable, but fragility analyses will most likely require an outside contractor. The add-on course to A-46 training will address only general differences between A-46 and SPRA, and the supplemental walkdown and relay evaluation requirements for SPRA. Detailed SPRA methodology (e.g., event/fault tree methods, fragility analysis) will not be taught.
Screening Requirements		
Equipment functionality is investigated based on seismic experience or test data. Equipment anchorage is analyzed for each component.	The screening criteria in the GIP are recommended for equipment common to both A-46 and SMA programs (see EPRI Implementation Report). Requirements in EPRI NP-6041 are used for elements common only to the SMA structures/equipment list. Caveats and guidance are provided in SMA screening criteria tables for three ranges of seismic input.	The screening criteria given in EPRI NP-6041 can be used to screen elements in a SPRA. It is assumed that screened out elements are not significant contributors to core damage. This assumption can be verified easily by bounding analysis.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Walkdown Procedure		
Walkdown procedures are documented in the GIP. Principal elements of the walkdown are: • Seismic capacity versus seismic demand; • Caveats based on earthquake experience and generic testing data base; • Anchorage adequacy; and • Seismic-spacial interaction with nearby equipment, systems, and structures.	Walkdown using the GIP is recommended for equipment common to both programs, with a supplemental walkdown sheet for SMA requirements. EPRI NP-6041 is used for elements not common to both program (i.e., piping, structures, passive mechanical and NSSS equipment, and containment). Elements not screened out are identified as outliers for further review. Potential for earthquake-induced flooding is considered in SMA walkdown.	Walkdown procedures for SPRA will follow the requirements for SMA in EPRI NP-6041. The level of required detail and effort is the same as for SMA per NP-6041. Field data required for both SMA and SPRA reviews is essentially the same.
Evaluation of Outliers		
Evaluation of outliers follows requirements in GIP and procedures in plant Licensing basis.. The details for resolving outliers is beyond the scope of the GIP. It is the responsibility of the utility to resolve outliers using their existing engineering procedures as they would resolve any other seismic concern.	For focused- and full-scope reviews evaluation of outliers follows EPRI NP-6041. Factors of safety for development of HCLPFs are generally more realistic than for A-46 review. For Reduced-scope review evaluation follows GIP and/or FSAR requirements. Performing A-46 and SMA calculations at same time will minimize cost.	For elements not screened out during walkdown fragility parameter values (i.e., median capacities and combined logarithmic standard deviations) are calculated. Performing A-46 and SPRA calculations at same time will minimize cost.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Relay Review		
Detailed review per GIP is required of those relays needed for successful operation of safe shutdown equipment and support systems, and those relays whose inadvertent actuation due to chatter could result in an unacceptable event. Low ruggedness relays which have demonstrated low resistance to impact and whose chatter could prevent successful operation of safe shutdown systems or cause an unacceptable event are identified for corrective action.	It is recommended that plants that perform a full-scope SMA only conduct a review for low-ruggedness relays. However, plants performing an A-46 review should conduct a relay review according to the procedures in the GIP for all relays within the scope of the A-46 review. Investigation of relays outside the A-46 review but within the scope of the IPEEE review should address only the issue of low-ruggedness relays. For A-46 plants in reduced-scope category, the GIP relay evaluation scope is performed. If low seismic-ruggedness relays (bad actors) are found, perform bad actor review outside A-46 but within IPEEE. For non-A-46 sites in reduced-scope category, only location and evaluation of bad actors is requested. Relays are not investigated for plants in low seismic hazard areas which perform reduced-scope reviews, except for A-46 plants, which will require relay evaluation per the GIP.	The scope of relay chatter review should be consistent with the site's SMA review level. An SMA type review could be performed in lieu of embedding relay chatter investigation into the SPRA event/fault trees.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Containment Review		
Containment review is not required in A-46 review.	For focused- and full-scope SMAs, review for containment integrity, isolation, and prevention of bypass will be conducted. Scope is based on internal events IPE PRA. Note that with the SME equal to 0.3 g pga, the review of concrete and certain steel containments is minimal. Containment review is not conducted in reduced-scope review.	Containment evaluation is required consistent with SMA-type review. The SPRA for IPEEE relies substantially on IPE PRA; i.e., overlay IPEEE seismic hazard onto IPE PRA and report sequences which are negatively impacted.
Quality Assurance		
A-46 review includes some construction checks (e.g., equipment anchorage, cable tray/conduit raceways, interaction hazards, and mounting of internal elements).	SMA assumes that plant is constructed according to design. Also, for older plants it is assumed that either A-46 review has been conducted, or is being conducted concurrently.	SPRA assumes that plant is constructed according to design. Also, for older plants it is assumed that either A-46 review has been conducted, or is being conducted concurrently.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Products		
Specific plant review information is produced per GIP and includes: • Description of safe shutdown path(s); • Lists of equipment on composite, seismic review, and relay review SSELs; • Description of SSE used in A-46 program; • Qualification of review personnel; • Results of the screening verification and walkdown for equipment; and • Summary of main steps in plant operating procedures to bring plant to safe-shutdown condition.	The results of a SMA using the procedure in NP-6041 include: • General plant description and seismic design basis; • Seismic margin earthquake and development of demand on elements; • Seismic margin evaluation (approach, screening criteria, systems description, review team, walkdown, and shutdown path selection); • Assessment of elements not screened out (structures, equipment, and soils); • HCLPFs for components that are less than RLE; and • Results of evaluation and insights gained.	The results of a SPRA include the following: • Description of methodology and key assumptions; • Hazard curve(s); • Walkdown team, procedures, and findings; • Systems information (event/fault trees, nonseismic failures, and dependencies); • List of fragility parameter values; • Mean core damage frequency and ranking of accident sequences, systems, and structures/equipment which are significant contributors); • Seismic-induced containment failures and other containment performance insights; and • Results of evaluation and insights gained.

Table 5-1

COMPARISON OF TECHNICAL REQUIREMENTS BETWEEN A-46, SMA AND SPRA REVIEWS

A-46	SMA	SPRA
Documentation Requirements		
Requirements for documentation are given in the GIP, including SSEL Relay Evaluation and Seismic Evaluation Reports. A final completion letter is required.	Requirements for documentation are given in EPRI NP-6041, which includes general plant description, plant seismic design basis, development of SME demand, SME evaluation, assessment of elements not screened out, and summary and conclusions. Less documentation is required than for A-46 review.	The amount of documentation for a SPRA will be similar to SMA.
Closure Process		
Elements not meeting A-46 criteria (i.e., GIP) are corrected; or if safety benefit is not sufficient, are reported as not warranting corrective action. Back-fit provisions of 10 CFR 50.109 apply for these cases.	HCLPF results below SME can be resolved using NUMARC Severe Accident Issue Closure Guidelines (34), which would include cost/benefit of raising plant HCLPF.	Numerical results are relatable to CDF goals and are resolved using NUMARC Severe Accident Issue Closure Guidelines, which would include cost/benefit analysis of proposed risk reduction opportunities. Use of EPRI vs. LLNL hazard curves would need resolution.

Table 5-2

COMPARISON OF MANAGERIAL ISSUES BETWEEN SMA AND SPRA REVIEWS

SMA	SPRA
Regulatory Acceptability	
The EPRI NP-6041 methodology has been used in two trial reviews, and the NRC NUREG/CR-4334 methodology has been used in one trial review. The results of all three reviews have been accepted by the NRC.	SPRAs have been conducted for over 30 plants. Seven SPRAs have been submitted to NRC as nonlicensing analyses, and one submitted as a licensing condition. All have received favorable evaluations. There is greater potential for argument (compared to SMA) if absolute results are stressed. However, NRC states relative results will be emphasized.
Compatibility With IPE	
SMA is not directly compatible with IPE risk results; however, a risk goal-based closure procedure is given in NUMARC Severe Accident Issue Closure Guidelines when SMA is performed (34).	SPRAs are compatible with PRAs conducted for internal event and other external event accident initiators (e.g., fire and wind). Closure procedures are given in NUMARC Severe Accident Issue Closure Guidelines (34).
Compliance With IPEEE Requirements	
The SMA is an acceptable methodology for IPEEE, although NRC Generic Letter 88-02 does not explicitly state that SMA conforms with the general requirements: • Find risk from seismic events; • Develop an appreciation of severe accident behavior; • Understand the most likely core melt sequence and gain a qualitative understanding of core melt probability; and • Reduce probabilities.	SPRA complies directly with requests in NRC's Draft Supplement 4 to Generic Letter 88-20.

Table 5-2

COMPARISON OF MANAGERIAL ISSUES BETWEEN SMA AND SPRA REVIEWS

SMA	SPRA
Ease of Use by Utilities	
Procedures are well documented in EPRI NP-6041. Add-on training course will provide explanation of differences with A-46 review so work can be performed by utility personnel.	Procedures for performing SPRA are documented in NUREGs CR-2300, 2815, and 4840, and past SPRA submittal to NRC. PRA technical skills from IPE will transfer to SPRA at many plants. However, supporting fragility analysis skills are limited to few practitioners. It is likely that outside contractors will be used to perform fragility analyses. As a minimum, utility personnel will participate in walkdown and relay evaluation. Add-on training courses will address these aspects.
Relative Cost	
Cost is minimized if walkdowns for A-46 and SMA are conducted simultaneously. Costs can be further decreased if the A-46 and SMA evaluation of outliers is performed at same time by same engineers.	Cost is minimized if walkdowns for A-46 and SPRA are conducted simultaneously. Costs can be further decreased if the A-46 and SPRA evaluation of outliers is performed at same time by same engineers. Cost for SPRA likely will be greater (i.e., 5 to 20%) compared to SMA. Additional systems work is required (see Compliance with IPEEE Requirements above). Also, calculation of logarithmic standard deviations will require some extra work. In general, calculation of HCLPFs for SMA and median capacities for SPRA are comparable. However, in some cases calculation of median capacities will require some added effort.

Table 5-2

COMPARISON OF MANAGERIAL ISSUES BETWEEN SMA AND SPRA REVIEWS

SMA	SPRA
Ease of Use by Utilities	
Procedures are well documented in EPRI NP-6041. Add-on training course will provide explanation of differences with A-46 review so work can be performed by utility personnel.	Procedures for performing SPRA are documented in NUREGs CR-2300, 2815, and 4840, and past SPRA submittal to NRC. PRA technical skills from IPE will transfer to SPRA at many plants. However, supporting fragility analysis skills are limited to few practitioners. It is likely that outside contractors will be used to perform fragility analyses. As a minimum, utility personnel will participate in walkdown and relay evaluation. Add-on training courses will address these aspects.
Relative Cost	
Cost is minimized if walkdowns for A-46 and SMA are conducted simultaneously. Costs can be further decreased if the A-46 and SMA evaluation of outliers is performed at same time by same engineers.	Cost is minimized if walkdowns for A-46 and SPRA are conducted simultaneously. Costs can be further decreased if the A-46 and SPRA evaluation of outliers is performed at same time by same engineers. Cost for SPRA likely will be greater (i.e., 5 to 20%) compared to SMA. Additional systems work is required (see Compliance with IPEEE Requirements above). Also, calculation of logarithmic standard deviations will require some extra work. In general, calculation of HCLPFs for SMA and median capacities for SPRA are comparable. However, in some cases calculation of median capacities will require some added effort.

SPRA Reviews - General. The A-46 and SPRA review approaches are similar mainly in the areas of walkdown screening and evaluation of relays. The NRC Generic Letter requests that the walkdown and relay evaluations be done in accordance with SMA procedures; therefore, either method will result in the same review for these issues. The key difference is that a probabilistic approach requires initial consideration of all equipment which could lead to core damage; thus, a broader scope of equipment must be reviewed for SPRA than for A-46 and possibly for SMA. Also, site-specific hazard curves are typically at a higher level ground motion than either the A-46 or SMA review. SPRA considers the probability of earthquake, which may offset the effect of the higher magnitude, i.e., larger earthquakes have lower probability of occurrence. The bulk of the SPRA evaluation will need to be performed by experienced systems and seismic capability engineers with PRA expertise who can perform fragility analysis. While many utilities will have PRA skills developed from IPE, the seismic capability engineers will most likely be outside contractors, especially for the fragility analysis. Utility personnel will participate, as a minimum, in at least the walkdown and relay evaluation.

Scope of Plants. Unlike A-46, seismic IPEEE applies to all nuclear power plants. For SMA, plants are divided into three groups according to relative seismic hazard: full-, focused-, and reduced-scope. SPRA applies to all plants, but there is no grouping by relative seismic hazard, except for the requirements for relay chatter, containment and soil failure evaluation.

Seismic Review Level. The A-46 review is based on the plant's licensing basis SSE ground motion spectra and corrective action is required for those items which do not meet the A-46 criteria at this level, unless the utility invokes the backfit provisions of 10 CFR 50.109.

For focused- and full-scope reviews, the SMA review level is higher than the SSE (0.3g PGA together with a NUREG/CR-0098 median-centered spectra for most plants). For reduced-scope reviews, SSE input, based on licensing basis, is used.

For the SPRA review, site-specific hazard curves for peak ground acceleration and response spectra are used. Ground motion covers the range of accelerations from zero to maximum physical values which are higher for SPRA than in either A-46 or SMA. However, in a PRA the accelerations are weighted by their probability of occurrence.

Governing Criteria . The A-46 and SMA reviews are directed primarily at assuring

safe shutdown of the plant following an earthquake. The A-46 review is generally limited to safe shutdown equipment; the SMA includes piping, containment and structures as well. In both cases, the safe shutdown equipment is that equipment needed to achieve and maintain safe shutdown for 72 hours following an earthquake. In the A-46 review, concurrent LOCAs are not postulated to occur; in the SMA, a small break LOCA (SBLOCA) is postulated to occur. The primary impact of these differences is the addition of SBLOCA mitigation systems (i.e., high pressure make-up capability) to the safe shutdown equipment to be reviewed.

Scope of Review

The safe shutdown equipment included in the A-46 review consists of active electrical and mechanical equipment, tanks and heat exchangers needed for safe shutdown and cable tray and conduit raceways. The SMA scope includes these components and the following additional areas that are on the safe shutdown paths:

- Nuclear steam supply system components
- Containment systems (those which affect early containment failure)
- Piping
- Category 1 civil structures
- Soil failure mechanisms

In the case of the safe shutdown equipment selection, the A-46 and SMA rules are somewhat different. Both programs require redundant safe shutdown equipment and the systems necessary to support the primary safe shutdown systems. However, in the A-46 program the safe shutdown equipment may be in a single safe shutdown path with redundancy for all active components, or alternatively, in two separate safe shutdown paths. The SMA requires that the safe shutdown equipment be based on two separate and independent paths. In addition, the A-46 rules require assumption of single active failures, while the SMA methodology does not. However, in the SMA method, paths are chosen based on a screening criterion applied to nonseismic failures (e.g., battery depletion, PORV failure) and human actions (e.g., delays or failures in performing specified actions). These differences could lead to significantly different safe shutdown equipment lists. However, initial selection of the safe shutdown paths with both sets of rules in mind can result in significant overlap of the required active safe shutdown and support equipment, as was demonstrated in the Plant Hatch A-46/SMA review (30).

For SPRA, structures, components and systems whose failure could lead to core damage are considered initially. Event trees and fault trees are developed for the SPRA using the IPE internal event/fault trees. Structures, components and systems whose failure due to a seismic event could impact and fail safety-related elements are added to the trees. Nonseismic failures and human actions are to be included, unless shown to be insignificant, as in the SMA approach. Fault trees are "pruned" based on systems and fragility considerations. Overlap with the A-46 equipment list can also be accomplished by coordinating reviews. The extent of overlap which can be achieved is believed to be significant, but little direct experience is available for performing this conditional review.

Review Methodology. Some key differences in review methodology between the A-46 and IPEEE are as follows:

- The SMA and SPRA approaches assume the plant is constructed according to design; the A-46 review includes some installation checks.

- The factors of safety for equipment anchorages are generally more conservative in the A-46 program, i.e., the A-46 program uses a factor of safety of 3, the SMA uses factors of safety ranging from 2 to over 3 in some cases. SPRA doesn't explicitly use factors of safety. The variability in median capacity reflects a range of factors of safety.

- The methods for evaluation of ground mounted storage tanks are more realistic in the IPEEE review compared to the more conservative A-46 criteria, e.g., water hold-down forces are included in SMA but not A-46 evaluation.

- An extensive equipment experience data base is utilized in the screening of outliers in the SMA. This process results in estimates of equipment capacity which are intended to provide approximately 95% confidence that the probability of failure does not exceed about 5% at the HCLPF capacity level. SPRA develops median capacities and logarithmic standard deviations for variability. The A-46 or SMA capacity review is pass/fail while in SPRA there is flexibility for a graded response with the possibility of no action being recommended.

- The A-46 and IPEEE seismic reviews both include evaluation of seismic spatial interactions. For IPEEE the potential for earthquake-induced pipe ruptures and, if necessary, effects of possible flooding, are also reviewed.

- Review methodologies are included in the SMA for soil structure interaction, soil failure evaluation, piping assessment, containment systems review and evaluation of civil structures, consistent with the required level of review. SPRA includes similar added scope as SMA.

- The scope of containment reviews for both SMA and SPRA are based on the IPE internal events. However, the IPEEE SPRA relies substantially on this IPE evaluation, with the overlaying of the IPE PRA with the IPEEE seismic hazard, and reporting of those sequences which are negatively impacted.

Relay Evaluation. The A-46 methodology requires a detailed review of those relays which are necessary for functioning of safe shutdown equipment and support systems and those relays whose inadvertent actuation due to chatter could result in an unacceptable event. This relay evaluation process is defined in the GIP and includes a two-pronged approach involving review of the effect of relay chatter on system function and seismic adequacy review of those relays whose function is essential. As part of this process, essential relays which have low fragility and/or have demonstrated low resistance to impacts are separately identified for consideration of corrective action.

In the SMA review approach, the extent of relay evaluation is dependent on the earthquake review level. Those plants in the full-scope category will require a detailed review of low seismic ruggedness relays. Plants which are identified as low seismic plants will not require a relay seismic evaluation. The remainder of the plants, consisting of the large majority of plants in the U.S., are in the focused-scope category. For A-46 plants in the focused scope category, the GIP relay evaluation scope and procedure is used. If low seismic-ruggedness relays (bad-actor list) are discovered during the A-46 review, the relay reviews should be expanded outside the scope of A-46 but within the scope of IPEEE. For non-A-46 plants in the focused-scope category, the Generic Letter requests only the location and evaluation of low seismic-ruggedness relays (bad-actor list).

For an SPRA, the Generic Letter requests the scope of relay chatter review be consistent with the site SMA review level as described above. The same arguments as used in A-46 and SMA reviews can be used to screen relays from the analysis (i.e., operator recovery and/or high seismic capacity). Fragility curves are developed for relays which may be significant contributors to core damage. Both the potential for structural failure and operator failure to reset are considered as in an SMA review. An SMA type review could be performed in lieu of embedding relay chatter investigation into the SPRA fault trees.

Documentation and Quality Assurance Requirements. The A-46 review, as prescribed in the GIP (21) and required by the NRC's Safety Evaluation Reports, requires significantly more documentation and QA coverage than does the SMA or SPRA.

Specifically, the significant inspections, reviews and checks required by the A-46 process are included on checklists and on summary verification sheets which formally document the acceptability of each component on the safe shutdown equipment list. Personnel qualifications and training requirements are formalized. In addition, any corrective actions taken in the A-46 review which result in changes to plant licensing bases, plant procedures or plant hardware are required to be accomplished in accordance with the plant's formal 10 CFR 50 Appendix B QA program. The SMA or SPRA, on the other hand, are performed pursuant to an NRC request for information, are not related to the plant's licensing basis and accordingly, are not safety-related review programs. Any plant procedure or hardware modifications which result from the SMA or SPRA could be subject to each plant's QA program, independent of the reasons for such changes, although this would be at the discretion of the licensee.

Regulatory Acceptability. Two trial reviews using EPRI SMA methodology and one using NRC SMA methodology have been performed, and the results have been acceptable. Over 30 SPRAs have been performed--eight of which have been submitted to the NRC; all eight received favorable evaluations. There is greater potential for arguments with the SPRA if absolute results are stressed, but Supplement 4 to NRC Generic Letter 88-20 states that relative results will be emphasized.

Ease of Use by Utility. GIP methodology for A-46 evaluation is extensively proceduralized. SMA procedures are well documented in NP-6041. Training for SMA similar to that for A-46 will be provided to all interested utilities and should allow utility personnel to perform SMA reviews. Utilities may have some in-house PRA skills from the IPE; however, the fragility analyses will almost certainly need to be performed by an outside contractor. Training of utility engineers in SPRA would not be practical in the necessary time frame, although the walkdown and relay evaluation in the SMA training program would be applicable. As a minimum utility personnel will be participating in the walkdown and relay evaluation.

Future Use. SQUG/A-46 methodology can be used for new and replacement parts. SPRA can be used as a "living PRA" to address future changes and other safety issues although "pruned fault trees" may have to be reconstructed to consider certain specific systems and equipment. SMA is a single time point method; however, it would be relatively straightforward to apply the method when future plant modifications are planned.

Relative Cost. Cost can be minimized by performing walkdowns and outlier

evaluations for A-46 and either SPRA or SMA simultaneously. Overall cost for SPRA will likely be about 5% to 20% higher than SMA, based on additional systems work and potentially a longer list of equipment to be reviewed, although plant-specific factors (e.g., existing SPRA work, experience of personnel) could reduce the SMA cost. Calculation of HCLPF for SMA and median capacities for SPRA are generally comparable.

RECOMMENDATIONS

It is recommended that IPEEE and A-46 reviews be conducted concurrently and that the review tasks be combined, whenever possible. At the utility's option, the IPEEE and A-46 reviews can be conducted independently. However, this would require duplication of some work, which can be avoided if the reviews are coordinated. As a minimum the studies should be conducted at the same time, or the A-46 review performed first. This is necessary since the IPEEE review assumes that quality-related issues have been investigated and resolved. Since it is acceptable to perform the reviews independently, the SRT can always directly use the corresponding criteria for each review. Guidance for integration of IPEEE deterministic and A-46 reviews is given in Appendix C.

Combined Deterministic Reviews

Most of the equipment and components included in the A-46 and IPEEE deterministic programs should be the same; however, there may be some differences. For example, the philosophy for providing redundancy for the primary success path is different for the two programs. This may lead to the selection of different components.

For plants performing combined reviews, the walkdown requirements should follow the guidance given in the GIP (21). The GIP procedure will be generally more conservative for the 0.3 g pga RLE plants compared to the requirements given in EPRI Report NP-6041 for a SMA review, for those components also included in the IPEEE review. However, this approach should result in a cost-effective review. The SRT can always revert to the requirements in Reference (11) for components in the IPEEE assessment, if the requirements conflict with the A-46 review. For components outside the scope of an A-46 review, EPRI Report NP-6041 should be used for screening.

Equipment which is screened out during the plant walkdown is not considered further in the review. For the full-scope SMA and to a limited extent, the focused-scope SMA, HCLPF values should be determined for outliers. For the

components common to both programs, an evaluation to the design basis must be performed to satisfy the commitments in the A-46 program. For the reduced-scope assessment, components outside of the GIP should be analyzed based on the FSAR commitments. It is recommended that common calculations be performed which cover both programs, wherever possible, in order to minimize the work.

In documenting the evaluations, separate reports should be prepared for the IPEEE and A-46 programs, consistent with the reporting guidance of each program.

Section 6

DESCRIPTION OF CLOSURE PROCESS

BACKGROUND ON CLOSURE OF SEVERE-ACCIDENT ISSUES FOR SEISMIC EVENTS

This section describes guidelines on the appropriate use of seismic-IPE review approaches to assist licensees in formulating effective decisions for achieving closure on seismic severe-accident issues. These closure guidelines are substantially consistent with the guidelines and framework recommended by industry for resolving severe-accident issues for internal events (34), yet they emphasize the use of deterministic review procedures and they reflect important differences in NRC guidance for treatment of external versus internal severe-accident initiators. The seismic closure guidelines described in this section are included as part of an overall industry document for severe-accident-issues closure [see (34)]. That overall document describes how industry's severe-accident closure process and framework satisfy all six elements (see Section 1 of this report) of the NRC's integrated closure plan (2). The IPE plays a central role in the integration plan and in industry's closure process.

Probabilistic risk assessment, level-1 in scope (with enhancements for containment evaluation), is the format recommended by industry and the NRC for performing the internal-events IPE. The internal-events closure guidelines, therefore, are consistent with Level-1 PRA results; i.e., core-damage-frequency criteria and major accident-sequence groups are the basic elements of these guidelines. In contrast, as discussed in Sections 3 and 4, the recommended formats for performing the seismic IPE include: (1) the reduced-scope assessment, (2) the focused-scope SMA, (3) the full-scope SMA, and (4) the SPRA. The closure guidelines for seismic events must, therefore, support the use of deterministic (SMA) approaches in addition to the probabilistic (PRA) approach. In the EPRI SMA approach, success-path logic diagrams (SPLDs) are constructed to convey the various combinations of component or operator actions that lead to a long-term safe-shutdown condition, given a seismic margin earthquake (SME). To make consistent use of industry's internal-events closure guidelines, results in the SMA format associated with particular SPLDs must be related to major core-damage-sequence group frequencies.

OVERVIEW OF CLOSURE APPROACHES

The critical action in the seismic IPE, regardless of the specific implementation approach taken, is to walkdown the plant (reviewing the safe shutdown systems and equipment) and to evaluate elements identified as outliers (for instance, using the SMA screening tables in EPRI Report NP-6041, Rev. 1) or identified as dominant risk contributors.

In the reduced-scope assessment, the outliers identified during the review are evaluated using the plant licensing basis (FSAR) or the Generic Implementation Procedure (GIP) guidelines that were developed as part of Unresolved Safety Issue A-46, *Seismic Qualification of Equipment in Operating Plants*. The closure guidelines for the reduced-scope plants are described in the next subsection.

For the focused- and full-scope SMA plants, outliers (i.e., the elements that do not pass the SMA screening tables at the SME) are evaluated to estimate their HCLPF capacities. As mentioned above, to develop a seismic-IPE closure approach for these plants that is consistent with the internal-events closure guidelines, target HCLPF capacity values for each success path can be related to major seismic core-damage-sequence group frequencies. The key element in establishing this relation is the determination of appropriate plant-specific review-level ground motions (RLGMs). A closure RLGM is the plant-specific target HCLPF capacity (for a success path with a given functional plant state) that will satisfy a specified core-damage frequency criterion associated with the particular major functional state or sequence group. The procedure for obtaining plant-specific RLGMs is described in Appendix E [also see Reference (24)]. Three RLGMS, denoted RLGM-A, RLGM-B and RLGM-C, are determined corresponding to three different core damage frequency based closure criteria. RLGM results for 58 central and eastern United States plants are presented in Reference (35). (Table E-1 presents RLGM-PGA values obtained by scaling the 5%-damped NUREG/CR-0098 median spectral shape to just envelope the site-specific RLGM spectrum over the vibration frequency range of 2 to 10 Hz. These PGA values may be used as conservative surrogates to the RLGM spectra, and provide a simpler basis for comparison with HCLPF capacity results). The closure guidelines for the focused- and full-scope review categories are given in the subsection following the reduced-scope closure guidelines.

In a SMA review, two alternate success paths are chosen for distinct functional SPLDs. Each alternative success path should involve substantially different components and different functional-sequence conditions. The motivation for developing two alternative success paths is to demonstrate redundancy; it is

therefore important to evaluate each of the two success paths against the closure guidelines.

The success path can be used as a conservative surrogate to a functional accident sequence; hence, closure guidelines defined in terms of functional accident sequences may be applied to the success-path sequences. Failure along any success path will be dominated by the component having the lowest HCLPF capacity. So, instead of evaluating the success path as a complete sequence, components on the success path are treated individually. If the HCLPF capacity of each component on a given success path exceeds the guidelines based on a RLGM, then the corresponding guidelines in terms of functional accident sequence frequency are likewise demonstrably satisfied.

This understanding allows closure guidelines already established in terms of accident sequence groups for the internal-events IPE evaluation to be used in terms of component HCLPF comparisons for focused- and full-scope seismic IPE evaluation, to achieve a substantially consistent development. If a SPRA is performed, the resulting core-damage frequency is treated in terms of core-damage sequence groups, consistent with the PRA-based closure guidelines for internal events (34). The closure guidelines for SPRA review are presented following those for the SMA approaches.

CLOSURE GUIDELINES FOR REDUCED-SCOPE IMPLEMENTATION

If the seismic IPE is conducted using reduced-scope methodology, the closure evaluation process consists of the following steps:

- Delineate two alternate success paths and define their major functional states (Section 3);
- Develop a list of screened-in outliers using the SMA screening tables (Section 3); and
- Evaluate components for compliance with licensing commitments (FSAR) or with the GIP guidelines based on earthquake experience qualification (Figure 6-1).

If the FSAR commitment is satisfied for a particular component, then closure is reached with respect to that component. The GIP guidelines can be used in lieu of a direct FSAR-consistent evaluation for plants covered under the A-46 program (Appendix C). Figure 6-1 describes, in flowchart form, the framework for closure evaluation when a reduced-scope assessment is conducted.

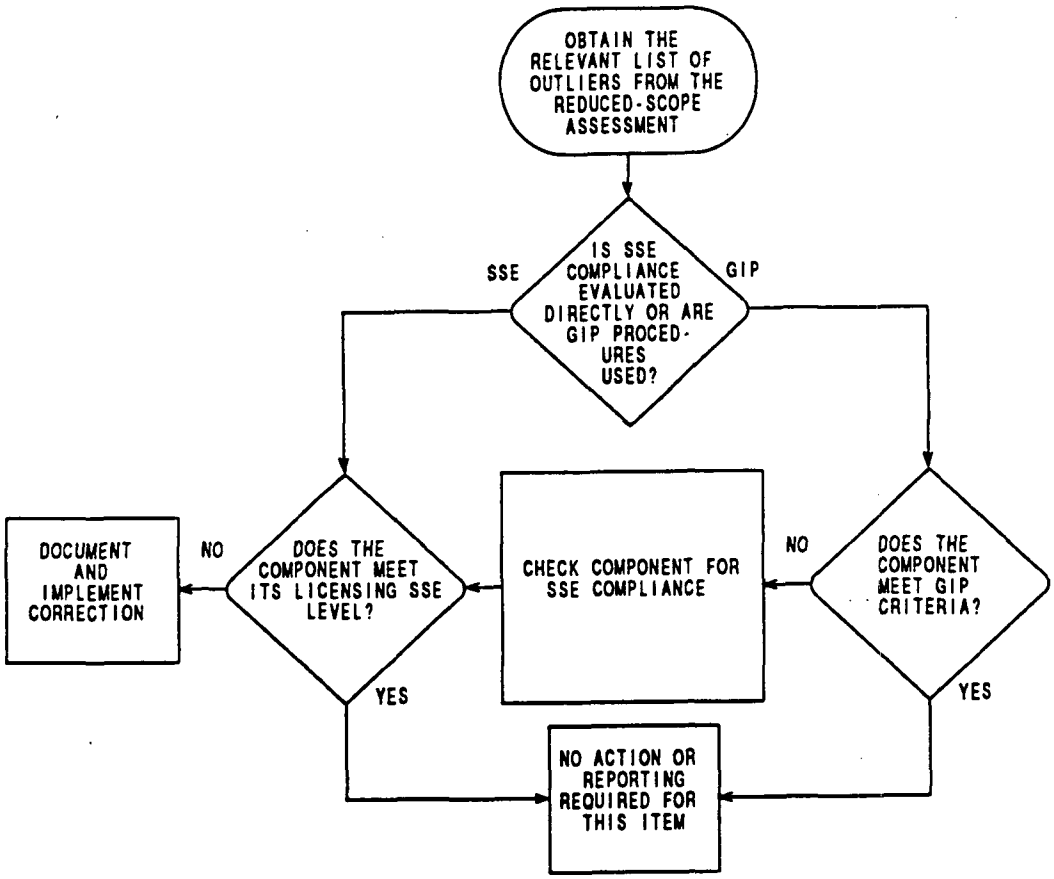


Figure 6-1. Closure process recommended for reduced-scope seismic-IPE evaluation.

The seismic-IPE-related closure process is completed when every reduced-scope SMA outlier has been evaluated, and all appropriate actions (if any) have been determined, documented, and scheduled for implementation.

CLOSURE GUIDELINES FOR FOCUSED-SCOPE AND FULL-SCOPE IMPLEMENTATIONS

If the seismic IPE is conducted using focused- or full-scope methodology, the closure evaluation process consists of the following steps:

- Delineate two alternate success paths and define their major functional states (Section 3);
- Develop a list of screened-in outliers using the SMA screening tables (Section 3);
- Calculate HCLPF capacities for outliers using the NUREG/CR-0098 (5%-damped) median spectral shape to characterize ground-motion input, and develop a list of screened-in remaining outliers (Section 3);

- Obtain RLGM spectra [Reference (35)] or RLGM-PGA values (Table E-1) to be used for evaluation of remaining outliers; and
- Evaluate remaining outliers against closure guidelines by comparing component HCLPFs with RLGMs.

Prior to evaluating success-path elements against RLGM-based closure criteria, HCLPF capacities are computed for the appropriate set of components identified in the seismic IPE, using the 5%-damped NUREG/CR-0098 median spectrum as input, as outlined in Section 3. A check for compliance with the SSE licensing commitment is performed when required (see Figure 6-2), similar to that in the reduced-scope SMA evaluation. Using the guidelines or results specified in Appendix E, three separate RLGMs for any plant (and given damping and soil type) are obtained for use in seismic-IPE closure; these three motions are determined for core-damage-frequency safety targets of 5.0×10^{-5}, 2.0×10^{-5}, and 5.0×10^{-6} per year.

Figure 6-2 describes the complete pre-closure process used to screen-in a list of remaining outliers from the initial list of SMA outliers. Although not specifically called out in the NRC staff guidance, the licensee should be cognizant of the status of the screened-in elements relative to the licensed SSE level. Although the IPEEE is not intended to be a confirmation of the current licensing basis, it is incumbent upon the licensee to assess those conditions in which there is some question as to SSE compliance. Instances of noncompliance would be handled via the applicable plant procedures.

Figure 6-3 describes, in flowchart form, the framework for closure evaluation of core-damage success-path elements when a focused- or full-scope SMA is conducted. With the exception of the RLGM-based criteria in the top row of (triangular-shaped) decision elements of Figure 6-3, this framework is identical to that for IPE core-damage evaluation [see Figure 1, Reference (34)] in internal-events closure. The RLGM-based criteria are themselves developed to be consistent with the corresponding core-damage-frequency related criteria in the internal-events IPE core-damage closure evaluation.

Figure D-1 shows a general extension of success-path elements associated with containment performance. For seismic containment sequences and related success-path elements (see Appendix D), the SPRA database suggests that substantial margin exists to prevent large early release and large containment bypass (24). Figure 6-3 is, therefore, also applied for closure evaluation of success-path elements needed to prevent large-early containment release and large

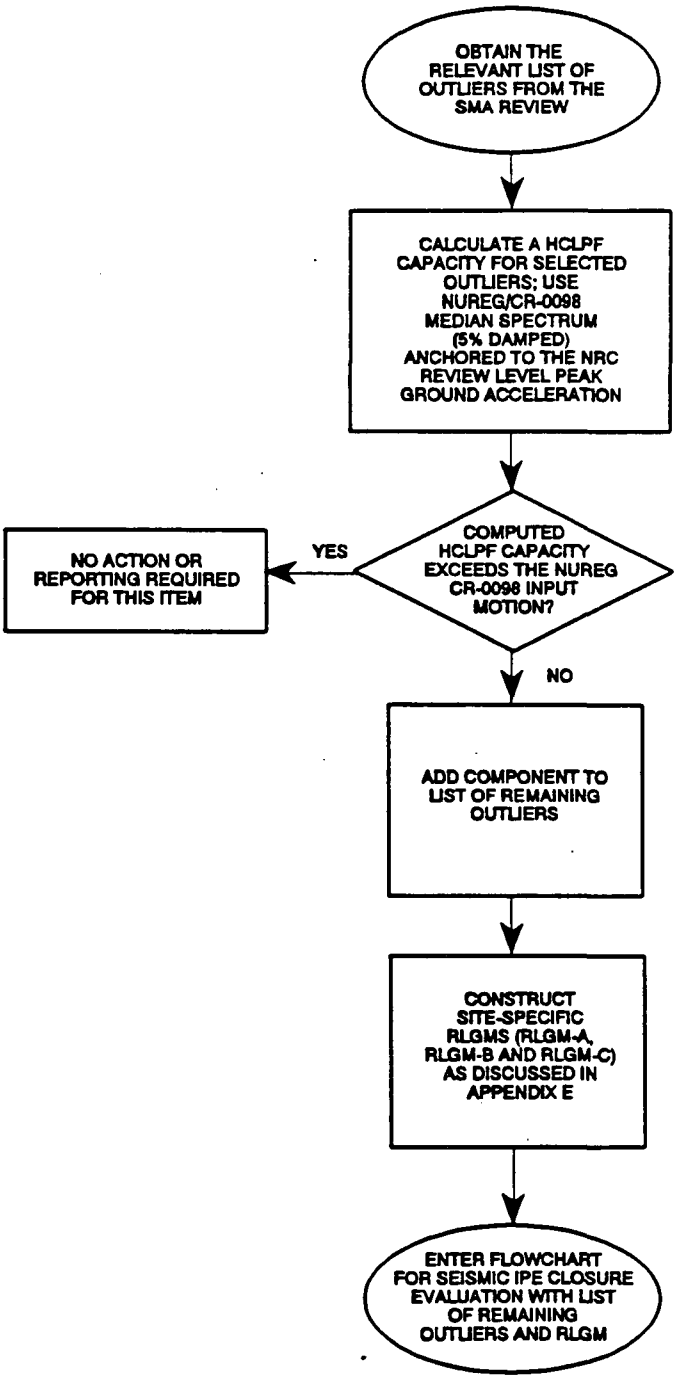
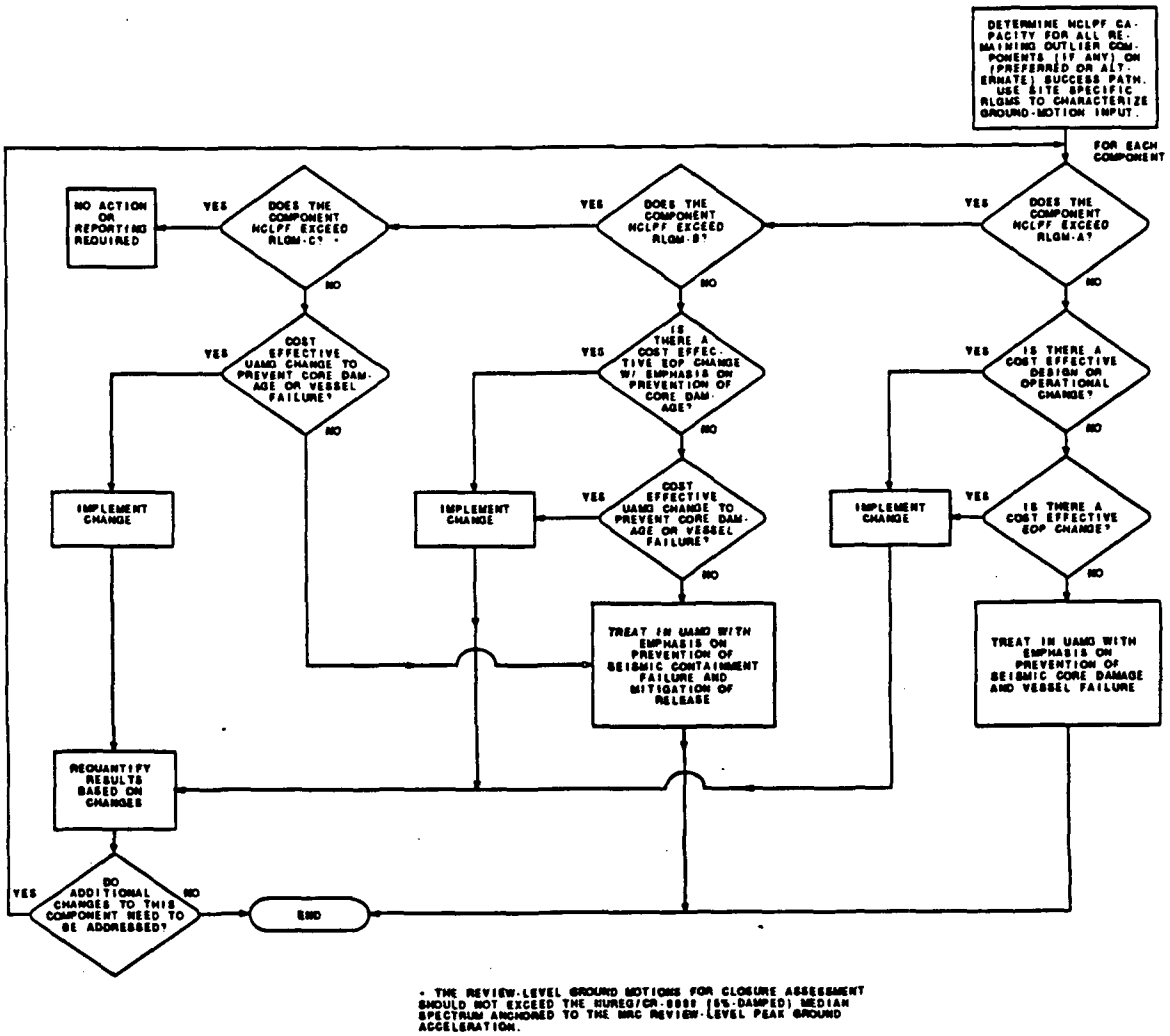


Figure 6-2. Pre-closure assessment for evaluation of outliers in focused-scope and full-scope SMA seismic IPEs.



UAMG: Utility Accident Management Guidelines
EOP: Emergency Operating Procedures

Figure 6-3. Closure process recommended for seismic-IPE core-damage evaluation: focused-scope and full-scope SMA.

containment bypass. Consistent with NRC guidance in Supplement 4 to Generic Letter 88-20 for seismic events [Reference (9)], separate closure guidelines for evaluation of containment-related elements and for evaluation of core-damage elements is not required.

The seismic-IPE-related closure process is completed when all components in the appropriate set of focused- or full-scope remaining SMA outliers have been evaluated and appropriate corrective actions (if any) have been determined, documented, and scheduled for implementation.

CLOSURE GUIDELINES FOR SPRA IMPLEMENTATION

If a SPRA is performed, the seismic core-damage frequency can be treated either as a single core-damage sequence group or as a multiple number of major sequence groups, for evaluation against closure guidelines. If multiple seismic core-damage sequence groups are utilized, a grouping philosophy similar to that intended for the internal-events IPE [see Reference (34)] should be taken in defining seismic accident-sequence groups. In the case of any external event, however, an appropriate set of accident-sequence groups can be simply obtained based on the nature of the sequence induced by the external hazard (e.g., seismic-induced LOCAs, seismic-induced station blackout, etc.). Components important in seismic core-damage sequences that may lead to containment bypass or release should also be considered in the closure process. An example grouping scheme for SPRA sequences is provided in Appendix B of Reference (34).

Once the SPRA sequences have been grouped, the seismic-IPE closure process involves comparing seismic core-damage group frequencies and containment-related sequence frequencies to the internal-events IPE closure guidelines in Tables 1A and 2A, respectively, of Reference (34). Thus, the closure process for SPRA implementation is similar to that for the internal-events IPE implementation.

This seismic-IPE closure process assumes compliance with the plant's licensing commitment. The closure process is completed when the SPRA sequences have been identified and evaluated, and all appropriate corrective actions (if any) have been determined, documented, and scheduled for implementation.

Section 7

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Appendix A

TECHNICAL BASIS AND DESCRIPTION OF APPROACH FOR REVIEW METHOD SELECTION

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CONTENTS

Appendix A

TECHNICAL BASIS AND DESCRIPTION OF APPROACH FOR REVIEW METHOD SELECTION

INTRODUCTION

This appendix describes the recommended approach and its basis for selecting an appropriate (i.e., effective and efficient) review method for performing the seismic IPE. This material supports the more-specific guidelines suggested in Section 2 for review-method selection. The approach is based on considerations of site seismic hazard results, of plant seismic design bases, and on consistent use of probabilistic measures. The selection of review method complements both the execution of specific recommended implementation procedures for seismic evaluation, the actual details of which are discussed in Section 3, and the use of decision criteria for severe accident policy closure, as discussed in Section 6.

The development in this appendix is consistent with the objectives of the IPE (or, more generally, objectives of the severe-accident policy (SAP) statement (1) of the Nuclear Regulatory Commission (NRC)); i.e., the purpose of seismic evaluation is to identify potential plant vulnerabilities to severe accidents that may be initiated by earthquakes.

SEISMIC REVIEW METHODS, HAZARD RESULTS, DESIGN BASES

Development of an approach for selecting an efficient seismic review method for severe-accident evaluation requires knowledge of available seismic review methods, characterizations of seismic hazard, and basic descriptions of plant seismic capacity. We provide below a brief review and description of each of these items.

Seismic Review Methods

The set of common choices available for plant-specific seismic evaluation -- seismic probabilistic risk assessment (SPRA) and seismic margin assessment (SMA) -- is limited, and each choice is generally costly. Out of need for a more cost-effective implementation of severe-accident evaluations, additional seismic review procedures have been designed; these include the *focused-scope* margins assessment and the *reduced-scope* assessment (see Section 3 and References (2, 3). Attributes of the four available alternative review methods are given below:

- Seismic Probabilistic Risk Assessment (3, 4): Broad systems modeling, detailed plant walkdown, fragility quantifications [including analyses for anchorage and (if warranted) relay chatter], and risk quantification.

- Full-Scope Seismic Margin Assessment (5): Limited systems modeling, detailed plant walkdown, anchorage evaluation to seismic margin earthquake, component screening (and identification of potentially vulnerable relays), HCLPF capacity quantifications.

- Focused-Scope Seismic Margin Assessment (Section 3): Limited systems modeling, detailed plant walkdown, anchorage evaluation to seismic margin earthquake, component screening, limited and conservative HCLPF capacity quantifications.

- Reduced-Scope Assessment (Section 3): Limited systems modeling, detailed plant walkdown, anchorage evaluation to licensing commitment, component screening, component evaluation with respect to the SSE.

The common denominator in each of these review methods is a thorough plant walkdown, the task that experts agree is the most important part of conducting an effective seismic evaluation for potential plant vulnerabilities.

If conducted with the same detail in plant walkdowns and component capacity calculations, the SPRA must involve somewhat more cost than the full-scope SMA, due to (among other things) the effort required to more-thoroughly describe plant systems and to quantify risk. SMA methodologies have been developed (5, 6, 7) with the explicit purpose of more efficiently determining plant resistance to ground motions that exceed the seismic design basis; hence, SMA may often be a preferred alternative to SPRA for conducting severe-accident evaluations.

On average, a focused-scope SMA will involve significantly lower cost than a full-scope SMA, due to selectivity (and conservative simplifications) in HCLPF calculations and more-limited relay chatter considerations. Similarly, on average, a reduced-scope assessment will involve significantly lower cost than a focused-scope SMA, due to elimination of HCLPF calculations. (The reduced scope assessment is evaluated to the licensing commitment, and is appropriate for plants where the seismic hazard is very low, such that the design basis can itself be considered a severe-accident level).

By design, this hierarchy of four review choices is interwoven with the development of procedures to select review methods appropriate for severe-accident implementation in seismic IPEs. In other words, an increased set of review

choices was developed (for seismic-IPE considerations) to accommodate the varying array of results found for seismic characterizations important to severe-accident analysis. This common development was designed to encourage efficiency and cost-effectiveness in the review process.

Roughly speaking, in the context of the seismic IPE, the reduced-scope assessment is considered appropriate where the seismic hazard is low; the full-scope SMA is considered appropriate where the seismic hazard is comparatively high relative to the design basis; the SPRA, an alternative to the full-scope SMA, is considered appropriate in special situations where risk results may be anticipated to facilitate decisionmaking; and the focused-scope SMA is considered appropriate for the remaining bulk of plants that have comparatively moderate seismic hazard relative to design basis.

Seismic Hazard Results

Seismic hazard results (in particular, uniform hazard spectra) are also needed to make review choices for severe-accident evaluation. The NRC has expressed the position that both EPRI and LLNL hazard results should be considered in decision-making on seismic review-type selection (i.e., plant binning) (3). The proper intent is to find consistency in the two methods and resulting consistency in identification of potential vulnerabilities or dominant risk contributors. Consistency (among plants) in the two approaches, however, is (generally speaking) found only in the median hazard results. On statistical bases, one expects the median of a distribution to be one of the most stable among distribution parameters; hence, it is not surprising that the greatest [both relative (among plants) and absolute] consistency in the two approaches is found in the medians. The mean seismic hazard results for EPRI and LLNL methodologies, on the other hand, differ substantially in absolute and relative (among plants) comparisons. Because the LLNL mean results are generally substantially greater than the EPRI mean results, equal weighting of these means implies that decisions would be dominated by the LLNL results. The advantages of consistency and more equal decision weighting clearly suggest a preference for use of median seismic hazard results in developing systematic conclusions from the EPRI and LLNL hazard methodologies.

Plant Design Bases

In addition to having hazard information, it is important to also factor available plant-specific information into the seismic IPE review-development process. For

example, if one has a-priori information that suggests a plant has large seismic margin compared to another plant, then (given similar seismic threats) one would have a basis for implementing an alternate set of review procedures of a more limited nature (aimed mostly at confirming the a-priori expectation) for that plant. The individual utility is in the best position to make such judgments concerning the seismic capability of its plant, and may wish to factor its own experience and information-base into the seismic IPE decision process.

Without more detailed knowledge of a plant, however, the best characterization that is conveniently available is the plant design basis, or SSE, spectrum. For a number of reasons, the plant SSE will not correlate extremely well with plant seismic margin (as measured by the plant HCLPF capacity, for example). There are a number of reasons, however, to believe that the HCLPF capacity does at least have an underlying, although non-deterministic, relationship with SSE. The available HCLPF data supports this to some extent; for instance, a correlation coefficient of roughly 0.75 between HCLPF and SSE (in terms of peak ground acceleration) is implied by published SPRA and SMA results. (A summary of plant HCLPF data is provided in Reference (9). In obtaining our results here, we start with that data set, make a modification for one plant to account for more recent fragility data, and add results from recent studies for two additional plants).

Although we do not expect plant SSE to be a very good predictor of the HCLPF, it is valid to conclude that SSE may be a reasonable predictor of the HCLPF. More assuredly, SSE is at least a basis to bound the HCLPF. For instance, HCLPF data derived from past SMA and SPRA studies suggest an average factor of seismic margin (i.e., HCLPF/SSE) for plants of about 1.7, with values ranging from 1.2 to above 2.0. If the SSE sufficiently establishes a lower bound on plant HCLPF, and the lower bound qualifies as a severe-accident event (considering the level of seismic hazard), then there is no compelling need for detailed quantifications to determine the HCLPF precisely. For explicit consideration of severe-accident behavior, the seismic-IPE itself is undertaken (in any case) and is critical to identifying plant-specific weak-links (i.e., anchorage, bracing, interaction, etc., problems). The extra effort required to precisely quantify the HCLPF capacity, however, will likely not provide additional insights for improving plant safety, and thus may have limited merit in a severe-accident evaluation.

The currently available data appear to be adequate to support a rough bounding relationship of HCLPF with SSE. Consequently, factoring design basis into the review-method selection process is considered appropriate.

The SSE spectrum is only one aspect of the plant seismic design basis. For instance, load factors, load combinations, damping levels, etc., are all elements of design requirements which may be different from plant to plant. To the extent these factors can be included in characterizing the seismic capability of a plant, refinements in severe-accident review selection can be made. It is often impractical, however, to consider anything other than the SSE spectrum and perhaps the damping value used in the design analyses.

APPROACH FOR SELECTING SEISMIC REVIEW METHOD

Basis of Approach

Following is discussed details of the approach for selecting a review method (among the four available methods described earlier) for the seismic IPE. The approach checks for consistency (appropriateness) in design-basis hazard, i.e., probabilities of exceeding a plant's design-basis spectrum, as compared to design-basis hazard for other plants (including those designed to modern criteria).

In addressing severe-accident issues, the approach relies implicitly on well-founded conclusions derived from earthquake engineering experience. First, the seismic experience database (10) reveals that components in industrial facilities (of the type found in existing nuclear power plants) are generally seismically rugged. Second, the database also indicates that cases of poor seismic performance result primarily from basic inattention to anchorage and bracing details, and from situations where an opportunity for adverse physical interaction (e.g., valve impact) exists.

These observations suggest that the designs of existing plants are generally adequate to insure severe-accident resistance through functionality of safe shutdown components (structures and equipment) following a larger-than-design-basis earthquake. An exception may exist, however, if the perception of seismic hazard has changed (since design) to be markedly larger, for example due to the Charleston earthquake issue[1]. For such an exception, one may suspect that the threat of a design-dependent failure may be significant. Such potential design-dependent vulnerabilities, although rare, may need to be considered in the severe-accident review of existing plants that have seismic hazards that are

[1]Documentation describing the effects, on seismic hazard results, of the Charleston earthquake issue indicates that such cases are rare (11).

comparatively high relative to their designs.

Problems of anchorage, bracing, and physical interactions can be referred to as design-independent vulnerabilities, because they have the potential to exist in any plant, regardless of its design level. Such potential design-independent vulnerabilities should be considered in the severe-accident review of any plant, unless the seismic threat is negligibly low.

The NRC statement on severe-accident policy (1) declares that the existing population of U.S. plants poses no undue risk to public health and safety, and that the intent of severe-accident policy is not to re-evaluate the design bases of existing plants. For U.S. plants, therefore, the severe-accident concern should be properly focused on identifying potential vulnerabilities of the design-independent type.

Experience indicates that potential vulnerabilities, of either the design-independent or design-dependent type, are most effectively found in a thorough plant walkdown that follows recommendations for seismic margin reviews (5). No detailed calculations (for instance, of HCLPF capacities) need be conducted to find and resolve design-independent potential vulnerabilities, which are generally inexpensive to fix.

In addition, based on actions taken in light of results from past seismic studies, it is unlikely that implementation of modifications to a design-dependent condition will either be cost-effective or will achieve a substantial (absolute) reduction in plant or plant-population risk. These arguments further support the conclusion that analysis of design-dependent conditions is generally unwarranted except where the seismic hazard is comparatively high relative to the design basis.

Description of Approach

Differentiating plants based on design-basis hazard is, therefore, clearly a useful process for making decisions regarding severe-accident evaluations. The process of determining design-basis hazard (or, even more generally, the hazard associated with any spectrum) is relatively straightforward, as can be described by the following steps (see Figure A-1):

1. Obtain plant seismic design (SSE) spectra. For each plant, adjust the spectral ordinates appropriately to account for the damping used in the plant design analyses. For plants located

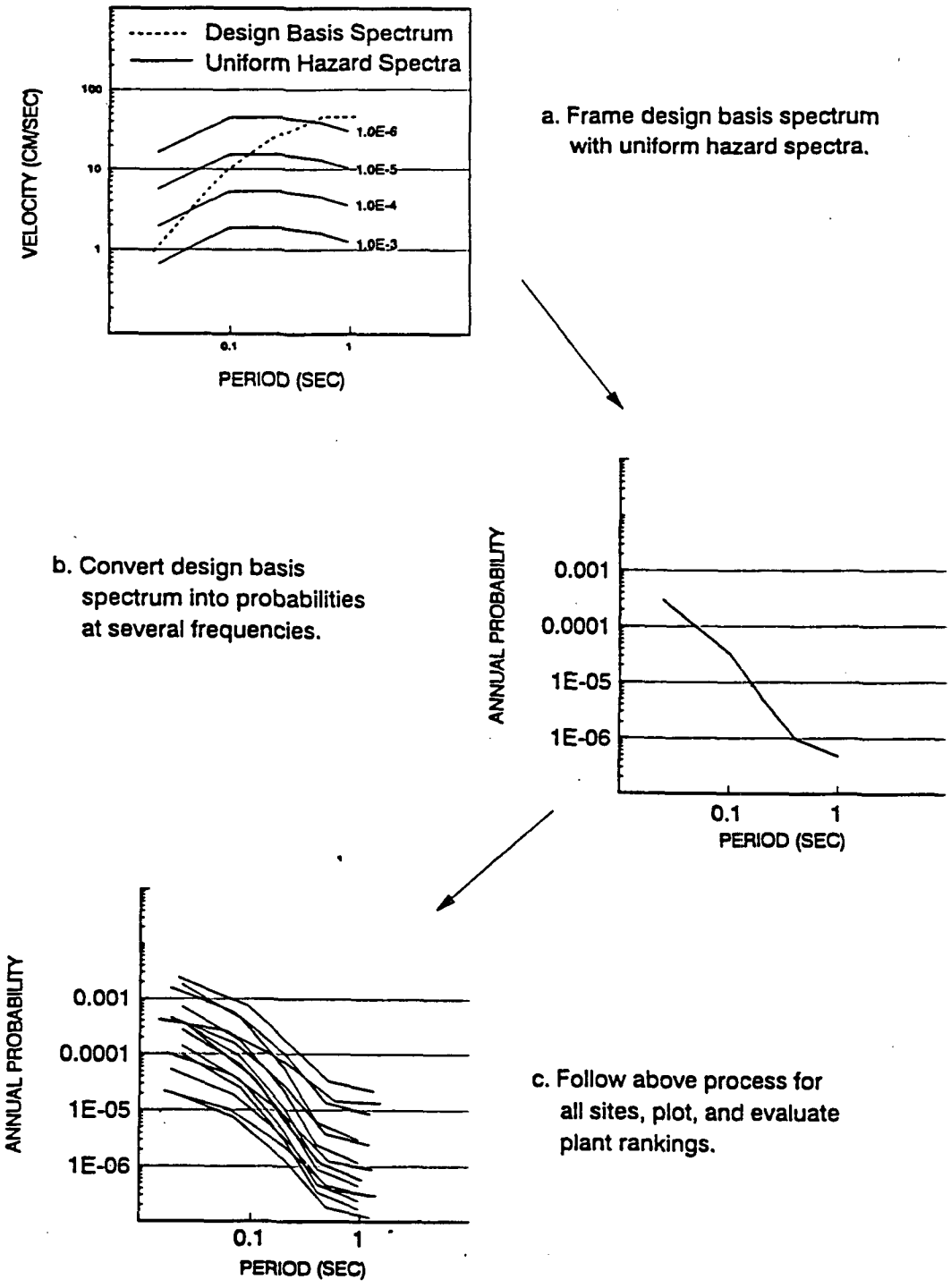

Figure A-1. Illustration of process to obtain design-basis hazard.

on both rock and soil, there may be two or more design spectra to consider. (Figure A-2 plots SSE spectra for the population of eastern U.S. nuclear power plants).

2. Obtain, for various exceedance levels, the (median[2]) plant uniform hazard spectra (5%-damped[3]), reflecting the different types of soil conditions at the power plant.

3. For each design-basis, frame (interpolate or graphically overlay) the spectrum within the array of UHS curves.

4. Select a given frequency of vibration and interpolate, between UHS for different exceedance frequencies, the value of hazard at the design-basis spectral ordinate. Repeat this interpolation procedure for several vibration frequencies, to construct a spectrum of design-basis probabilities of exceedance.

5. Perform the analysis for all design-basis spectra of interest.

Results of performing this process for eastern U.S. (EUS) nuclear power plants, using the EPRI median hazard results, are shown in Figure A-3. It is seen there that seismic design-basis hazards vary substantially both from plant to plant and from (vibration) frequency to frequency. (A similar observation is obtained if one examines design-basis hazard spectra derived using median results of the LLNL methodology).

Application of Approach

As discussed in Section 2, the design-basis hazard results guide the selection of review methods for seismic IPEs. The question each licensee must answer is: should its plant undergo a reduced-scope assessment, a focused-scope SMA, a full-scope SMA or SPRA?

In approaching this question, to simplify the description of design-basis hazard, a composite probability of design-basis exceedance is determined, considering

[2]Use of the median UHS is recommended here because it produces the greatest consistency in comparisons between EPRI and LLNL hazard results.

[3]Uniform hazard spectra are typically constructed for a 5% damping level. When making comparisons among plants, it is important to use a consistent reference damping level, regardless of the damping used in plant design analyses. In this manner, the reference damping level is associated with realistic seismic response, whereas the design damping level is associated with plant seismic capacity. A lower design damping (resulting in a higher capacity for a given design-spectrum shape), for instance, should properly produce a lower probability of exceedance (as measured by hazard results based on the reference response damping.)

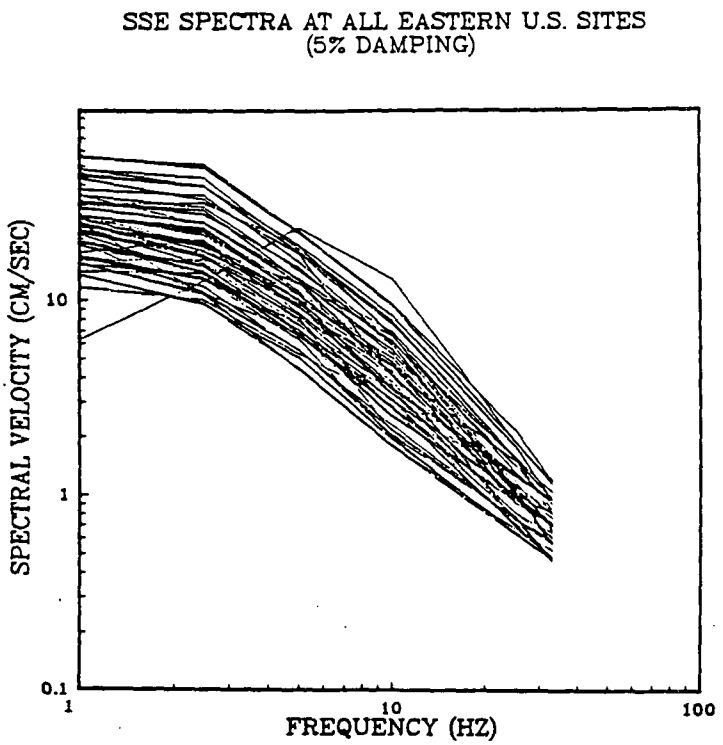

Figure A-2. Design spectra for the population of EUS nuclear power plants.

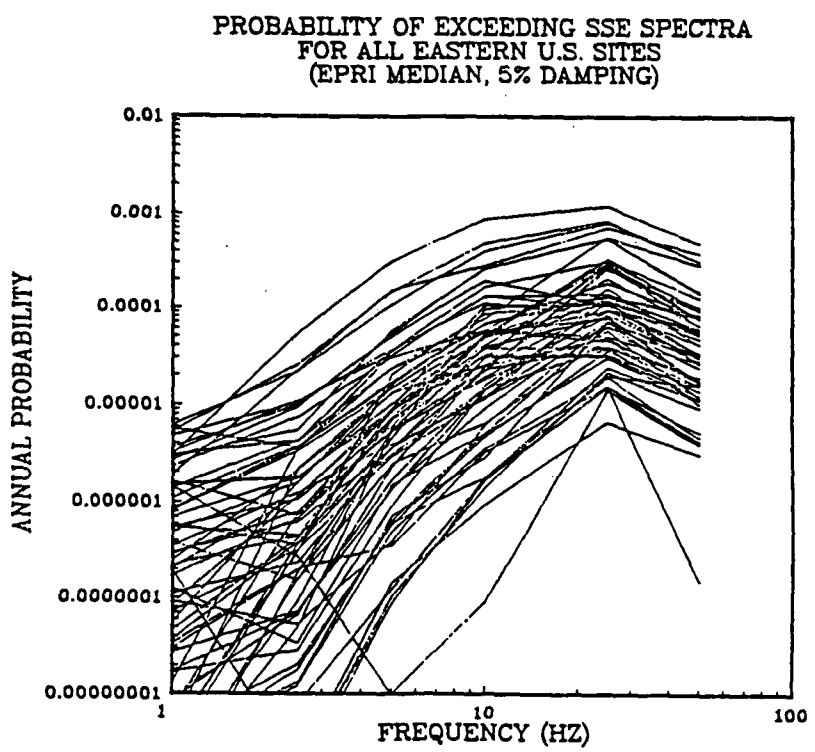

Figure A-3. Design-basis hazards for EUS nuclear power plants, based on EPRI median results. (For simplicity, 5\% damping is used for all SSE spectra).

results at multiple (vibration) frequencies. The method to use for obtaining this composite design-basis hazard measure is that recommended by the NRC (3), where 2/7 weight is given to the design-basis hazard at each of the three frequencies: 2.5 Hz, 5.0 Hz, and 10.0 Hz; and 1/7 weight is given to the design-basis hazard at the peak ground acceleration (PGA). By this process, one obtains a composite measure, (separately) from EPRI median and from LLNL median hazard results, for each EUS plant considered.

Next, these composite design-basis hazards are ordered, from highest to lowest, and the results plotted separately for EPRI and LLNL hazards analyses, as have been shown in Figures 2-1 and 2-2. Using these results of ordered design-basis hazard, one can differentiate plants based on the level of composite design-basis hazard, as discussed in Section 2.

Assisted by the information in Section 2 and, in consideration of NRC's final revisions to Supplement 4 of Generic Letter 88-20, licensees can select appropriate review methods for seismic IPEs of their plants. The decision on an appropriate review method will depend on the specific situation that exists for any given plant. For instance, some plant-specific considerations that must be factored into the seismic-IPE review selection process include:

- Whether or not an SMA or SPRA already exists for the plant;
- Whether or not seismic hazard results have been obtained for the plant site;
- Whether or not the plant must satisfy A-46 criteria; and
- To what level of detail, if any, the licensee intends to conduct cost-benefit analyses for potential modifications.

If a deterministic review method is selected, the licensee has the option to justify a lower review scope and/or lower review level than that recommended by the NRC, based on consideration of Section 2 or based on more site-specific information or evaluations. Alternatively, the licensee may wish to exercise its option to perform an SPRA review (e.g., if it intends to conduct explicit cost-benefit analyses to provide quantitative support for decisions), even though Section 2 of this document or the NRC may recommend a deterministic review. Such decisions are best made by licensees on a plant-by-plant basis. In formulating these decisions, the licensee may wish to keep in mind that the guidances in this document and in NUREG-1407 on review method selection provide general recommendations based on general analyses; hence, the licensee is given latitude

in selecting a review method, in consideration of its preferences and of more plant-specific information it may have or may develop.

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Appendix B

RECOMMENDED PROCEDURES TO ADDRESS HIGH-FREQUENCY GROUND MOTIONS IN SEISMIC MARGIN ASSESSMENT FOR SEVERE ACCIDENT POLICY RESOLUTION

by

John W. Reed
Robert P. Kennedy
Robert P. Kassawara

CONTENTS

Section 1

INTRODUCTION

The purpose of this appendix is to provide recommendations for addressing high-frequency seismic motions in a Seismic Margin Assessment (SMA) for Severe Accident Policy resolution. A research program to address high-frequency seismic motions was performed by the Electric Power Research Institute (EPRI) to provide the basis for the recommendations (1). An analytical investigation was conducted, and a series of tests are planned by EPRI in the near future.

The recommendations for addressing high-frequency motions are made in the context that the SMA procedures (2) are acceptable to both industry and the NRC for seismic capacity reviews of nuclear power plants and that only minor changes are needed for the effects of high frequencies. It is recommended that the SMA screening guidelines for structures and equipment be keyed to response spectrum parameters, not to the peak ground acceleration (PGA). Also, it is recommended that the input ground response spectrum to be used in a SMA be modified in the high-frequency region to reflect the lack of damageability from motions in this range. Recommendations for electrical functionality failure modes (e.g., relay chatter) for relays and other electrical components, which may be sensitive to motions with frequencies greater than about 16 hz, are also provided.

It is important to recognize that the recommendations made here for structures and components reflect the ability of the supporting structural elements to resist seismic motions in a ductile fashion. Although some elements (e.g., welds) are brittle relative to low-frequency damaging seismic motions, they have some ductile capacity to resist the small displacements associated with high-frequency motions. Separate recommendations are also given for electrical functionality failure modes, which maybe acceleration sensitive and do not have ductile capability.

Section 2 of this appendix gives the background on the high-frequency seismic motion issue. In Section 3 recommendations are given for modifying the SMA procedures to include the effects of high-frequency seismic loads. Finally, Section 4 lists the references used in this appendix.

Section 2

BACKGROUND ON HIGH-FREQUENCY SEISMIC MOTIONS

The results of recent seismic hazard analyses conducted for the eastern U.S. (EUS) indicate that uniform hazard spectra (UHS) have relatively large high-frequency spectral acceleration content above approximately 10 Hz. This is in contrast to typical plant design or seismic margins-type input, such as the NRC Regulatory Guide (R.G.) 1.60 or NUREG/CR-0098 response spectra. However, these same UHS have significantly lower spectral acceleration content at the low-frequency end of the response spectrum (i.e., between about 1 and 10 Hz), which indicates to the seismic engineering community that these motions are less damaging than traditional design or evaluation-type input. Figure B-1 shows the relationship between the median NUREG/CR-0098 response spectrum anchored to 0.3 g peak ground acceleration (pga) and an example UHS, with a 10,000 year return period (at the 0.85 fractile), from a hazard analysis using the EPRI methodology for a EUS nuclear power plant site.

In the analysis and design of nuclear power plant structures it is widely felt that to be damaging, earthquakes must be rich in spectral content in the 1 to 4 Hz frequency range. Because most nuclear power plant structures and equipment have fundamental frequencies significantly above 1 Hz, strong amplified motions below 1 Hz caused by soft soil conditions, although potentially damaging to flexible structures such as high-rise buildings, do not affect typical nuclear power plant sites. Thus for stiff nuclear power plant structures at typical soil or rock sites it is necessary for energy to be present in the 1 to 4 Hz frequency range for earthquakes to be potentially damaging. It is believed that the damaging frequency range for equipment probably is slightly higher, but less than 10 Hz. A possible exception is anchorage of equipment at high ground motion levels, but for this case spectral displacement, not acceleration, is more important in defining damage potential. In Figure B-1 the NUREG/CR-0098 spectral shape has damage-potential characteristics but the example UHS does not.

It has also been observed that damage correlates well with elastic spectral acceleration in the 1 to 4 Hz range. Thus, it is likely that it is an "effective" frequency in this range as opposed to the fundamental frequency which is important

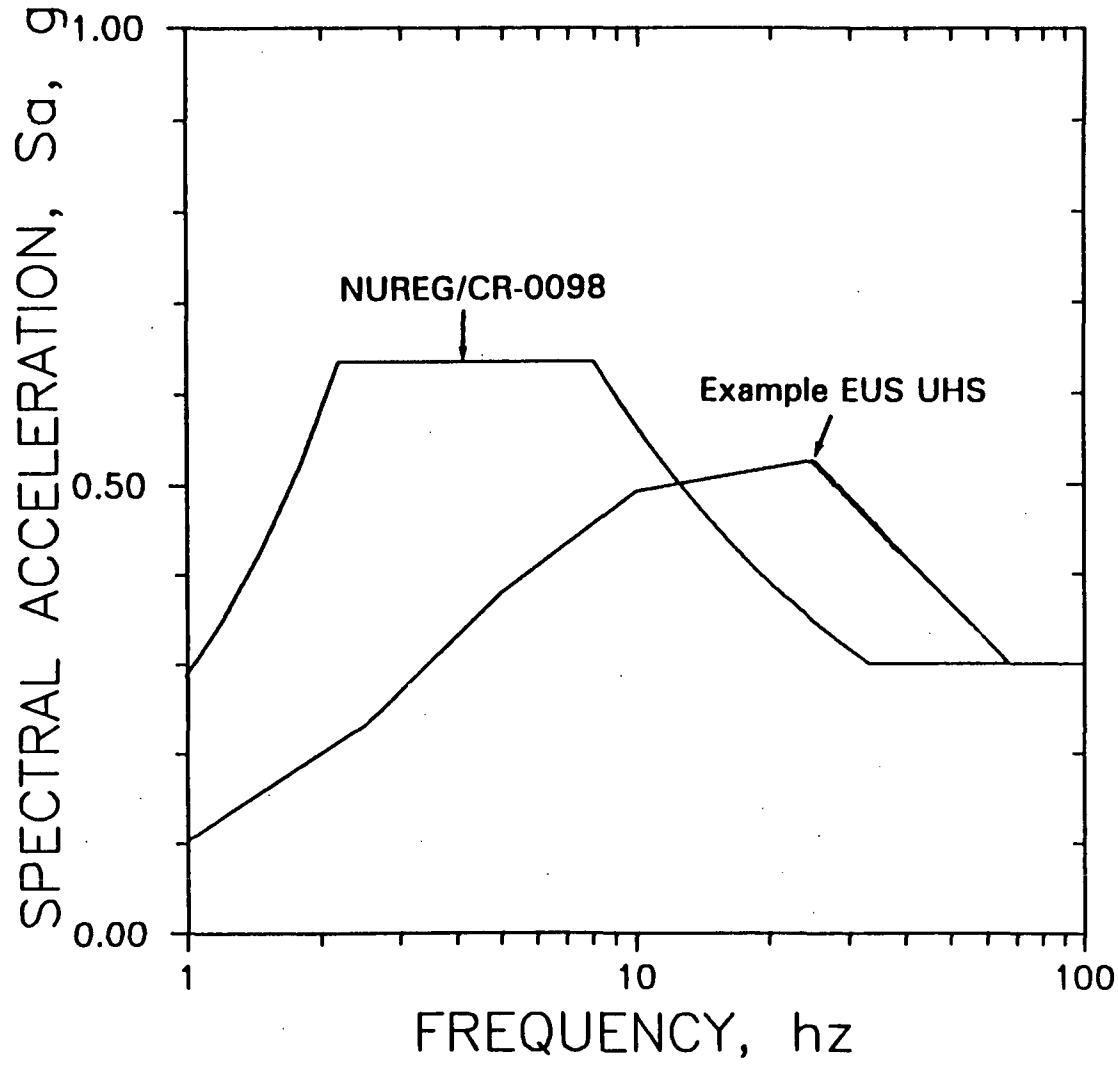

Figure B-1. Comparison of example uniform hazard response spectrum with NUREG/CR-0098 response spectrum (5 percent damped).

to significant damage (1). These results support the conclusion that it is the low-frequency content of earthquake records which causes damage.

In a recent detailed study of the Diablo Canyon Turbine Building, conducted for the Diablo Canyon seismic probabilistic risk assessment (PRA), 25 earthquake records were input to a nonlinear building model where the model material properties were varied in a probabilistically consistent manner (3). Several hundred deterministic analyses were conducted in this study, which spanned all realistic combinations of structural and ground motion parameters. It was found that for a ground motion to be damaging it must have broad frequency content. Specifically, high spectral accelerations were required at the elastic frequencies for those dynamic modes corresponding to the relatively weak building shear walls (i.e., approximately 8 Hz) to initiate substantial inelastic drift. In addition, high spectral acceleration content also was required in the approximately 2 Hz range in order to drive the shear walls to drift levels associated with the onset of severe distress (i.e., significant damage).

One example of a high-frequency earthquake which occurred near a nuclear facility was the Leroy earthquake (magnitude M_L 5.0) which took place January 31, 1986 about 11 miles from the Perry Nuclear Power Plant (4). At the time of the event Cleveland Electric Illuminating Company was within days of core load and receipt of their 5 percent power license for Perry. The staff observed no indications of damage to systems in operation at the time, which was confirmed by follow-up inspections. However, after the earthquake the analysis of the motion records indicated that the Operating Basis Earthquake (OBE) ground response spectrum had been exceeded at frequencies above 10 Hz, and the Safe Shutdown Earthquake (SSE) response spectrum had been exceeded above 15 Hz. The maximum acceleration of the ground motion at the foundation level was 0.18 g with a corresponding peak 5 percent damped spectral acceleration of about 0.8 g; however, the peak ground velocity was only 0.9 inches per second, and the maximum ground displacement was only 0.06 inches. The latter two values indicate low energy content in the 1 to 4 Hz frequency range; thus, it is not surprising that no damage occurred. In addition, the cumulative absolute velocity (CAV) value for this event was only 0.08 g-sec which is much lower than the OBE exceedance criterion limit of 0.30 g-sec (4).

Experience with high-frequency motions from other environments indicate that they are much less damaging than their low-frequency counterparts. In the development of a criterion for determining the exceedance of the OBE, earthquake experience

concerning the damaging characteristics of high-frequency motions was collected and documented (4). Based on a review of ground motions caused by conventional high explosive blasts, and their effect on structures, it was concluded that no damage will occur to engineered structures and equipment for short-duration ground motions with spectral accelerations below the envelope of the OBE response spectrum and a threshold cracking spectrum (established from blast data) as shown in Figure B-2. However, it has not been established whether these conclusions can be extrapolated to longer duration large-magnitude high-frequency events. The conclusions that damage is not caused by high-frequency motions is also supported by fragility data for ductile equipment, performance of equipment under industrial environment vibrations and code requirements for high-frequency loading.

Based on past experience, both from earthquakes and other physical phenomena which produce high-frequency motions, it is believed that high-frequency input, which is a predominant part of UHS, as shown in Figure B-1, will not be damaging to ductile structures and equipment. However, there is still a potential concern that these motions may be an issue for acceleration sensitive equipment and relays, contactors, motor starters and switches. For this category of components the effects of high-frequency motions should be addressed only if it is determined from a systems perspective that functional failure (e.g., relay chatter) during strong motion is detrimental or equally that the safety function affected can not be confidently recovered after an earthquake.

In order to evaluate the capacity of relays, high-frequency input at the floor level is required. The difficulty in analyzing acceleration-sensitive components for high-frequency input is in the development of realistic floor response spectra. The number of finite elements and mass nodes used in traditional dynamic analysis of nuclear power plant structures is generally not large enough to properly model the effects of high-frequency input. On one hand, using the current models, the high-frequency motions will be filtered out giving the incorrect impression that high-frequency response at the various floor levels does not exist. On the other hand, if models are constructed which properly pass the high-frequency motions through the structure, the analysis cost would be prohibitive. Based on past experience high-frequency motions do propagate through structures. The earthquake near the Perry plant demonstrated this to be true. However, there is evidence to suggest that high-frequency motions are not significantly amplified, which allows a simplified approach for addressing these motions as recommended in the following section.

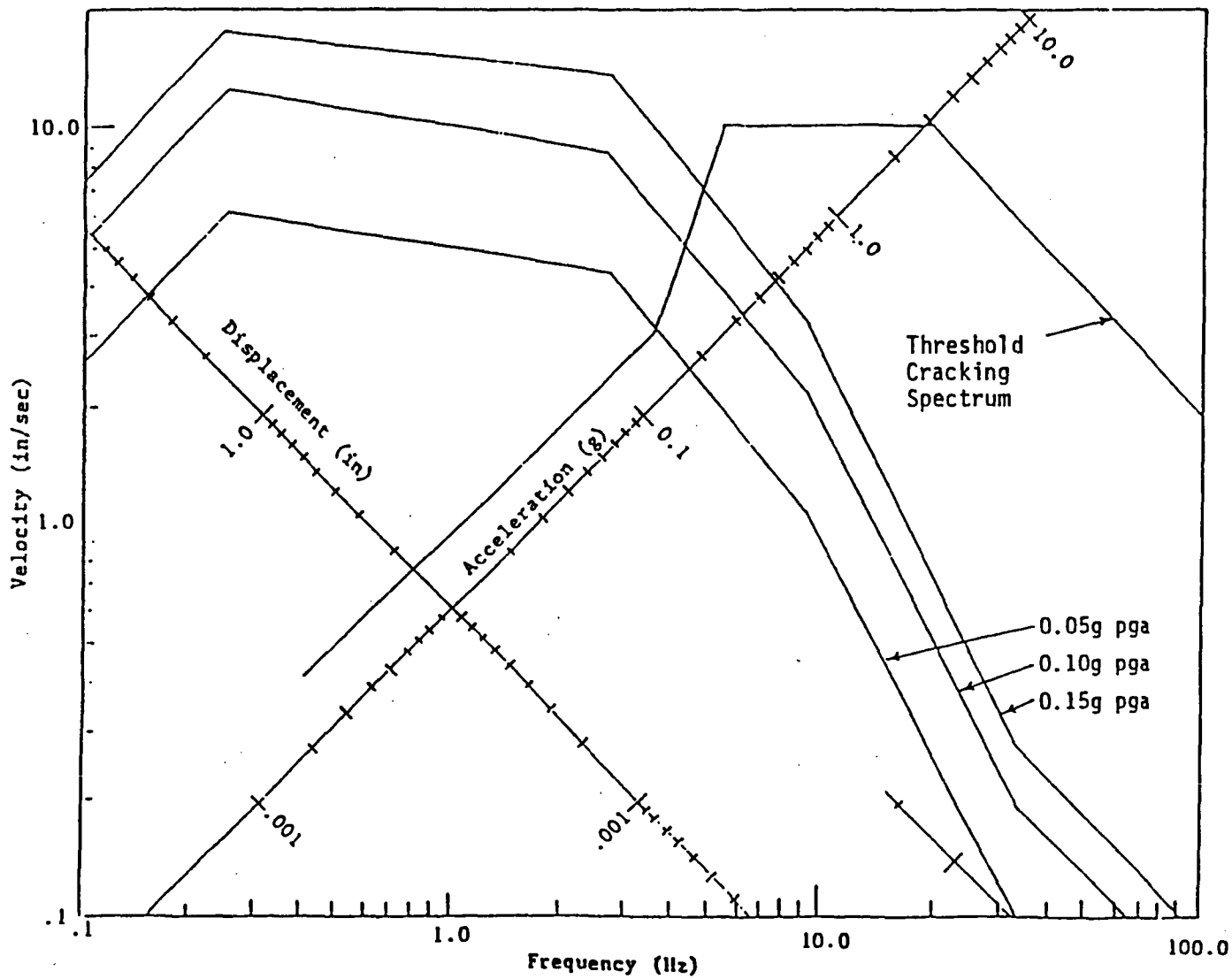

Figure B-2. Comparison of R.G. 1.60 ground response spectrum to the threshold cracking spectrum developed from ground motion due to blast (3).

It is believed that high-frequency ground motions which have dynamic characteristics consistent with the EUS UHS in the range of about 10,000 year return period will be much less damaging than low-frequency earthquakes with the same peak spectral acceleration. These latter ground motions traditionally have been the basis for seismic analysis and design (e.g., Olympia, Taft and El Centro No. 5).

Section 3

RECOMMENDED PROCEDURES

Conservative recommendations for incorporating the effects of high-frequency seismic motions in SMA are provided in this section. These recommendations reflect the current data and engineering understanding of high-frequency ground motions that might occur at nuclear power plant sites in the central and eastern United States.

References (2) and (5) describe the procedure for performing a seismic margin assessment and are incorporated here by reference. The following modifications for consideration of high-frequency seismic motions are recommended to the SMA procedures:

- The seismic capacity screening guidelines, which are keyed in References (2) and (5) to peak free-field ground acceleration values (i.e., 0.3 g and 0.5 g), are referenced to response spectral parameters: i.e., spectral acceleration (S_a) and spectral velocity (S_v).
- Input ground response spectra used in a SMA should be modified at high frequencies to reflect the lack of damageability to safety-related nuclear power plant components.
- For relays that are sensitive to input motions above 16 hz special considerations are recommended.

A discussion of the three recommended modifications and their supporting bases are given in the following subsections.

RESPONSE SPECTRUM PARAMETER LIMITS FOR SEISMIC CAPACITY SCREENING GUIDELINES

The seismic capacity screening guidelines used in SMA are keyed to peak free-field ground accelerations (2 and 5). Three categories are provided: when the PGA corresponds to less than 0.3 g, 0.3 g to 0.5 g and greater than 0.5 g. For each of the three categories, requirements are given for different structural and equipment classes. For example if the PGA is less than 0.3 g, concrete containments do not need to be evaluated, but masonry walls do. Detailed screening tables are given in Reference (2) which extend the tables originally developed in Reference (5).

Consistent with the level of conservatism provided in the screening guidelines in References (2) and (5), Figure B-3 shows the corresponding spectral acceleration and spectral velocity limits which should be used with the SMA screening guidelines for earthquakes with high-frequency motions. The lower curve which consists of a S_a limit of 0.8 g and a S_v limit of 20 in/sec corresponds to the 0.3 g PGA value. Similarly, the top curve which has a 1.2 g S_a limit and a S_v limit of 30 in/sec corresponds to the 0.5 g PGA value.

When making a comparison using the screening guideline limits, the amplified portion of the input ground response spectrum in the high-frequency region should be compared to the S_a limits. Note that the check in the high-frequency region only is made with the spectral acceleration limits, not with the PGA values. Since damage is ultimately related to strain, or equivalently displacement, the PGA check is not required.

The basis for the response spectrum limits shown in Figure B-3 has its roots in the original development of the SMA screening guidelines, where the first two authors of this appendix were principal contributors (5). At that time the Seismic Qualification Utility Group (SQUG) had formed in response to the NRC issue designated as: "Seismic Qualification of Equipment in Operating Plants," which is also referred to as Unresolved Safety Issue (USI) A-46.

As SQUG began evaluating the earthquake experience data this information also was available to the authors (referred to as the Expert Panel) of Reference (5). Ultimately, the Senior Seismic Review and Advisory Panel (SSRAP) endorsed the Reference Spectrum given in Reference (6) (which is controlled by S_a equal to 1.2 g) and the Bounding Spectrum (which is controlled by S_a equal to 0.8 g). The Bounding Spectrum, which is a factor of 1.5 below the Reference Spectrum, is currently being used by SQUG in the evaluation of safety-related equipment in existing nuclear power plants. Note that the 1.5 factor reflects conservatism in using the experience data base and for the possibility that floor response spectra in nuclear power plants might be amplified more than motions in the data base buildings (6).

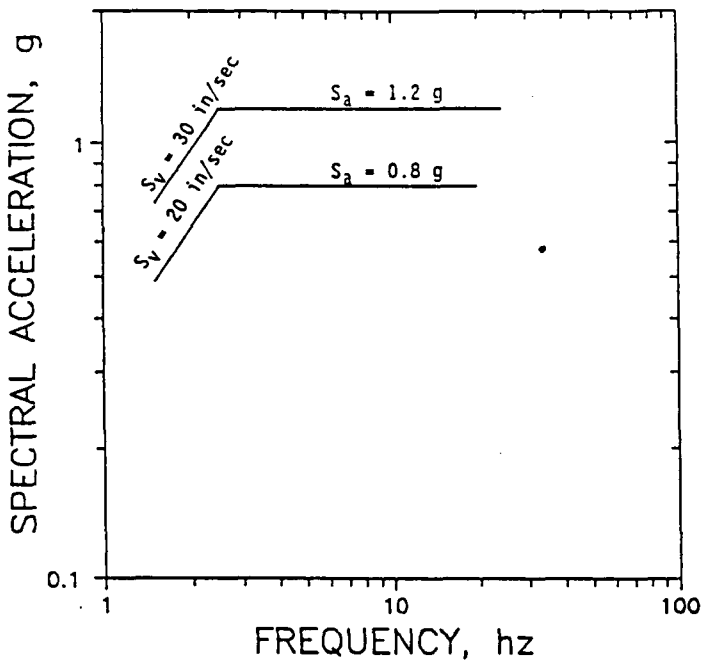

Figure B-3. Spectral parameter limits for SMA seismic capacity screening guidelines.

The equipment covered by the SMA screening guidelines essentially corresponds to the SQUG equipment. Structures are included in the SMA review which are not included in the A-46 program. However, nuclear power plant structures are rugged, and the guidance given in the SMA screening tables conservatively reflect their capacities. Moreover, a walkdown will be conducted in both SQUG and SMA reviews and there are important caveats and exclusions that must be followed in order to use the screening guidelines, both of which ensure conservative assessments.

The 1.2 g S_a limit used in the SMA screening guidelines is used to designate capacities for equipment in the 0.8 g S_a to 1.2 g S_a category and for the greater than 1.2 g S_a category. Although the SQUG guidelines do not include multiple categories, the SMA guidelines for the two ranges are consistent with the earthquake experience data base, which corresponds to the higher Reference Spectrum. Thus, the upper curve in Figure B-3 reflects the philosophy of the NRC Expert Panel and is appropriate for delineating the two higher SMA screening categories.

At the lower frequency end of the spectral capacity curves shown in Figure B-3 the spectral acceleration limits change to spectral velocity limits at about 2.5 hz,

which again correspond directly with the Bounding and Reference Spectra. The lower frequency limits of the spectra in Figure B-3 are about 1.5 hz, which is believed to be at about the practical frequency limit of nuclear power plant structures and equipment on soil sites.

The experience data base, upon which the SQUG program derives nuclear power plant equipment capacities, is tied to the earthquake characteristics at the data base sites. Thus, the Bounding (and Reference) Spectrum roll off at frequencies above 7.5 hz as shown by the spectral curve in Figure B-4. However, at the upper frequency end of the limit spectra shown in Figure B-3 the spectral acceleration values at frequencies higher than 7.5 hz are projected flat at the S_a limits of 0.8 g and 1.2 g corresponding to the two curves. The limitation on the Reference and Bounding Spectra is not the equipment capacities per se, but rather the frequency characteristics of the data base earthquakes, which are western North American events that may be less rich in high-frequency content.

Extending the spectral acceleration capacity limits at constant levels for frequencies beyond 7.5 hz is justified based on an understanding of the potential failure characteristics of structures and equipment at nuclear power plants. For critical components (i.e., safety-related components that have structural properties in the range that require seismic evaluation) the fundamental frequencies are generally less than about 10 hz with a few classes of equipment with fundamental frequencies up around 15 hz. Some classes of equipment have higher fundamental frequencies, but their elastic strength is also relatively high (e.g., check valves and well-anchored heat exchangers). Thus for most components the high-frequency earthquake motions are at frequencies above their fundamental frequencies.

As discussed in Reference (1) most components in nuclear power plants are very strong. If components have high fundamental frequencies (i.e., relatively stiff compared to their mass) they also have high strength, since stiffness and strength are correlated. In addition to being strong most high-frequency components in nuclear plants also have ample capacity to resist the small displacements associated with high-frequency motions. Examples include the casing on a vertical pump motor and the exterior shell on a small heat exchanger. These types of components are also ductile, and small displacements associate with high-frequency motions can be safely accommodated if the yield capacity is exceeded.

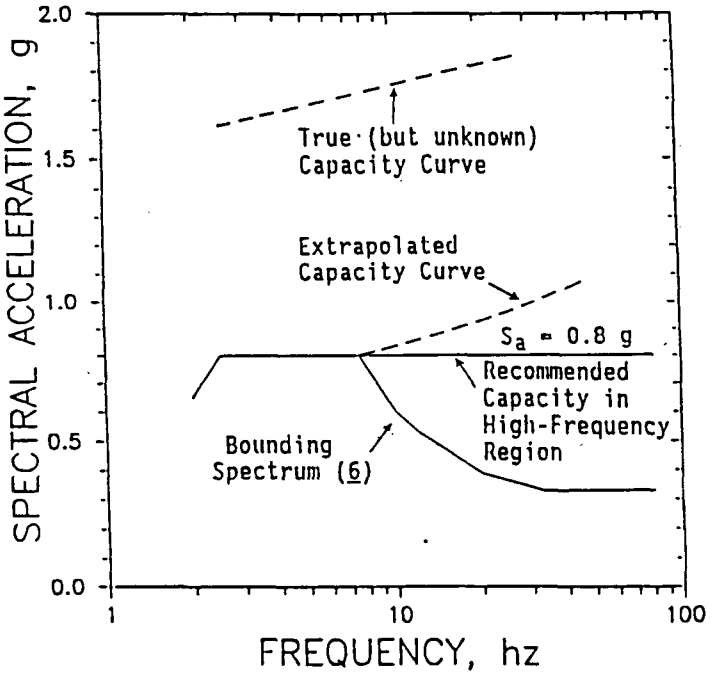

Figure B-4. Extension of SQUG bounding spectrum into high-frequency region.

In Reference (1) it is concluded that the potentially limiting case for high frequency components is welded equipment anchorage where the welds are sized exactly to met the seismic demand. Note that anchorage is not included in the SMA screening tables and must be considered separately in addition to the screening table decisions for the equipment per se. Thus, the extension of the spectral limits into the high-frequency region is for functionality and structural issues covered in the screening tables, not for equipment anchorage.

In general, the capacity of a component is either controlled by an elastic displacement limit (e.g., the gap between a fan and its housing) or a strain limit where the component goes into the inelastic range (e.g., bending of a steel support). At the top of Figure B-4 an idealized capacity curve is shown above the recommended capacity spectrum, which corresponds to the 0.8 g S_a value. It is assumed here that the fundamental resonance frequency is to the left of the left end of the true capacity curve. This capacity spectrum increases as the frequency of input increases. Figure B-5 shows example real fragility curves for a pressure control valve from Reference (7). Note that the capacity in Figure B-5 increases with frequency in a manner similar to the idealized curve at the top in Figure B-4.

The results of the high-frequency analytical study reported in Reference (1) demonstrate that the capacity beyond yield also increases at higher frequencies. In general, if components are designed to yield at a given earthquake level the ratio of the earthquake input at failure to the earthquake input at yield increases as the component frequency is increased. This is observed in detailed nonlinear time history analyses and also is explained by pseudo linear-elastic models which fit the time history results. The principal reason for this increased capacity is due to the shift in frequency from the elastic case to the effective frequency at failure (1). A greater shift occurs as the component frequency is increased, which leads to a greater margin from yield to failure.

It is expected that the capacity beyond 7.5 hz should increase as indicated in Figure B-4. For either an elastic or inelastic type of failure model the S_a capacity increases with frequency, and the assumption that the capacity is constant (i.e., flat as shown in Figure B-3) for frequencies above 7.5 hz is conservative.

There are some limitations to these arguments. The principal issue is when the fundamental frequency of the component under consideration occurs at a frequency higher than the upper bound frequency at which the capacity has been established from experience data. For this case there may be a dip in the capacity versus frequency curve in the vicinity of the fundamental frequency, and the shape of the capacity curve would be different than shown in Figure B-5. However, it is unlikely (anchorage aside) that high-frequency components exist that either do not have high yield capacities, or which do not have high displacement capability between yield and failure which can accommodate the small displacements associated with high-frequency earthquakes. Experience from other environments supports the conservative recommendations. Based on the work in References (1) to (4), strong evidence is found that high-frequency seismic motions will not be damaging to nuclear power plant components. Again these results emphasize that it is the seismic motions in the low frequency range (i.e., approximately 1 to 4 hz) that potentially can cause damage to nuclear power plant components.

INPUT RESPONSE SPECTRUM REDUCTION AT HIGH-FREQUENCIES

Reference (1) provides a procedure and technical basis for reducing ground-level response spectra at high frequencies for the purposes of analyzing components which have not been screened out and for equipment anchorage. For acceleration sensitive components such as relays the unmodified ground response spectra must be used in the evaluation. Consideration of electrical functionality failures is

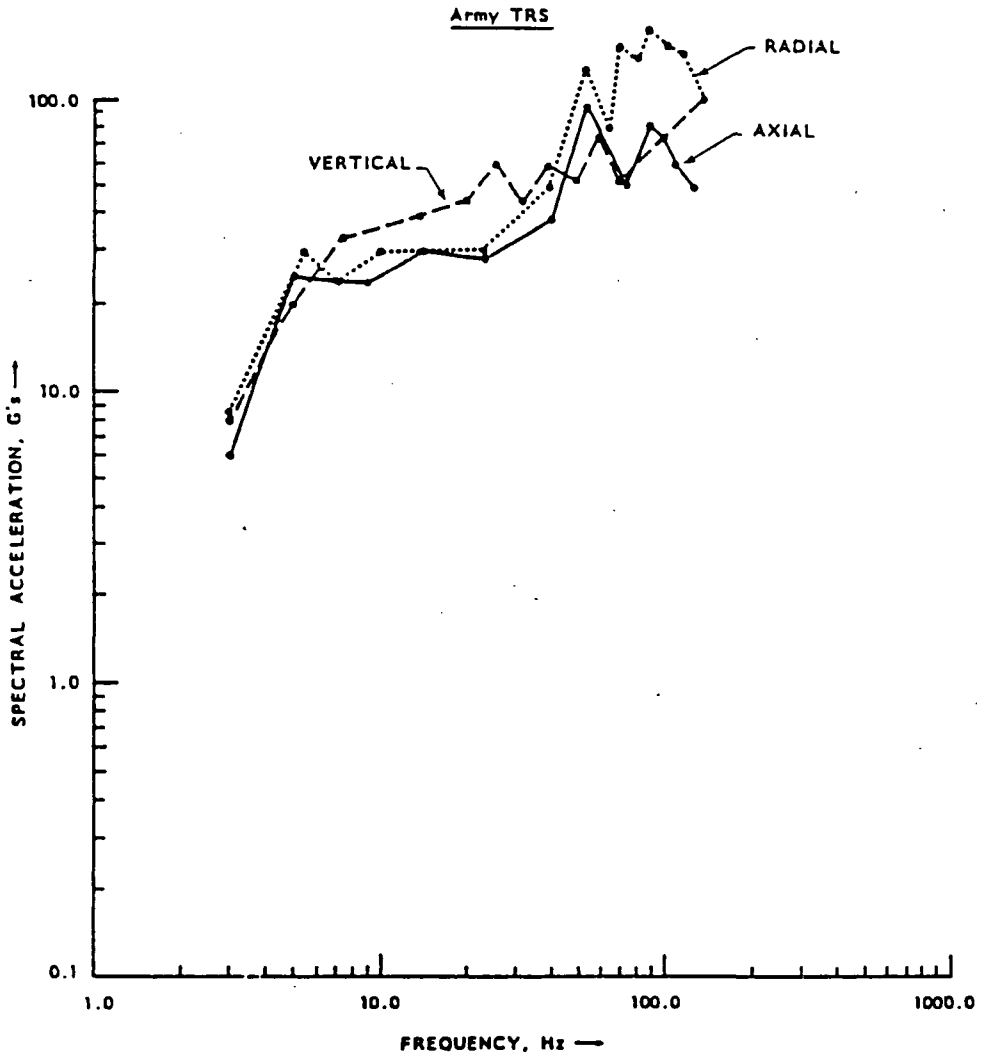

Figure B-5. 2 percent damped pressure control valve capacity spectra (7).

discussed the last section of this appendix. The reductions for high-frequency should be applied in addition to the reductions recommended for the incoherence of ground motion recommended in Reference (2).

In resolving severe accident issues a probabilistically-based review level earthquake (RLE) is recommended to be used as discussed in Appendix E. The RLE for a EUS site generally will be rich in high-frequency content. Because of the increased capacity for ductile components as the component fundamental frequency increases the RLE can be reduced at high frequencies. This follows in order to be risk consistent with the demand-to-capacity ratios for low-frequency components, where the ultimate capacities are essentially at the yield level. It is recommended that the RLE ground response spectrum be modified to a review level ground motion (RLGM) response spectrum following the procedures given in Reference (1) as summarized below.

Note that in screening components the RLE should be compared to the screening spectral limits shown in Figure B-3. This avoids double counting the benefits of ductility that are used in the arguments for extending the 0.8 g and 1.2 g spectral limits into the high-frequency region. The same arguments are also used for decreasing the RLE, as presented in Reference (1). However, once the screening step is completed components which are not screened out should be analyzed using the modified RLGM spectrum.

Figure B-6 shows an example RLE for a EUS site and two RLGM spectra. The higher RLGM response spectrum should be used to generate in-structure response spectra for analysis of equipment up high in a building, and the lower RLGM can be used to analysis equipment which is supported at the ground level. The higher RLGM spectrum is required because of amplification of the ground motion at higher elevations.

It is recommended that spectral reduction factors used to reduce a RLE response spectrum to obtain a RLGM response spectrum for analyzing high-frequency components in a SMA be obtained using the simplified sliding model presented in Reference (1). It is recommended that the fraction of mass at the model base and the coefficient of friction both be set at zero. Figure B-7 shows the model for component sliding from Reference (1). The following guidance is given when applying these recommendations:

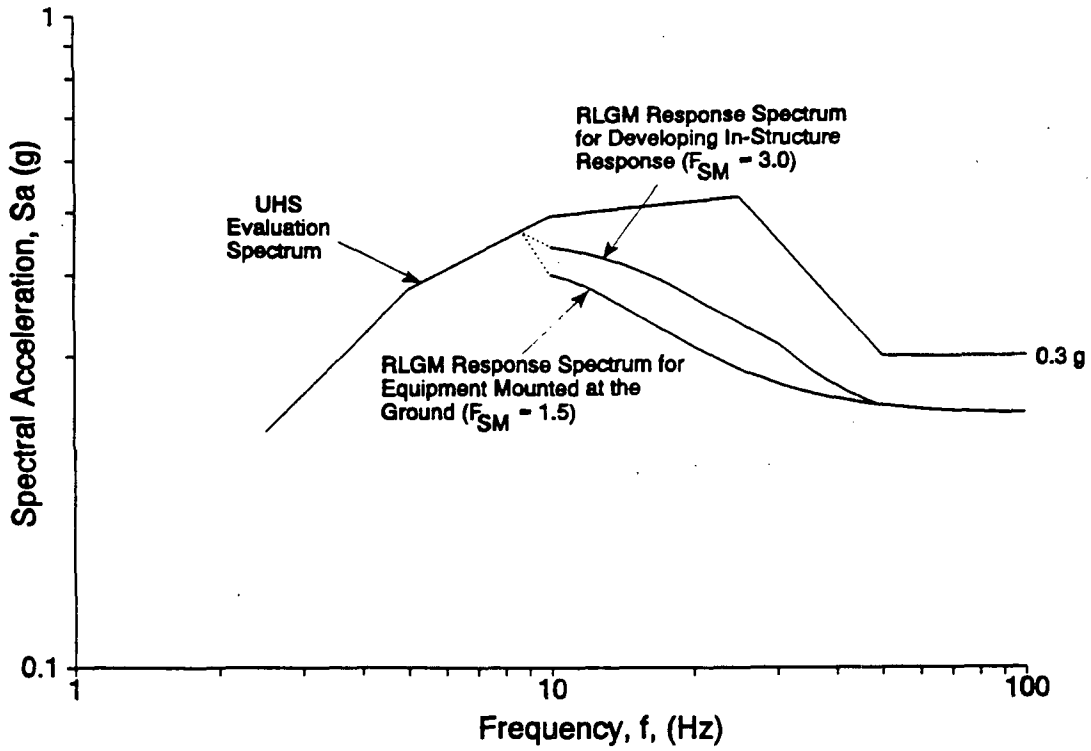

Figure B-6. Example Reduced Response Spectra for UHS Anchored to 0.3 g PGA.

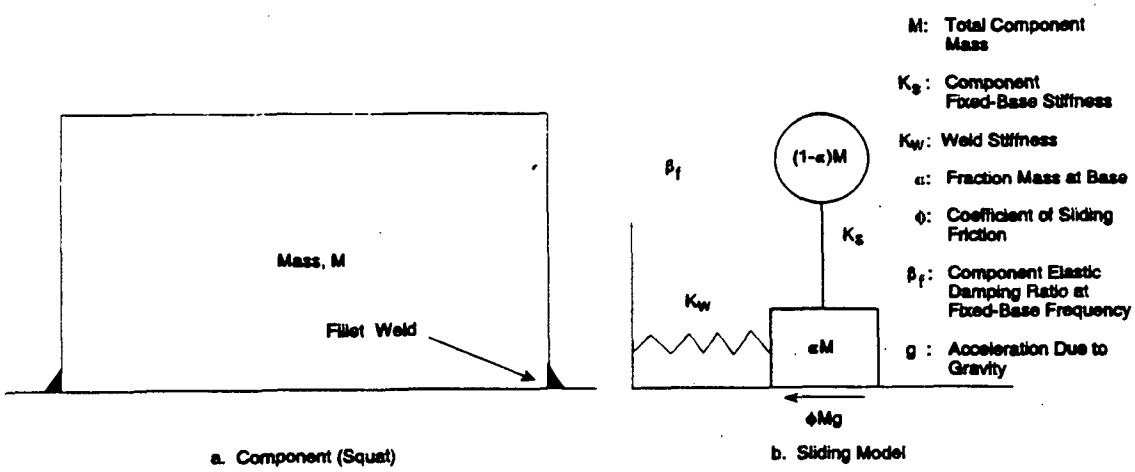

Figure B-7. Model for Component Sliding

1. The reduction should be performed for frequencies above 10 Hz, and the reduced response spectrum should be reconnected to the RLE evaluation spectrum at 8 Hz.

2. The reduced response spectrum should not be reduced below a response spectrum value equal to the peak evaluation spectral acceleration at 10 percent damping divided by 1.6 unless additional cases are considered as discussed in Reference (1).

3. The RLGM to be used to generate in-structure response spectra should be based on a safety factor, F_{SM}, of 3.0. This is equivalent to using a F_{SM} of 1.0, but based on a response spectrum equal to three times the RLE ground response spectrum. The resulting reduction factors are applied to the original ground response spectrum.

4. For equipment mounted at grade the evaluation ground response spectrum with a F_{SM} value of 1.5 can be used to determine the reduced response spectrum.

The safety factors given above include consideration of amplification of typical high-frequency EUS ground motions up in nuclear power plant structures as well as the safety margin between a HCLPF capacity and the median yield capacity.

A permissible anchorage distortion of 0.01 inch should be used to develop the reduction factors. This corresponds to the ultimate displacement capacity of a 3/16-inch weld. It is assumed in selecting a 3/16-inch weld that there may be 1/8-inch welds used to attach the sides of electrical cabinets to embedded floor plates. However, there are other sources of flexibility in this type of component that are equivalent to the models used in the high-frequency study which assumed 3/16-inch welds. Thus, the use of a 3/16-inch weld size also represents these cases. The authors believe that a permissible anchorage distortion of 0.01 inch is conservative and that all nuclear power plant components have a least the minimum amount of displacement capacity.

For plants that can justify greater permissible anchorage distortions, it may be possible to base the spectrum reduction factors on these larger permissible anchorage distortions. However, the capacity of electrical cabinets anchored with nominal welds in the plant must be considered in justifying larger permissible distortions.

The analyst should realize when performing a SMA evaluation using reduced response spectra as input that the portion of the reduced response spectrum above 8 Hz takes partial credit for ductile capacity (0.01 inch nonlinear distortion). In performing a SMA the inelastic energy absorption factor recommended in Reference

(2) also may be used, in general, for high frequency components since these factors are based on the characteristics of WUS ground motions. Note that Reference (2) recommends $F_\mu = 1.0$ for welds and other small distortion capability anchorages.

Figure B-6 shows example reduced ground response spectra for a UHS anchored to 0.3 g pga based on the sliding model following the above recommendations. The two spectra are based on F_{SM} values of 1.5 and 3.0 and would be used for an SMA evaluation for equipment at the ground and for developing in-structure response spectra, respectively.

The shape of the reduced response spectra in Figure B-6 above 10 hz are very similar to the type of ground response spectra used in the past to design and evaluate nuclear power plants (e.g., R.G. 1.60 and NUREG/CR-0098 response spectral shapes). Thus, the UHS ground response spectra currently being obtained for Eastern U.S. sites have similar characteristics to traditional design spectra when the UHS are modified to have consistent safety margins.

An intuitive justification for the recommended reductions is provided by examining displacement capacities of equipment anchorages, which are the critical structural links. Figures B-8 through B-12 show force-deflection curves for single-direction loading for fillet welds, wedge-type expansion anchors and cast-in-place concrete inserts. As can be seen from these figures welds have the minimum displacement capacity.

In Fig. B-8 the fillet weld data in the transverse direction (i.e., θ equal to 90°) has a minimum displacement capacity. However, compared to the longitudinal direction weld (i.e., θ equal to 0°), which has a high ductility limit but lower strength capacity, the transverse direction weld strength is approximately 40 percent stronger. This extra strength in the transverse direction offsets its lack of ductility. The recommended reduction factors are based on transverse loading for the smallest practical weld size, which is conservative.

Note that typical high-frequency motions are small. For example, a 1 g acceleration at 10 hz is only about a 0.1 inch displacement. Ductility capacity is relative, depending on the frequency of the input motion. For low-frequency damaging motions the displacements are large and the inelastic response of welds as shown in Figure B-8 does not provide ductile capacity. In contrast, at higher frequencies where the displacements are small, the ductility characteristics of

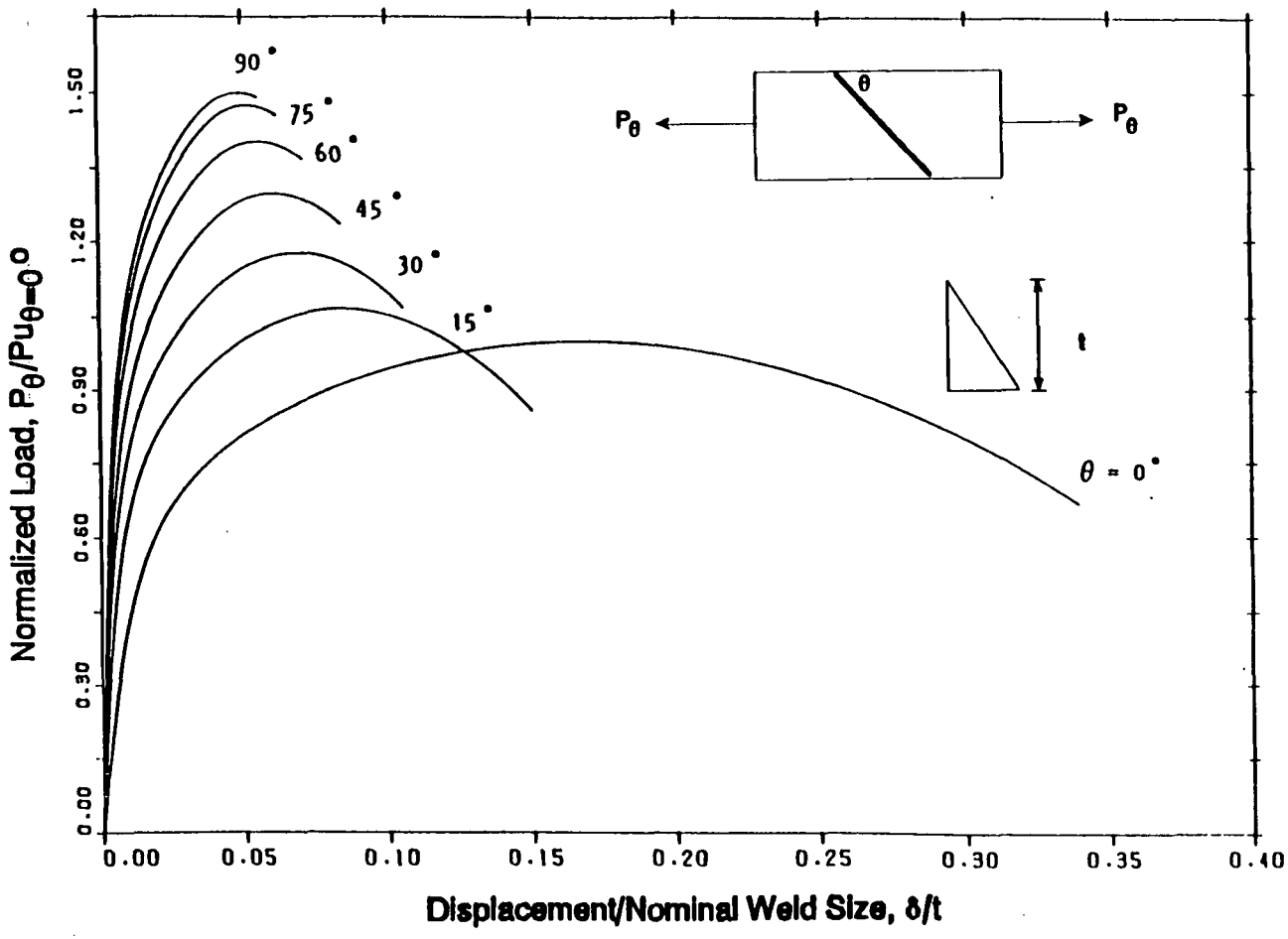

Figure B-8. Normalized Weld Curves (8)

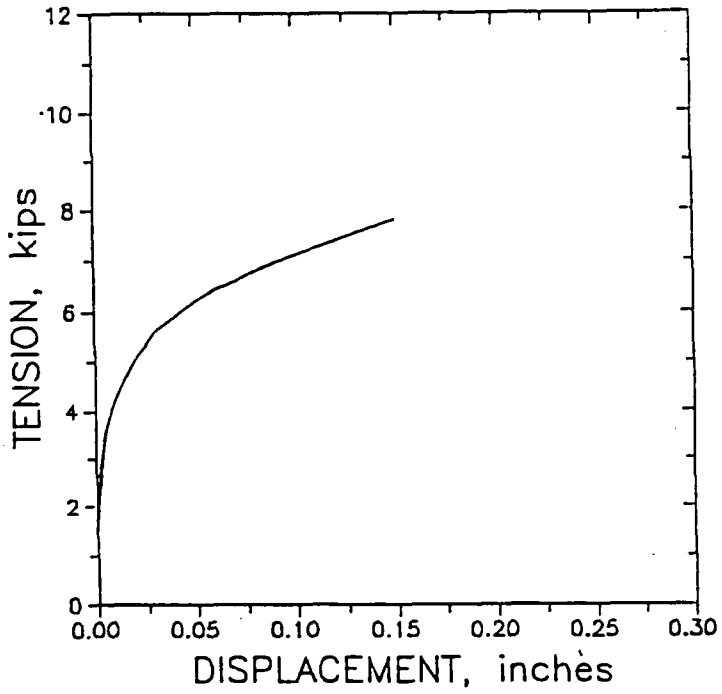

Figure B-9. Example force-displacement curve for 3/4 in. wedge type anchor bolt in tension, f'_c=3700 psi (concrete failure mode) (9).

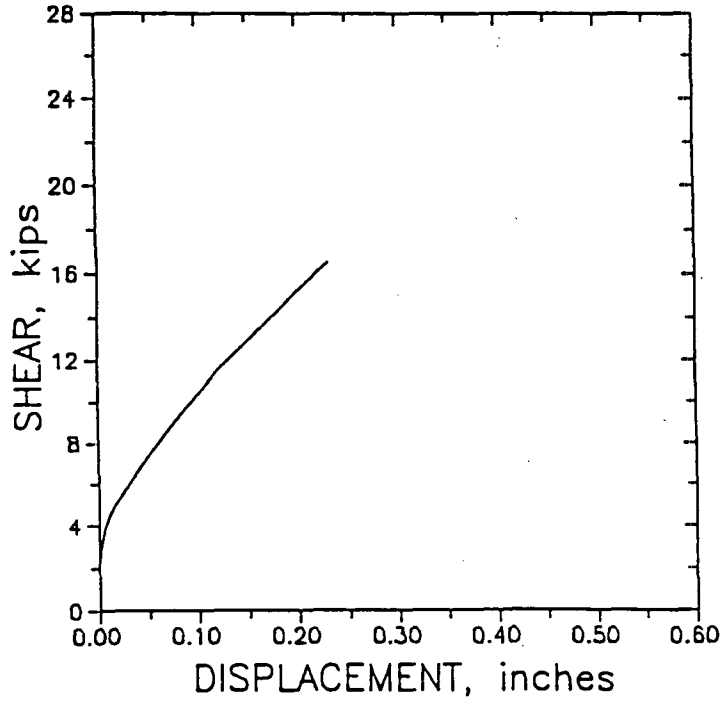

Figure B-10. Example force-displacement curve for 3/4 in. wedge type anchor bolt in shear, f'_c=3700 psi (shear failure through threads) (9).

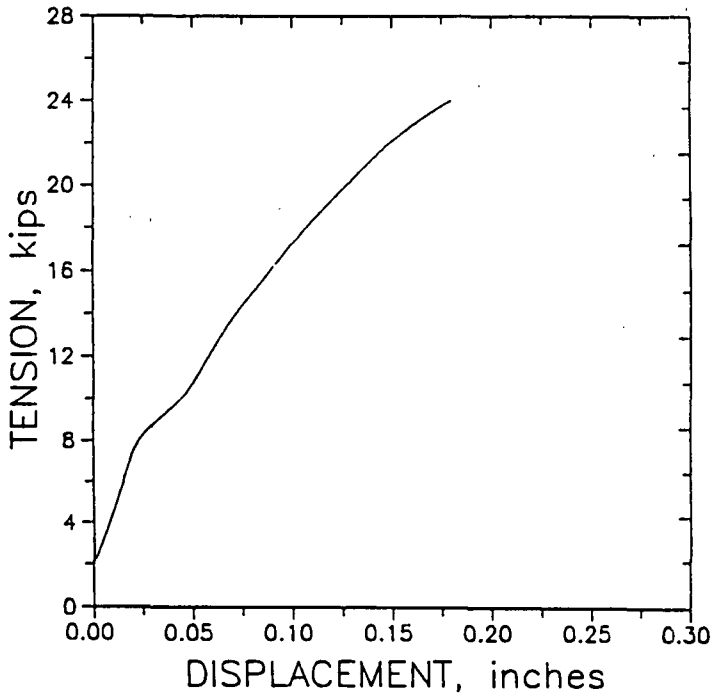

Figure B-11. Example force-displacement curve for 1 in. Richmond insert in tension, f'_c=2850 psi (10).

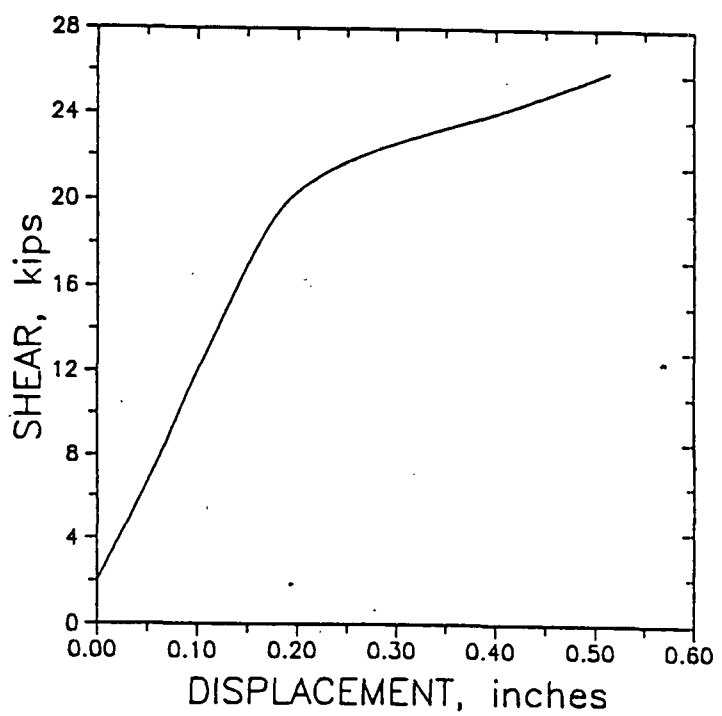

Figure B-12. Example force-displacement curve for 1 in. Richmond insert in shear, f'_c=2850 psi (10).

weld material contribute significantly to the total capacity and must be considered in determining component capacity. Note that in both the low and high-frequency cases the ultimate capacity is assumed to occur at the same displacement; thus, for high frequency components the inelastic energy absorbing factor is larger.

ELECTRICAL FUNCTIONALITY FAILURE MODES

For relays and other electrical components which are acceleration sensitive, the procedures given in Reference (2) for addressing functionality concerns are adequate when the relay frequency is less than about 16 hz because cabinet amplification factors can be confidently estimated in this range. Also, the peak of floor response spectra will occur below 10 Hz in typical nuclear power plant structures. This coupled with the use of a flat relay capacity spectrum which ranges up to at least 16 Hz ensures that the procedures in Reference (2) can be confidently applied. For these cases no new requirements are needed and the SMA procedures are adequate.

For relays and other electrical components which are acceleration sensitive, and are sensitive to frequencies in excess of 16 hz, alternate procedures are recommended. Strategies other than direct evaluation should be attempted before trying to qualify the relays per se, for the seismic floor motion. Recently a list of relays that are sensitive to high-frequency low-level seismic motions has been developed (11). These relays should be avoided if at all possible. Other high capacity relays are also identified in Reference (12). The recent relay functionality guidelines developed by EPRI can be used to determine whether chatter is acceptable (11). The following approaches are recommended:

- If relay chatter would cause an unacceptable change of state, determine whether the plant operators can recover the required plant operating condition in a relatively short period of time after the earthquake.
- If necessary, modify the electrical circuitry to bypass the dependance on the critical relay in question.
- Replace the relay in question with a rugged model which can be easily qualified (e.g., critical frequency less than 16 hz).

If these options are not feasible then a direct capacity check of the candidate relay will be required. For this case realistic floor response spectra based on the unmodified RLE ground response spectrum which span the frequencies of interest will have to be estimated. It is anticipated that simplified analytical procedures could be developed to modify in-structure response spectra which are

based on the RLGM response spectrum. This will avoid having to generated two sets of in-structure response spectra for equipment evaluation. For example, there is evidence for typical ground response spectrum shapes currently being developed for the EUS, which are rich in high-frequency energy and significantly reduced at frequencies below 10 hz, that the spectral amplifications at higher elevations in nuclear power plant structures are less than a factor of two relative to the ground (1).

Next, an in-cabinet capacity spectrum should be developed which includes the critical relay mounted in the cabinet. This step is necessary since at high frequencies (i.e., greater than about 16 hz) motion amplification factors from the floor to the mounting point of the relay device are highly variable and a single acceleration value that can be used in SMA can not be confidently determined. It is likely that a shake-table test will have to be performed to develop the required cabinet-level capacity spectrum. Using this capacity spectrum, the calculated floor response spectrum can be compared in the standard way to evaluate the relay.

SUMMARY

In summary, the capacity screening guidelines have been defined in terms of spectral acceleration and velocity limits (see Fig. B-3). In performing a seismic evaluation for ductile components the input ground response spectrum should be modified as demonstrated in Figure B-6 in order to reduce the portion of the response spectrum which is not potentially damaging to ductile components. The reductions for high-frequency should be applied in addition to the reductions recommended for the incoherence of ground motion recommended in Reference (2). For evaluation of relay chatter and other functionality failure modes the unaltered response spectrum should be used. Care should be exercised in developing floor response spectra for relay chatter evaluation in order to properly transmit the high-frequency motion to the floor level.

Section 4

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Appendix C

INTEGRATION OF SEISMIC IPE AND A-46 REVIEWS

by

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CONTENTS

Appendix C

INTEGRATION OF SMA AND A-46 REVIEWS

INTRODUCTION

Appendix C provides guidance for Individual Plant Examination for External Events (IPEEE) when a seismic review also is being conducted for NRC Unresolved Safety Issue (USI) A-46. In the following discussion it is assumed that the plant will be reviewed for A-46 using the Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment (GIP) (1). It is recommended that the IPEEE and A-46 reviews be conducted concurrently and that the review tasks be combined, whenever possible.

Background on the two review programs is given in the next section. This is followed by the procedures for conducting combined A-46 and IPEEE reviews. Finally, suggested strategies for combining tasks for the two reviews are given in the last section.

BACKGROUND ON IPEEE AND A-46 PROGRAMS

As discussed in Section 3 the A-46 and IPEEE reviews serve different purposes. The A-46 program is directed at assuring that the plant has been constructed properly and that the capacity of components are consistent with the plant design basis. In contrast, a SMA review is directed at the issue of seismic margin and seeks to answer the question of whether there is high confidence of a low probability of failure at an earthquake level which is higher than the plant design basis. In general, the seismic input and the component capacities are different for the two programs. Depending on both the relative seismic input and factors of safety for capacity for the two programs one review will control over the other. However, it is possible that for one type of element (e.g., anchor bolts) one review may control, but for another type (e.g., relay chatter) the other review may control. It can not be determined a priori which one will control in all cases for all components.

In general, it is expected that the HCLPF capacity using the requirements for the SMA review will be higher than the capacity using the requirements for an A-46

review. In other words, for the SSE design level the engineer should have very high confidence that the plant has a low probability of failure. This is true since the factors of safety to resist the plant design earthquake will be higher because of the underlying review philosophies. For example, the HCLPF capacity of flat-bottom tanks should be about a factor of 1.5 higher than the A-46 capacity as determined using the procedure in the GIP. By examining the plant at the SME level (which is higher than the SSE) the potential for a brittle failure can be investigated. This gives assurance that there is still adequate capacity if the SSE is exceeded. In fact, there is an assumption in the SMA methodology that the median capacity (i.e., the capacity at which there is a 50 percent chance of failure, or alternately, a 50 percent change of survival) of a component is at least twice the HCLPF capacity (2). This assures the engineer that there is no "cliff" just above the HCLPF capacity.

The objective of this appendix is to provide examples for the plant seismic capability engineers in order to encourage pursuing the differences between the requirements for the two reviews. This will provide a basis for determining which review controls for each component at the beginning of a IPEEE study to enable the analysts to perform only a single capacity calculation for each outlier that covers both reviews.

PROCEDURES FOR PERFORMING IPEEE AND A-46 REVIEWS CONCURRENTLY

There are three types of deterministic reviews in the IPEEE program, full-scope seismic margin assessment (SMA), focused-scope SMA and reduced-scope assessment. At the option of the plant owner the IPEEE and A-46 reviews can be conducted independently. However, this would require duplication of some work, which can be avoided if the reviews are combined. Since it is acceptable to perform the reviews independently, the SRT can always use directly the corresponding requirements for each review.

The procedures for each type of IPEEE review are discussed in the following subsections.

Full- and Focused-Scope Seismic Margin Assessments

Section 3 gives general requirements for performing a full- or focused-scope SMA. Guidance is provided below for plant walkdown, seismic capability assessment and review documentation when a SMA is conducted for either of these two approaches at the same time as an A-46 seismic verification of nuclear power plant equipment is

performed.

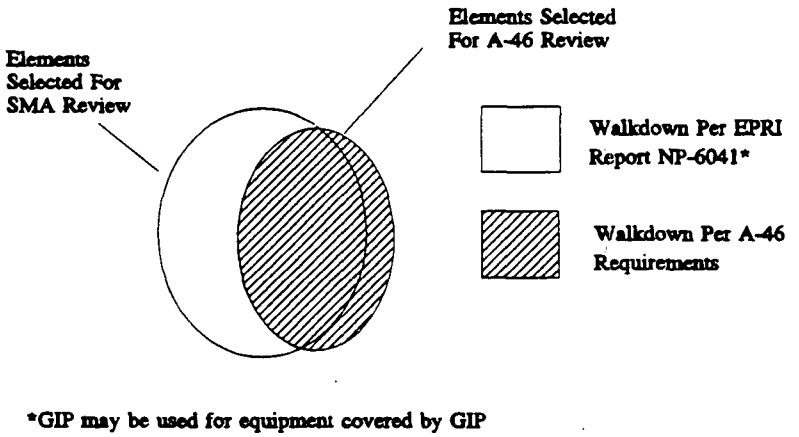
Walkdown, screening and determination of outliers. The components selected in the success paths will generally be different to some extent for IPEEE and A-46. For example, structures, piping and secondary equipment which could fail and affect success-path components are included in a SMA review but not in an A-46 review. In addition, the philosophy for providing redundancy for the primary success path is different for the two programs. This also leads to the selection of different components.

Figure C-1a shows schematically the overlap between the two sets of elements for plant walkdown for the two programs. It is recommended that elements which overlap the two programs (as well as equipment in the A-46 program) be walked down using the requirements in the GIP for the A-46 program. For elements which are common only to the A-46 review the GIP must be used. For elements which are only in the SMA review the requirements in EPRI report NP-6041 (3) can be used; however, the GIP may be used instead of EPRI report NP-6041 for all elements covered by the GIP if the bounding spectrum exceeds the seismic margin earthquake ground response spectrum and the GIP caveats are followed. This is true for the case of the median shape NUREG/CR-0098 spectrum anchored to 0.3 g.

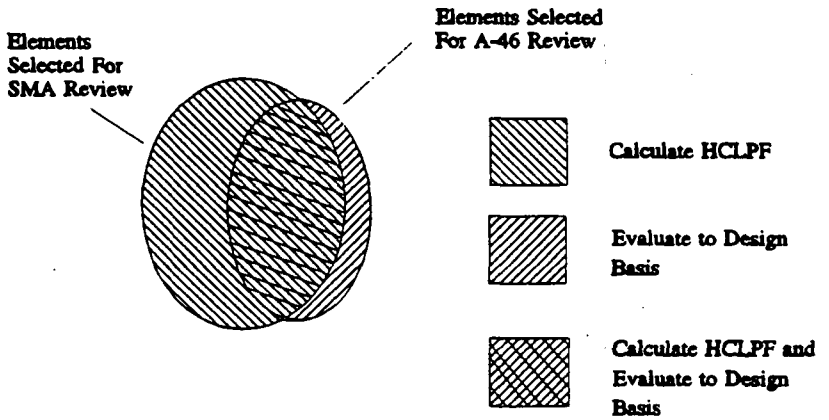
The walkdown requirements in the GIP are more demanding than the walkdown requirements for a SMA. There are some constraints which must be considered in the review when the GIP is used for the walkdown of a SMA component. The GIP requires that the following four criteria be met (1):

- The seismic capacity must be greater than the seismic demand, as defined by the earthquake experience data base and bounding spectrum, the generic seismic testing data base or equipment-specific seismic qualification data.
- Specific caveats are given in the GIP for use of the data as listed above. It is necessary to verify that the equipment meets those caveats.
- The equipment anchorage capacity, installation and stiffness must be adequate to withstand the seismic demand.
- Nearby equipment, systems and structures must not fail and prevent safety-related equipment from performing its function.

For elements common to both reviews, the seismic input is generally different. However, for functionality or structural integrity failure modes the GIP provides conservative screening procedures for equipment included in the SMA review.



a. Walkdown



b. Seismic Capability Work

Figure C-1. Combined Seismic Margin Assessment and A-46 Review Strategy

For cases where a component in the SMA review is not screened out when using the GIP requirements, the SRT can always utilize the evaluation procedures in EPRI report NP-6041. In most cases it will be more efficient to use the A-46 screening guidelines initially for all common equipment, since they are already being used for equipment in the A-46 program.

When evaluating equipment anchorage, the SRT should first investigate the relative differences between the seismic input as well as the assumed anchorage capacities for the two programs. Since the seismic input, as well as the factors of safety for anchorage capacity, is generally different for the two programs the anticipated demand-to-capacity ratios should be calculated for the two reviews. The requirements for the program with the largest ratio should be used initially for both programs. Again, if the anchorage capacity for a particular component is found to be inadequate using this approach then a specific calculation should be conducted for each program using the corresponding seismic input and acceptance criteria.

The benefits from this approach are more apparent in the walkdown process when component anchorage systems are similar to each other and generic values can be developed prior to the walkdown. However, even if most components have unique anchorage details it still will be easier to perform the anchorage capacity review for the most conservative criteria. This is true as long as it is generally found that the capacities are adequate. For cases where the anchorage is not adequate, a capacity check should be conducted using the criteria for each program.

The equipment anchorage installation and stiffness requirements for the A-46 program are more conservative than required for a SMA review. This follows from the general philosophies where it is assumed in the SMA program that the plant has been designed and constructed according to licensing requirements. In contrast, one aspect of the A-46 review is to verify systematically that equipment anchorage has been properly installed. Thus, the A-46 anchorage walkdown requirements are more rigorous. For equipment outside the scope of the A-46 program a torque test for the purpose of the IPEEE program is not required, unless the SRT suspects that a problem exists.

When walking down equipment in a plant the potential for adjacent equipment, systems or structures to fail, fall and impinge on safety-related equipment is reviewed as part of the A-46 program. This also is required when conducting a SMA walkdown. One additional consideration must be included for common components in

order to comply also with SMA requirements. The possibility of a component failing which could flood a safety-related component must also be considered.

When conducting a plant walkdown, screening and evaluation work sheets must be filled out for each element reviewed. Work sheets are provided in the GIP for A-46 walkdowns and in EPRI report NP-6041 for SMA walkdowns. These sheets should be used for elements which are included in one program. For equipment common to both programs (or for IPEEE equipment initially being screened using the GIP) the work sheets in the GIP should be used whenever possible. In addition, they should be supplemented by the additional work sheet shown in Figure C-2. The supplemental work sheet documents the required information beyond the data requested in the GIP which is required in the SMA review.

Calculation of HCLPF capacities and resolution of outliers. Equipment which is screened out during the plant walkdown is not considered further in the review. It is assumed that screened-out equipment which is common to both programs has a HCLPF greater than the seismic margin earthquake (SME) and also has a capacity which complies with the plant design basis. For elements which are not screened out during the walkdown, capacity calculations must be performed. For equipment in the A-46 review, capacities are computed using the GIP. For equipment in the SMA review HCLPF values are computed based on the requirements in EPRI report NP-6041. Thus, two capacity calculations are in general required for equipment common to both programs. Figure C-1b shows schematically the seismic capability requirements for elements in the two reviews.

As discussed in the last section in this appendix there are strategies where a common calculation can be made which provides an efficient procedure for obtaining capacities required by both programs. As discussed above for anchorage, and in Section 3 of this report, generic anchorage calculations should be performed prior to the plant walkdown. By calculating the expected demand-to-capacity ratios for the two programs the most conservative criteria can be used for equipment common to both reviews.

As stated above, when the capacity of a component is less than the demand the SRT has the option to perform the reviews independently using the requirements of each program separately. However, the purpose of combining calculations is to minimize the total effort.

Status Y N U

Sheet I of _____

Equip. ID No. _____ Equip. Class _____

RELAY WALKDOWN

1. Does spot check of essential relays indicate relays present and properly mounted? Y N U N/A

2. Are essential relays required to function during earthquake screened out? Y N U N/A

   If no, attach list of relays with locations in cabinet and general dimensions, thicknesses and details of mounting plates that support relays for later analysis.

3. No other relay concerns? Y N U N/A

Requirements for relays satisfied? Y N U

SYSTEMS INTERACTION EFFECTS

1. No potential sources could flood or spill onto cabinet? Y N U N/A

DESCRIBE POTENTIAL PROBLEMS INDICATED BY NO OR UNSATISFACTORY (Use additional sheets, if necessary)

IS EQUIPMENT FREE OF NEED FOR FURTHER INVESTIGATION, EXCLUDING RELAY CHATTER?
YES _____ NO ____

IS EQUIPMENT FREE OF NEED FOR FURTHER RELAY CHATTER INVESTIGATION? YES ___ NO ___

Evaluated by: _____ Date: _____

Evaluated by: _____ Date: _____

Figure C-2. Additional Screening and Evaluation Worksheet For SMA

Documentation of review. The review documentation should follow the requirements of each program (i.e., EPRI report NP-6041 and GIP). In general, two reports should be prepared, each which follows the documentation requirements of the applicable program.

Reduced-Scope Assessment

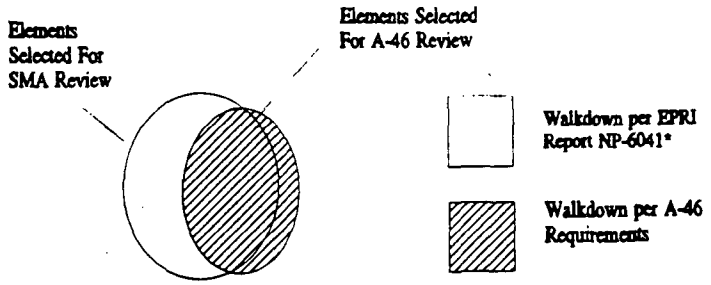
Section 3 gives general requirements for performing a reduced-scope review. Guidance is provided below for plant walkdown, seismic capability assessment and review documentation when a reduced-scope assessment is conducted at the same time an A-46 seismic verification of nuclear power plant equipment is being performed.

Walkdown, screening and determination of outliers. For the same reasons given above for the full- and focused-scope SMA reviews the elements selected for evaluation will be different for the A-46 and SMA programs. The walkdown requirements for a reduced-scope review should follow the guidance given in NP-6041. The Seismic Review Team (SRT) can use either the GIP or the screening tables in EPRI report NP-6041 to screen equipment in the IPEEE review. If a plant is conducting an A-46 review it is recommended that the screening requirements in the GIP be used for all components, whenever possible. Figure C-3 shows schematically that all components identified in both reviews should be reviewed using the requirements in the GIP.

Elements such as piping and structures are outside the scope of an A-46 review. For these elements the screening requirements in EPRI report NP-6041 should be used.

Calculation of element capacities and resolution of outliers. Equipment which is screened out during the plant walkdown is not considered further in the review. It is assumed that screened-out equipments have capacities which comply with the plant design basis. For elements which are not screened out during the plant walkdown, capacity calculations for the SSE input must be performed.

It is recommended that the acceptance criteria in the GIP be used to develop capacities for all components in a combined review, whenever possible. For structures, piping and other components which are not included in the GIP the requirements in the FSAR should be used.



*GIP may be used for equipment covered by GIP

Figure C-3. Combined Reduced Scope and A-46 Review Strategy

Documentation of review. The review documentation should follow the requirements of the GIP for the A-46 review. The basis should be documented for the selection of additional components identified for the reduced-scope review.

STRATEGIES FOR REVIEW EXECUTION

As discussed above it is suggested that the requirements for the IPEEE and A-46 reviews be considered at the beginning of a plant seismic investigation to develop short cuts to avoid doing the work twice. Some examples of how this can be done are provided in this section.

During the screening step it is suggested that the requirements in the GIP (1) be used in the review. This implies that the caveats in the GIP are satisfied. The principle requirement is that the SME ground response spectrum is enveloped in the frequency range of interest by the Bounding Spectrum in the GIP. For example, this will occur when a NUREG/CR-0098 median response spectrum shape is anchored to a 0.3g, or less, peak ground acceleration (pga). This was the argument made in the review of the Edwin I Hatch Nuclear Plant SMA for using the GIP in the plant walkdown (4). For SMEs higher than 0.3g it may be still possible to use the GIP at lower elevations. For this case it must be shown that the in-structure response spectra are enveloped by 1.5 times the Bounding Spectrum in the GIP.

For the evaluation of outliers it may be possible to perform one calculation for each component which serves for both the IPEEE and A-46 reviews. For example, in the Hatch SMA/A-46 combined review, expansion anchor bolts were evaluated using the input and capacities as required for the SMA as given in Reference (5). In the Hatch review, new in-structure response spectra were developed for the SMA where the SME was selected to be a median shape NUREG/CR-0098 response spectrum anchored to 0.3g pga. The SSE pga for Hatch is 0.15g, and it was assumed that the A-46 input could be obtained by scaling the SMA input by the ratio of the SSE pga to the SME pga. General procedures for scaling in-structure response spectra are given in Reference (3). In order to determine which review controls, both the seismic demand and the corresponding bolt capacities must be considered. At the time that the Hatch SMA was performed factors of safety for shell-type expansion anchors were 4 and 3 for the A-46 and SMA reviews, respectively.

Since the in-structure response spectra for both reviews are considered to be median centered, the A-46 spectra are obtained by using half the SMA spectra (because the A-46 pga is 0.15g and the SMA pga is 0.30g) times a factor of 1.25 to convert to approximately the one standard deviation level. Thus, the demand to capacity ratio, D/C for the two studies is proportional to the following factors:

		A-46	SMA
$\frac{D}{C}$	$\propto$	$\frac{0.5 * 1.25}{1/4}$	$\frac{1.0}{1/3}$
		2.5	3.0

As seen in this example the D/C ratio is higher for the SMA review; thus, by evaluating shell-type expansion anchors for the SMA input and the capacities required in Reference (5) both reviews were covered.

Currently, the factor of safety in the GIP for the shear capacity of both shell-type and non-shell-type anchor bolts is 3.0, while the corresponding SMA value which is recommended in Revision 1 to Reference (5) is 2.0 (3). Even with these revised factors of safety the D/C ratio for the Hatch example SMA D/C ratio would be about 7 percent higher than the A-46 D/C ratio. Again, for this case performing the analysis for the SMA review would cover both reviews for bolts subjected to only shear forces.

Similar type generic demand to capacity calculations can be performed for other

elements (e.g., tension in bolts, welds and relay chatter) to determine which review controls. One component class where the calculations are generally involved is flat-bottom liquid storage tanks. Methodologies are given separately for the A-46 and SMA reviews to calculate the tank capacities (1, 3). In lieu of performing two analyses for each tank it is acceptable to use the procedure in Reference (3) for the SMA review to obtain a HCLPF capacity and to divide it by a factor of 1.5 to be used as the capacity for the A-46 review.

For all components it is conservative to calculate a HCLPF in-structure response spectrum capacity using the procedures in Reference (3) and divide the result by a factor of 1.875 to obtain an equivalent A-46 evaluation capacity, when realistic median-centered methods are used to generate the A-46 in-structure response spectra. Similarly, a factor of 1.50 is adequate when conservative design in-structure response spectra are used. If the equivalent A-46 capacity is greater than the A-46 input then the component satisfies the A-46 requirements. In general, the 1.875 factor is conservative (note above that a factor of 1.5 is adequate for flat-bottom tanks). If the A-46 capacity calculated in this manner is not greater than the A-46 input, then the analyst should use the procedures in the GIP directly since it is likely that a larger A-46 capacity will be found.

The factor of 1.875 above was obtained by examining the various requirements for calculating anchorage and equipment functionality capacities for both the SMA and A-46 programs (1,3). It was found that the largest reduction factor occurs for anchor bolt capacity for the case of pure shear. As stated above the factors of safety for anchor bolt shear capacity are 2.0 and 3.0, respectively for the SMA and A-46 requirements. However, to make the comparison compatible when using in-structure spectra generated with median-type methods, the A-46 input must be multiplied by a factor of 1.25. Thus, the conversion factor is 3/2 * 1.25, or 1.875. If conservative design-type methods are used in the A-46 evaluation, the 1.25 factor can be eliminated given a total factor of 1.50.

Note that it may be possible to develop more realistic A-46/SMA reduction factors by examining the specific factors of safety for each type of review. For example, for tension in expansion anchors the limiting factor of safety of 3.0 is applicable for both A-46 and SMA reviews for single bolts when hairline concrete cracks are considered unlikely. For the case when realistic median-centered methods are used dividing the HCLPF capacity by 1.25 (i.e., 1.25 * 3.0/3.0) to obtain the A-46 capacity is more realistic.

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Appendix D

CONTAINMENT PERFORMANCE REQUIREMENTS

by

David R. Buttemer

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Appendix D

CONTAINMENT PERFORMANCE REQUIREMENTS

INTRODUCTION

The current seismic margin assessment (SMA) methodology (Reference (1), developed in 1988 and revised in 1991) focuses on the verification of success paths necessary to prevent core damage and intentionally does not address containment functions/systems necessary to mitigate the consequences should core damage occur. Indeed, the success path logic diagrams developed in Reference (1) identify alternate means (e.g., systems) of achieving a stable long term safe shutdown (i.e., core damage prevention) following a seismic margin earthquake. In late 1988, the U.S. Nuclear Regulatory Commission (NRC) issued to all licensees generic letter (GL) 88-20, which addressed individual plant evaluation (IPE) requirements. This letter indicated that the IPE should follow NRC's "defense-in-depth" principle and should: (1) examine core damage prevention, and (2) conduct a containment performance analysis (following the general guidance for the "back-end" analysis given in Appendix 1 of GL 88-20). The GL 88-20 indicated that the initial IPE should address internal events only and that external events (including earthquakes) would "proceed separately on a later schedule." In late 1989 the NRC staff issued a draft letter extending the IPE process to include external events (i.e., the IPEEE); earthquakes were identified as a major class of external events. In 1990, the NRC issued draft documentation providing guidance for conducting an IPEEE and what is to be included in the licensees submittal. The purpose of the Appendix is to describe the proposed industry approach to reviewing containment performance for the seismic portion of the IPEEE process.

The system considerations, which lead to the selection of structures and equipment necessary for containment performance, are given in the next section. This is followed by guidance for performing a seismic capability assessment of the elements included in the selected containment success paths.

SYSTEMS CONSIDERATIONS

The methodology for identifying the systems required to assure a long term safe shutdown condition following a seismic margin earthquake (SME) with a safely

cooled core are described in Reference (1); by "long term", the evaluation required that a stable hot or cold shutdown condition be maintained for at least 72 hours.

The major systems related assumptions which were made were that the SME failed offsite power (and that no credit be taken for offsite power recovery), that a seismically induced small LOCA should be considered unless plant specific evaluation could rule it out, and that seismically induced electrical relay and contactor chatter needed to be evaluated. This Reference (1) methodology explicitly excluded consequence mitigation (read "containment") systems from the evaluation process. The relay chatter evaluation addresses the operability of those systems necessary to assure a long term safe shutdown condition, but also addresses the potential of relay chatter causing normally closed isolation valves that separate high and low pressure primary coolant lines to open. If such an event were to occur, it would cause overpressurization and could potentially fail the low pressure piping, resulting in what is referred to as an interfacing system LOCA (or the classical V-sequence in WASH-1400 parlance), which results in containment bypass and could cause core damage.

Before discussing the containment functions and systems that should be addressed in a seismic margin assessment, it is informative to consider what is typically done for an internal events containment evaluation. The "back-end" IPE analysis typically addresses a number of containment functions (including isolation, sprays, and long term heat removal), as well as the potential for containment bypass events. Ice condenser and Mark II/IIIs also address hydrogen control (e.g., ignitors) and recent NRC guidance to Mark I licensees (Supplement 1 to the GL 88-20, a result of the NRC Containment Performance Improvement Program) suggests enhancements to drywell spray and the ability to vent the torus. Clearly, the potential environment and demands on containment systems during a severe accident may exceed the original design basis. Superimposed on the containment systems issue are a large number of very complex phenomenological issues that currently have large uncertainties. These phenomena include in-vessel and ex-vessel fuel-coolant interactions, direct containment heating, hydrogen burning/detonation, core-concrete interactions with noncondensible gas production, basemat and/or liner melt-through, etc.

More recently (Supplement 3 to the GL 88-20, dated July 6, 1990, which announced the completion to NRC's Containment Performance Improvement Program), the NRC provided additional insights for PWR containments and BWR Mark II and Mark III

containments. Specifically, this letter addressed for BWR Mark II containments the pros and cons of suppression pool venting and the possibility that core debris may fail the downcomers or drain line and result in suppression pool bypass. For BWR Mark III and PWR ice condenser containment designs, the letter pointed out potential vulnerabilities of these plants wherein an extended station blackout could result in core damage and hydrogen accumulation because of unpowered ignitors and, should electric power be restored later, the ignitors could ignite detonable concentrations of hydrogen. The letter alludes to the use of an alternate AC option to the Station Blackout rule to ensure uninterrupted ignitor operation. The letter describes concern for local detonable hydrogen mixtures developing in large dry PWR containments or of a globally detonable hydrogen mixture developing in smaller subatmospheric PWR containments.

As described above, there are a large number of containment systems and containment phenomena that have an impact on containment performance. It can be generally stated that the original containment design criteria did not specifically address the full ramifications of a severe accident, which may result in vessel melt-through and significant amounts of molten core debris on the containment floor. Clearly, the complex phenomenological issues related to containment performance in severe accidents are beyond the scope of a SMA. It is recommended that the SMA demonstrate with a high degree of confidence that those containment related functions that are necessary to prevent early containment failure survive the SME (early means roughly the 12 hours following the seismic margin earthquake (SME)). For essentially all containment designs, this includes: (1) successful containment isolation, (2) that the SME does not fail the containment structural integrity (including penetrations and closures), and (3) the SME (or seismically induced relay chatter) does not result in containment bypass (this is in the current SMA methodology). For large, dry PWR containments with only successful isolation, containment failure would not be expected for several 10s of hours and containment spray and/or fan coolers are generally unimportant in the short term. The SMA already addresses long term heat removal for BWRs, but does not address drywell sprays (for Mark I) or hydrogen control (for uninerted Mark II, Mark III or ice condenser containments). These systems may be required to prevent early containment failure. Because there is a significant variation in containment designs, it is suggested that the IPE internal events containment evaluation team be consulted to determine which systems are required (not just desired) to assure early containment integrity following a SME.

The success path logic diagrams (SPLDs) describe in Reference (1) denote those systems which are necessary to provide a long term safe shutdown condition. These SPLDs should be extended to include early containment performance requirements following a SME and assuming that severe core damage has occurred. An example SPLD extension is shown in Figure D-1 for no plant in particular. Clearly, post SME containment isolation and containment integrity must be demonstrated along with assurance that relay chatter can not cause normally closed isolation valves to opening and induce an interfacing system LOCA.

Care must be exercised as to what active equipment or systems are required (not just desired) to prevent early containment failure. For example, in a large dry PWR containment, fan coolers may not be designed to operate in the harsh environment associated with a severe accident (and indeed, nonsafety class coolers generally trip on a safety signal) and the effectiveness of containment sprays in providing debris bed cooling depends on the sprayed water having access to the reactor cavity area. Another example would be the reactor building and standby gas treatment system in a BWR Mark I containment. Clearly, neither one of these has an impact on primary containment failure and, depending on plant specific design features, may not have much influence on consequence mitigation if containment failure occurs. For pressure suppression containment designs the evaluation must demonstrate that equipment necessary for the pressure suppression function (e.g., the suppression pool and the vent system for BWRs or the ice buckets, ice chamber and inlet/outlet "doors" for ice condenser designs) survives the SME. The SMA team should solicit guidance from the IPE containment evaluation team as to what functions are required to prevent early containment failure.

SEISMIC CAPABILITY ASSESSMENT

As discussed in Section 2 of this report it is recommended that potential containment failure be reviewed for those plants which elect to perform a full- or focused-scope deterministic review. The identification of structures and equipment in the containment function success path should be based on the guidance given in this appendix. Thus, the structures and equipment selected in the success path will be the same for all full- or focused-scope plants.

In performing a deterministic review (i.e., full- or focused-scope SMA) all elements in the success path should be evaluated according to the requirements in Section 3 of this report, in the same manner as for elements required to prevent

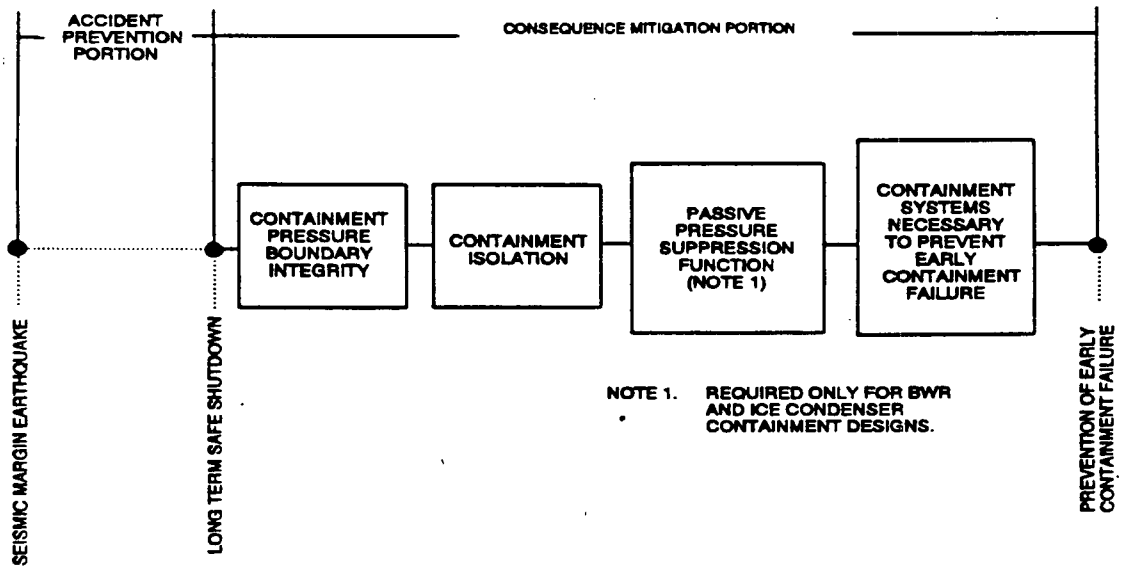

Figure D-1. Success Path Logic Diagram for Containment Performance Evaluation

core damage. Guidance for performing IPEEE in conjunction with an A-46 review, is given in Section 4.

The guidelines in EPRI report NP-6041 (1) are for seismic margin assessment to prevent a core damage accident. However, the same element types considered in that report also include structure and equipment types which will be selected in the containment success path. Requirements for both steel and concrete containments are provided in Section 6 of that report. In general, containment structures are rugged and a review of construction drawings and the design-basis calculations can be performed to screen out the containment structure or to identify features which require additional consideration. Penetrations need to be carefully reviewed to verify that relative motions between structures can be accommodated.

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Appendix E

DETERMINATION OF REVIEW-LEVEL GROUND MOTIONS FOR SEISMIC IPE CLOSURE

by

Robert T. Sewell
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CONTENTS

Appendix E

DETERMINATION OF REVIEW-LEVEL GROUND MOTIONS FOR SEISMIC IPE CLOSURE

INTRODUCTION

The tasks of determining whether or not a screened-in component is a potential vulnerability, and of deciding how a potential vulnerability should be treated, fall within the scope of seismic-IPE closure procedures. As discussed in Section 6, the determination of a set of review-level ground motions (RLGMs) is a key element of the seismic-IPE closure guidelines and framework. The purpose of this appendix is to outline the technical procedures for the development of plant-specific RLGMs; the development is consistent both with the use of focused-/full-scope SMA methods and with risk-based closure criteria presented for the internal-events IPE (1). The RLGMs themselves can be found in Reference (2) for plants for which EPRI-SOG seismic hazard results have been computed.

Section 3 and Reference (3) provide guidelines for identifying SMA outliers and for determining component HCLPF capacities of these outliers. The median 5%-damped NUREG/CR-0098 (4) spectrum is recommended in Section 3 as the basis for seismic input in HCLPF calculations when determining remaining outliers. This characterization of seismic input, however, is known to be substantially conservative for the majority of sites located in the central and eastern U.S. Because closure criteria ultimately rely on assessment of cost-benefits (i.e., costs incurred versus benefits achieved), it is important to have (in contrast to the NUREG/CR-0098 spectrum) a realistic, site-specific characterization of the ground-motion input. The uniform hazard spectrum (UHS) provides this realistic input characterization and, when tied to the appropriate exceedance frequency, describes the plant HCLPF capacity required to satisfy a specified safety target measured in terms of core-damage (or plant-damage-state) frequency. The following discussion pertains primarily to selecting appropriate exceedance frequencies for anchoring ordinates of the UHS.

APPROACH FOR DETERMINING REVIEW-LEVEL GROUND MOTION

Basis of Approach

To determine a RLGM, it is necessary to establish an appropriate safety criterion.

Severe-accident-issue closure guidelines for the IPE (1) specify such safety criteria. These criteria are consistent with a total mean core-damage frequency of 1×10^{-4} (or less), which is based on current recommendations (5, 6) for implementation of NRC safety-goal policy (7). As presented in Section 6 of this document, industry's IPE closure framework employs a graded set of decision elements with decision criteria expressed in terms of percentages of total core-damage frequency for major functional accident sequence groups; the greater the percent contribution, the greater the scope in cost-benefit comparison. For instance, in the IPE core-damage evaluation process: if the contribution to total core-damage frequency of a major functional accident sequence is less than 5%, no action is required; if the contribution is between 5% and 20%, a cost-effective change to utility accident management guidelines (UAMGs) only is considered; if the contribution is between 20% and 50%, cost-effective procedural and minor hardware changes are considered, in addition to UAMG changes; otherwise, if the contribution is greater than or equal to 50%, cost-effective changes in design, normal and emergency operating procedures, and UAMGs are all considered. For a consistent development here, three RLGMs (for any given plant) are needed for seismic-IPE closure, corresponding to safety targets of 5.0×10^{-5} (50% contribution to the total core-damage-frequency safety target of 1.0×10^{-4}) 2.0×10^{-5} (20%), and 0.5×10^{-5} (5%). Each of these three plant-specific RLGMs can be obtained using the procedure outlined below. Reference (8) outlines, in greater detail, the analysis procedure and justification for evaluating the safety-target-consistent RLGMs. As denoted in Section 6 (Figure 6-3), the review-level ground motions for mean[1] core-damage frequency safety targets of 5.0×10^{-5}, 2.0×10^{-5} and 0.5×10^{-5}, are termed RLGM-A, RLGM-B and RLGM-C, respectively.

For review-motion determination, one asks what is the minimum plant seismic capacity which insures that the particular seismic core-damage-frequency criterion is met. This safety-based seismic capacity determines a review-level-earthquake

[1]As an alternative to the use of a mean core-damage safety target, one may use a median measure that is consistent with the mean target. For determining a consistent median measure, results of seismic core-damage frequency distributions obtained in past risk studies can be used. NUREG-1150 (9), for instance, provides mean and median core-damage frequency results for two nuclear power plants; results are given separately for EPRI and LLNL seismic hazards. These results suggest that a mean-to-median ratio between 5.0 and 6.0 is appropriately representative of seismic core-damage-frequency distributions obtained from modern analyses. Such a factor would imply, for instance, that a median core-damage frequency target of about 4×10^{-6} is consistent with a 2×10^{-5} mean safety target.

(RLE) spectrum, used as a basis in obtaining a RLGM spectrum, for which a given plant may be reviewed in its seismic IPE. If the seismic IPE reveals that the plant seismic capacity equals or exceeds the review level, then the plant is shown to have adequate seismic resistance; consistent with safety objectives, the plant satisfies an acceptable (and conservative) safety criterion. For such a case, consideration can justifiably be focused away from major, expensive safety enhancements (such as design, hardware, and major procedural changes) without explicitly appealing to cost-benefit considerations. The seismic IPE closure process may reveal obvious cost-effective, simple upgrades (e.g., anchorage strengthening or installation) to improve seismic resistance; if a review level cannot be met after these obvious fixes, then cost-benefit analysis, consistent with NRC backfit policy (10), ascertains the extent of any further consideration of potential vulnerabilities. Using this approach for determination of the RLE spectrum and implementation of seismic-IPE closure guidelines is consistent with objectives of NRC severe-accident policy (11).

The general approach for obtaining a plant-specific review-level spectrum is outlined below:

- Select a target mean seismic core-damage frequency safety guideline (e.g., 2×10^{-5}).
- Obtain mean hazard curves, or obtain pseudo-mean curves by converting from median hazards[2]. Hazard curves should be obtained for spectral acceleration at appropriate (dynamic) response frequencies (i.e., 1, 2.5, 5 and 10 Hz).
- Assume a trial seismic capacity (i.e., in terms of seismic fragility curve parameters).
- Compute a mean seismic core-damage frequency corresponding to the trial capacity, given the seismic hazard.
- Iteratively adjust the seismic capacity to just meet the assumed target mean seismic core-damage frequency guideline.

[2]This document recommends the use of EPRI mean seismic hazard results. If, however, both EPRI and LLNL results are used, it would be appropriate to start with median hazard curves to achieve the greatest consistency among use of EPRI and LLNL analyses. Since safety comparisons are based on mean core-damage frequency, though, one would need to convert the median curves to equivalent or pseudo means. For this conversion, a representative variation (with ground-motion) of uncertainty in seismic hazard, characterized by the logarithmic standard deviation in hazard (at a given ground motion), β_H, can be used. The equivalent mean hazard is obtained by multiplying the median hazard by the factor $\exp(0.5\beta_H^2)$. A representative value of β_H, an average for ground motions between the typical plant HCLPF and median capacities, might be 1.0 or so.

- Find the mean seismic hazard (exceedance frequency) corresponding to the adjusted seismic capacity.

- Perform the above steps for 1 Hz, 2.5 Hz, 5 Hz and 10 Hz spectral acceleration hazard curves, and average the resulting hazard levels.

- Establish the RLE spectrum as the mean (or equivalent/pseudo mean) uniform hazard spectrum evaluated for the level of hazard determined above; alternatively, use the adjusted seismic capacities themselves, at 1 Hz, 2.5 Hz, 5 Hz and 10 Hz, to define the RLE spectrum.

No assumptions concerning plant median HCLPF capacity nor concerning resistances of plant structures and components are made in formulation of a RLE (i.e., target HCLPF) spectrum. Site-specific seismic hazard results are the key element required to implement the methodology. The RLE spectrum can, in fact, be simply thought of as an alternate description of the seismic hazard, i.e., it is a characterization of the seismic hazard that has direct significance to an acceptable plant core-damage risk.

Description of Approach

Specific details describing particular elements of methodology involved in assessing the RLE spectrum are discussed below:

Seismic hazard description. Available spectral acceleration seismic hazard curves for response frequencies ranging from 1 to 10 Hz are obtained. Ground motion frequencies in the range of 2 to 15 Hz, and in particular from 2 to 10 Hz, have been found in past seismic studies to have the most significant effect on critical plant structures and components. PGA, although used as a ground motion descriptor in past SPRAs, is a poor indicator of earthquake damageability to nuclear power plants, and is sensitive to high-frequency components of ground motion that are generally believed to be non-damaging. Determination of ground motions for seismic severe accident review should, therefore, be keyed to frequencies in the range of 2 to 10 Hz.

Seismic capacity description. The RLE analysis characterizes plant capacity by use of a family of seismic core-damage fragility curves. A core-damage fragility curve conveys, for all values of a given ground motion parameter (e.g., 10-Hz spectral acceleration), the likelihood or frequency of core damage, conditional upon the given ground motion parameter value. (A family of core-damage fragility curves is used to represent the uncertainty in the true core-damage fragility). A

convenient way to characterize the core-damage fragility family is by means of the familiar lognormal parameters $\breve{A}$, β_R, and β_U.

Examination of published, peer-reviewed PRAs reveals that values of β_R and β_U vary little from plant-to-plant. The results of plant-level values of β_R and β_U as derived from six PRA studies for eight reactor units, indicates plant-to-plant averages for β_R and β_U respectively, of 0.22 and 0.24. The plant-to-plant coefficients of variation in β_R and β_U are extremely small (0.14 and 0.16, respectively), indicating that the average values are sufficient as generic characterizations that may be used to describe the shape of plant-level fragility curve families.

In addition to parameters $\breve{A}$, β_R, and β_U, the HCLPF capacity can be obtained from the family of fragility curves. The core-damage HCLPF at a given frequency[3] f is given by the following expression (that is strictly valid only for lognormal plant fragilities):

$$HCLPF_f = \breve{A}_f \exp[-1.65(\beta_R + \beta_U)] \qquad \text{(E-1)}$$

When a HCLPF is determined in this manner and is used as an "anchor point" of a review earthquake spectrum, it is said to be a HCLPF reported at the 50% non-exceedance probability [denoted here as HCLPF(50%)]. HCLPF capacities are by convention, however, reported at an 84% nonexceedance probability. To convert from HCLPF(50%) to HCLPF(84%) we use the following expression (see Reference 12):

$$HCLPF(84\%) = HCLPF(50\%) \times e^{\beta_{pp}} \qquad \text{(E-2)}$$

where β_{pp} is the logarithmic standard deviation in spectral response due to peak-to-peak randomness. A typical value of β_{pp} has been estimated in past studies to be about 0.18. Assuming this value, HCLPF(84%) is obtained as:

$$HCLPF(84\%) \approx 1.2 \times HCLPF(50\%) \qquad \text{(E-3)}$$

[3]In the past, HCLPF capacities have typically been expressed in terms of a PGA value. Because of the drawbacks of PGA noted above, however, it is preferable to report a HCLPF capacity at some (response-significant) spectral frequency f. Consequently, in this study, when describing a HCLPF capacity (or core-damage fragility), it is necessary to specify a corresponding spectral frequency f used as the basis for its evaluation.

Mean seismic core-damage frequency assessment. Given the seismic hazard and fragility-shape descriptions, the primary effort in evaluating a RLE spectrum is conducting the many numerical determinations of mean seismic core-damage frequency. This numerical procedure is well documented in the seismic risk analysis literature (see, for instance, Reference 13).

In the determination of RLE values, a trial HCLPF is selected as the basis for scaling the fragility curve family. Once the trial HCLPF is selected, the fragility family is scaled by determining the corresponding value of $\check{A}$, as follows:

$$\check{A} = \text{HCLPF} \exp[1.65(\beta_R + \beta_U)] \qquad \text{(E-4)}$$

where parameters β_R and β_U that govern fragility shape are assumed to have the generic values of 0.22 and 0.24, respectively, as noted above. These values of $\check{A}$, β_R and β_U provide a complete description of the fragility family. This fragility family is integrated with seismic hazard curves of spectral acceleration at 1 Hz, 2.5 Hz, 5 Hz and 10 Hz to obtain estimates of mean seismic core-damage frequency.

Iterative determination of RLE values. The frequency-dependent plant-level HCLPF is iteratively adjusted, resulting in an iterative scaling of the seismic fragility, and corresponding hazard-fragility integrations are performed until a mean seismic core-damage frequency equal to the target seismic core-damage safety guideline is obtained. The HCLPF value for which the target safety guideline is just met is the minimum required plant HCLPF or the RLE value. This HCLPF is reported at the 50% nonexceedance probability; to report the required HCLPF at a 84% nonexceedance probability, Eq. E-3 is used.

RLE values are determined in this iterative manner for spectral frequencies of 1 Hz, 2.5 Hz, 5 Hz and 10 Hz. Once generic values of β_R and β_U are assumed, the RLE values are dependent entirely on the site seismic hazard and the chosen level of the core-damage safety guideline. No assumptions of median plant capacity are made in the assessment of the RLEs.

Determination of RLE spectrum. To obtain a RLE spectrum, mean seismic hazards for 1 Hz, 2.5 Hz, 5 Hz and 10 Hz spectral accelerations are evaluated, correspondingly, at the frequency-dependent RLE values. These levels of mean seismic hazard are averaged, in recognition of the fact that seismic plant response is governed primarily by ground-motion input energy between 2 and 10 Hz

(i.e., critical plant structures and components are sensitive to motions in this frequency band).

This resulting average hazard value from the frequency-dependent RLE evaluations is used to determine the RLE spectrum. The plant-specific RLE spectrum is constructed as the mean UHS evaluated at this average level of hazard. Alternatively, the frequency-dependent RLE values ($HCLPF_f$) may themselves be used directly as ordinates of the RLE spectrum.

Determination of review-level ground motion. High-frequency ground-motion input has generally been shown to have little potential for damaging equipment in nuclear power plants. As discussed in Appendix B, the high-frequency portion of a RLE spectrum may be reduced, therefore, when determining the input motion for severe-accident evaluations. This reduction should occur before making comparisons, for instance, with seismic-IPE closure criteria. The specific procedure for reducing the high-frequency input is presented and discussed in Reference (14). This procedure makes use of spectral ordinates for different damping values, and hence, RLE spectra must be obtained for multiple damping levels. With this reduction of high-frequency input, the RLE spectra for the various dampings are converted to a review-level ground-motion (RLGM) for use in severe-accident analysis.

When used as a basis for performing the seismic IPE, the RLGM insures that plants are evaluated for consistent safety levels. Being constructed as a target plant HCLPF spectrum, the RLGM serves as a natural basis in implementing a seismic margin assessment. The result of a SMA is an evaluation of a plant-level HCLPF capacity. If a SMA is conducted for a plant, and the plant is found to have a HCLPF at a level that meets or exceeds the RLGM, then the plant has met the safety criterion associated with the RLGM. The comprehensive walkdown performed in the SMA assures that any potential vulnerabilities that may compromise seismic resistance will be uncovered.

RLGM RESULTS FOR IMPLEMENTING SEISMIC IPE CLOSURE GUIDELINES

The above procedures have been implemented to obtain RLE spectra and RLGM spectra for 58 eastern U.S. plant sites, based on EPRI mean hazard curves. The resulting RLE spectra (for 5% damping) are assessed and presented in Reference (8) for RLE-A, RLE-B and RLE-C; corresponding RLGM spectra (i.e., after high-frequency reduction) are presented in Reference (2). To simplify their use by licensees, each RLGM has been enveloped, over the vibration frequencies of interest (i.e., 2

to 10 Hz), by the NUREG/CR-0098 5%-damped median spectrum, and the associated PGA values have been obtained, as indicated in Table E-1. Hence, for use in closure criteria, licensees can simply compare component HCLPF-PGA values (based on the NUREG/CR-0098 5%-damped median shape) assessed in their seismic IPE, against the PGA values of Table E-1. Alternatively, a new component HCLPF can be computed based on the RLGM spectral shape itself, as provided in Reference (2).

Table E-1

SAFETY GOAL BASED REVIEW LEVEL GROUND MOTION PGA VALUES
(BASED ON NUREG/CR-0098 SPECTRAL SHAPE); EPRI MEAN HAZARD INPUT

Plant No.	RLGM-PGA Values RLGM-A	RLGM-B	RLGM-C	Plant No.	RLGM-PGA Values RLGM-A	RLGM-B	RLGM-C
01	0.18	0.25	0.40	31	0.10	0.14	0.25
02	0.16	0.23	0.37	32	0.14	0.20	0.30
03	0.08	0.12	0.19	33	0.08	0.13	0.22
04	0.11	0.16	0.26	34	0.02	0.05	0.08
05	0.25	0.35	0.53	36	0.02	0.04	0.07
06	0.13	0.19	0.31	38	0.07	0.11	0.17
07	0.11	0.16	0.28	39	0.08	0.13	0.22
09	0.13	0.19	0.29	40	0.05	0.07	0.11
10	0.13	0.19	0.29	41	0.02	0.04	0.07
11	0.13	0.20	0.36	42	0.04	0.06	0.10
12	0.13	0.19	0.28	43	0.02	0.04	0.07
13	0.07	0.11	0.18	44	0.02	0.04	0.06
14	0.06	0.08	0.13	45	0.07	0.11	0.17
15	0.06	0.08	0.13	47	0.14	0.19	0.29
16	0.06	0.10	0.16	48	0.05	0.07	0.13
17	0.08	0.12	0.22	49	0.08	0.14	0.26
18	0.10	0.14	0.26	50	0.06	0.08	0.17
19	0.08	0.13	0.24	51	0.08	0.13	0.26
20	0.11	0.16	0.24	52	0.08	0.16	0.31
21	0.13	0.18	0.28	57	0.05	0.08	0.13
22	0.13	0.19	0.31	58	0.12	0.18	0.31
23	0.13	0.19	0.31	59	0.05	0.07	0.12
24	0.06	0.08	0.14	60	0.13	0.19	0.32
25	0.13	0.20	0.35	61	0.05	0.08	0.14
26	0.10	0.13	0.22	62	0.05	0.07	0.13
27	0.12	0.17	0.30	67	0.07	0.11	0.19
28	0.11	0.16	0.28	68	0.07	0.10	0.17
29	0.13	0.19	0.32	69	0.10	0.13	0.23
30	0.14	0.19	0.29	70	0.06	0.08	0.16

Although the use of EPRI mean seismic hazard curves is recommended in obtaining RLGM results, one may elect to consider both EPRI and LLNL seismic hazard curves

in implementation of the seismic IPE. In such a case, to obtain the most consistent result, one would use equivalent/pseudo-mean hazard curves and spectra derived from each (EPRI and LLNL) set of median hazard curves.

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