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

030089

JLP/bgr

1/1

Attachment

A011

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.

Mary Lou Skrocki

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.

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

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

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

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

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.

<u>Question</u>

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 ($F_2 < 1E\text{-}6$), 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 $F_3 < 1E\text{-}6$.

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 $F_2 < 1E\text{-}6$.

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	0.0311	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).

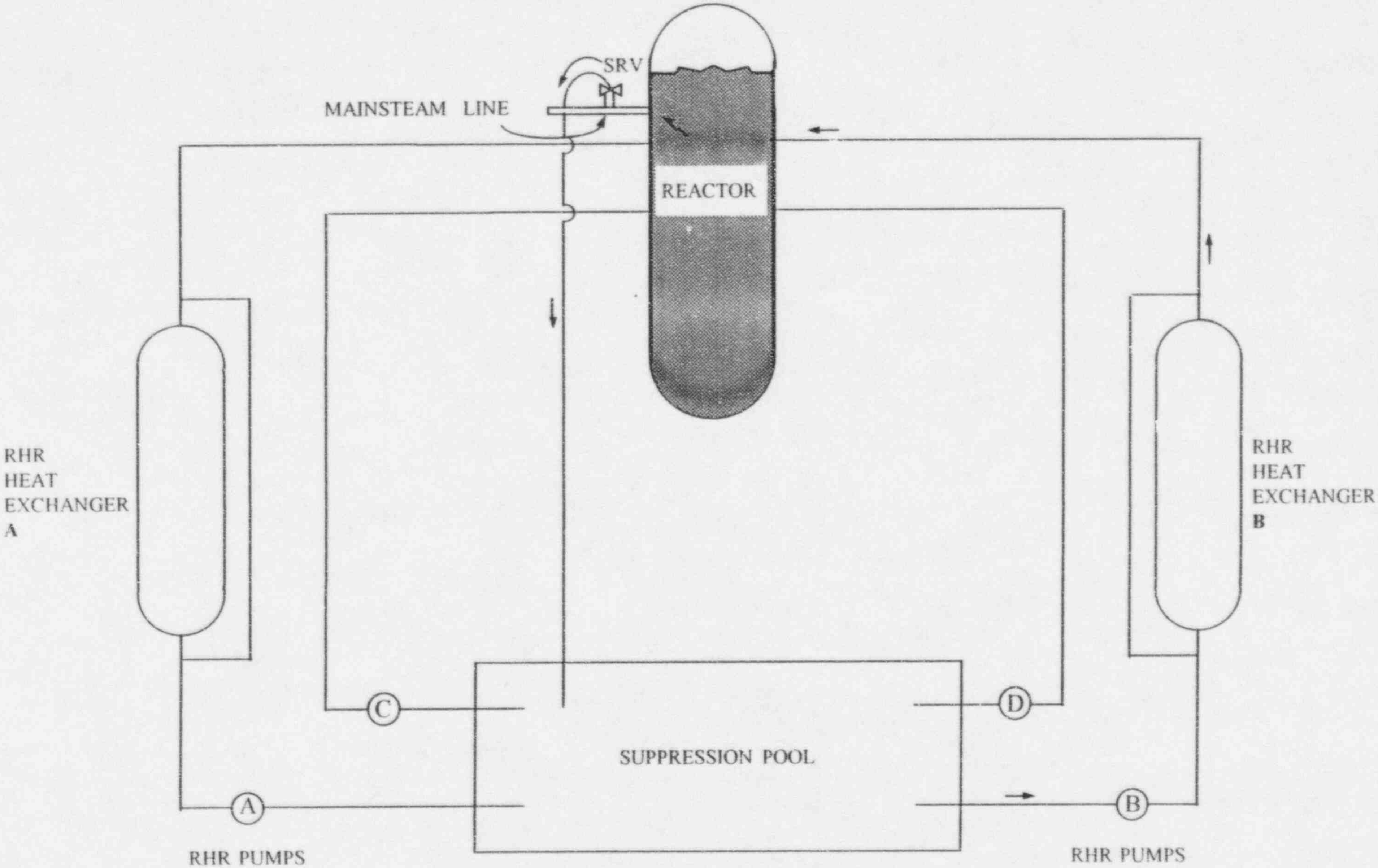
Question

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.

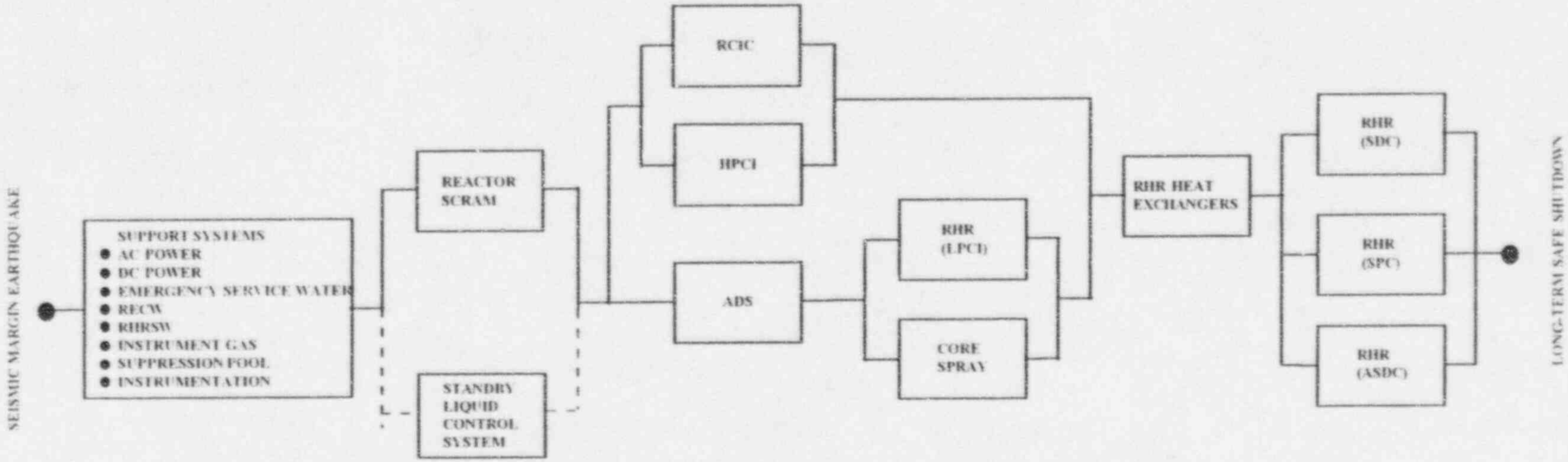
Response

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

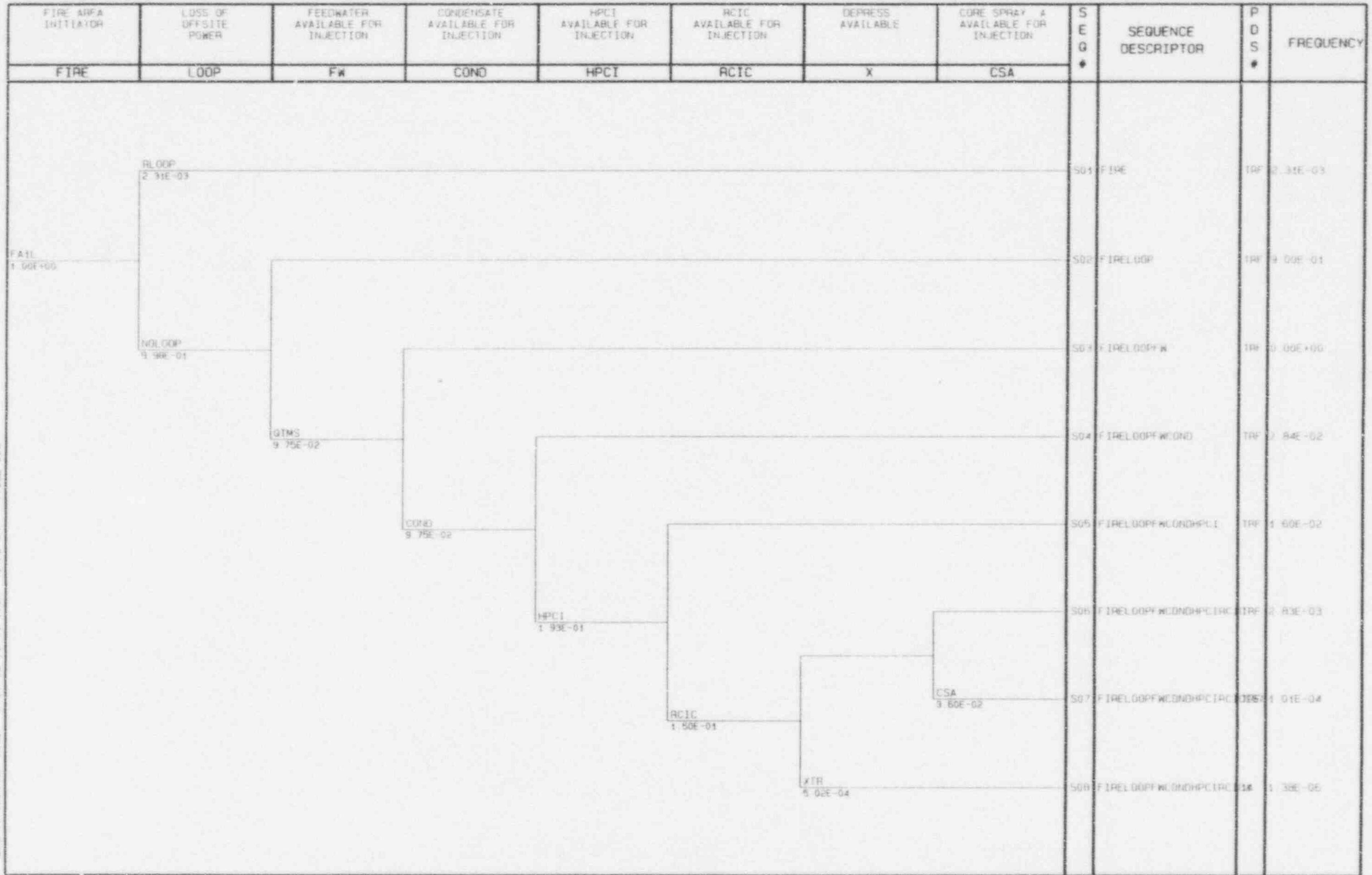


ALTERNATE SHUTDOWN COOLING



Limerick Success Path Logic Diagram

Figure S4.1



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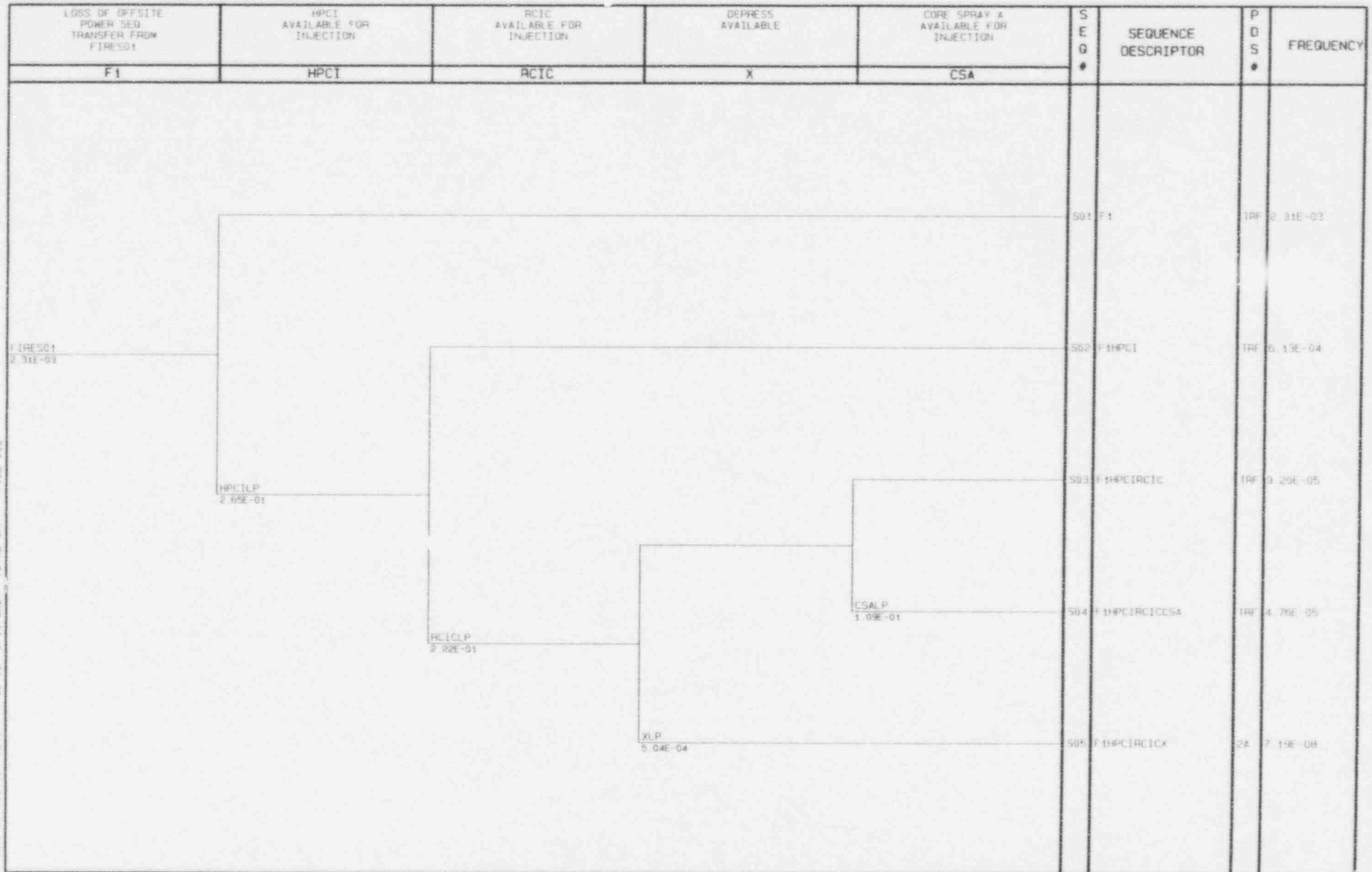
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

INITIAL FIRE EVENT TREE



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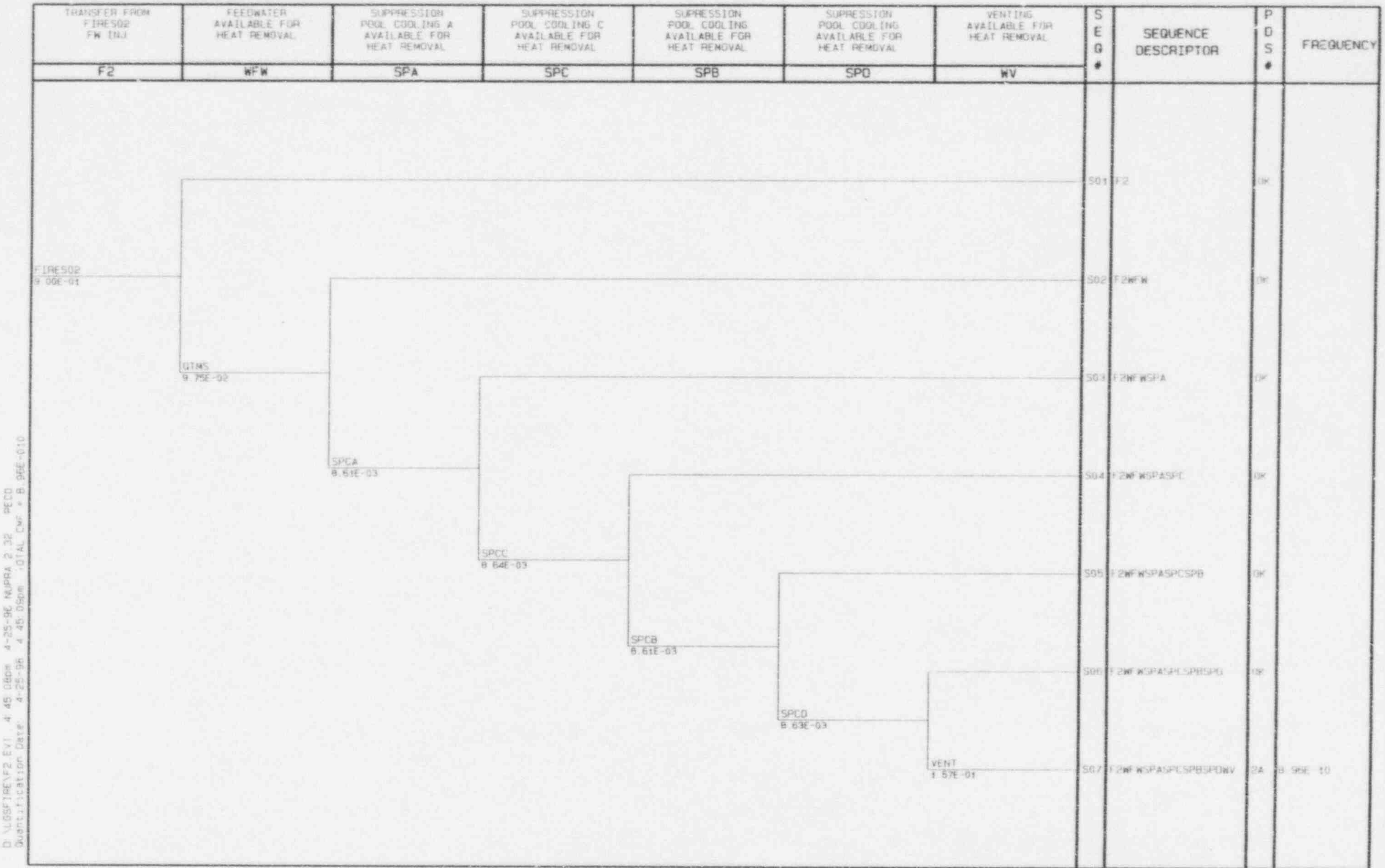
Quantification Date 4-25-96 5:11:32pm TOTAL CMF = 7.19E-008

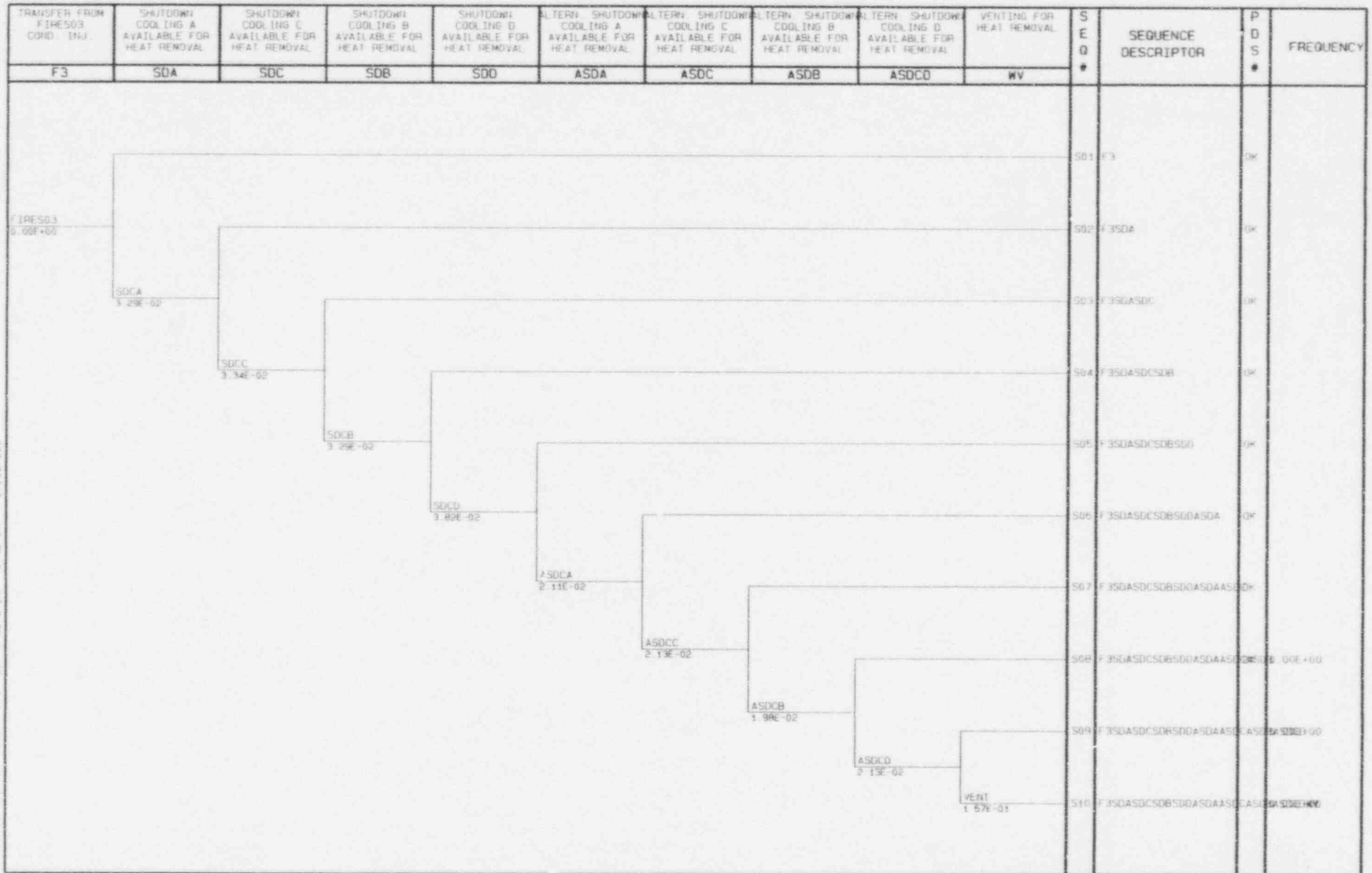
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

LOSS OF OFFSITE POWER SEQUENCES
TRANSFER FROM FIRES01



FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM FIRES02
FEEDWATER INJECTION SUCESSFUL



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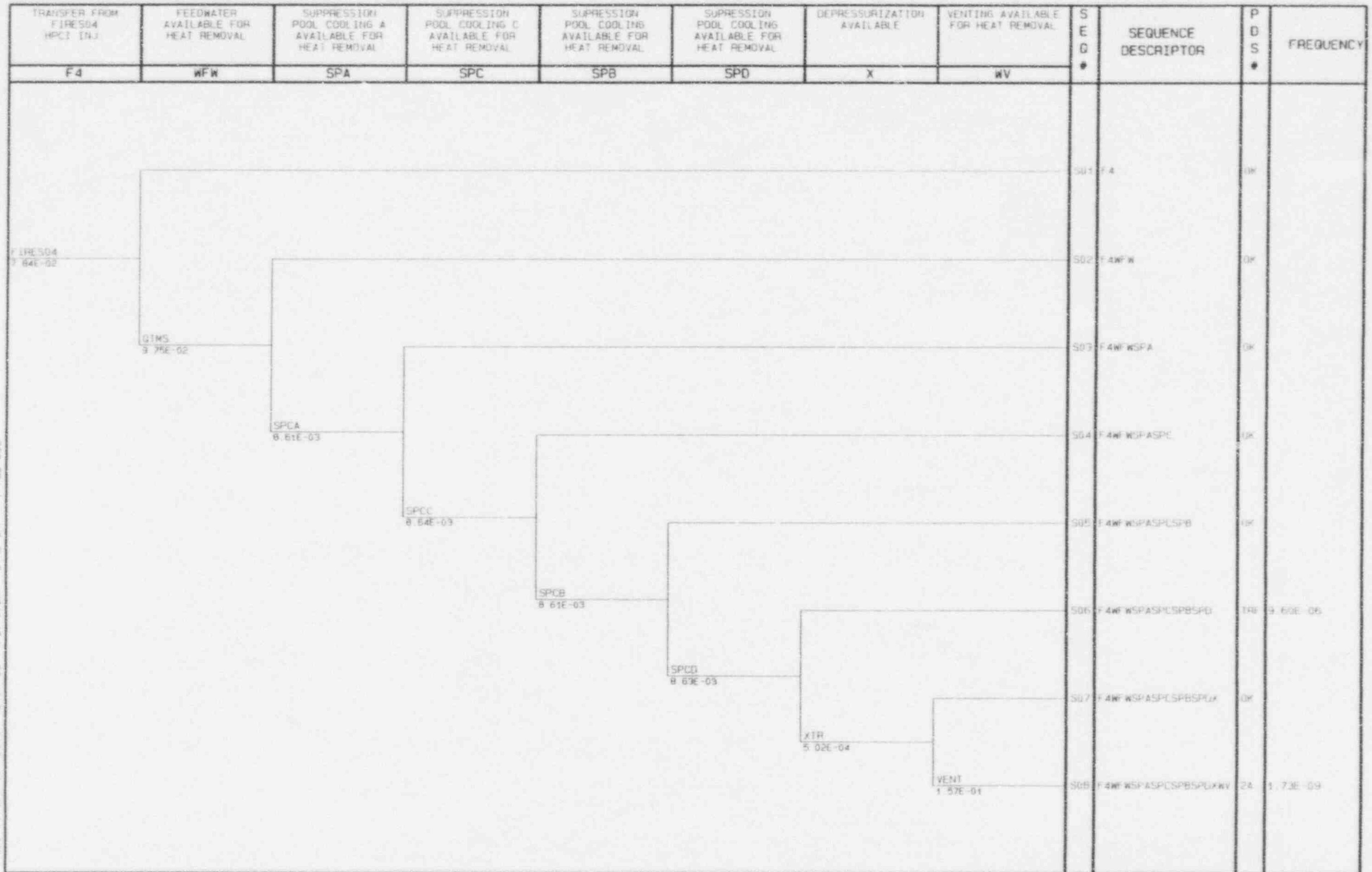
Quantification Date: 6-07-96 4:56.27pm TOTAL CMF = 0.00E+000

FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM FIRES03
CONENSATE FOR INJECTION



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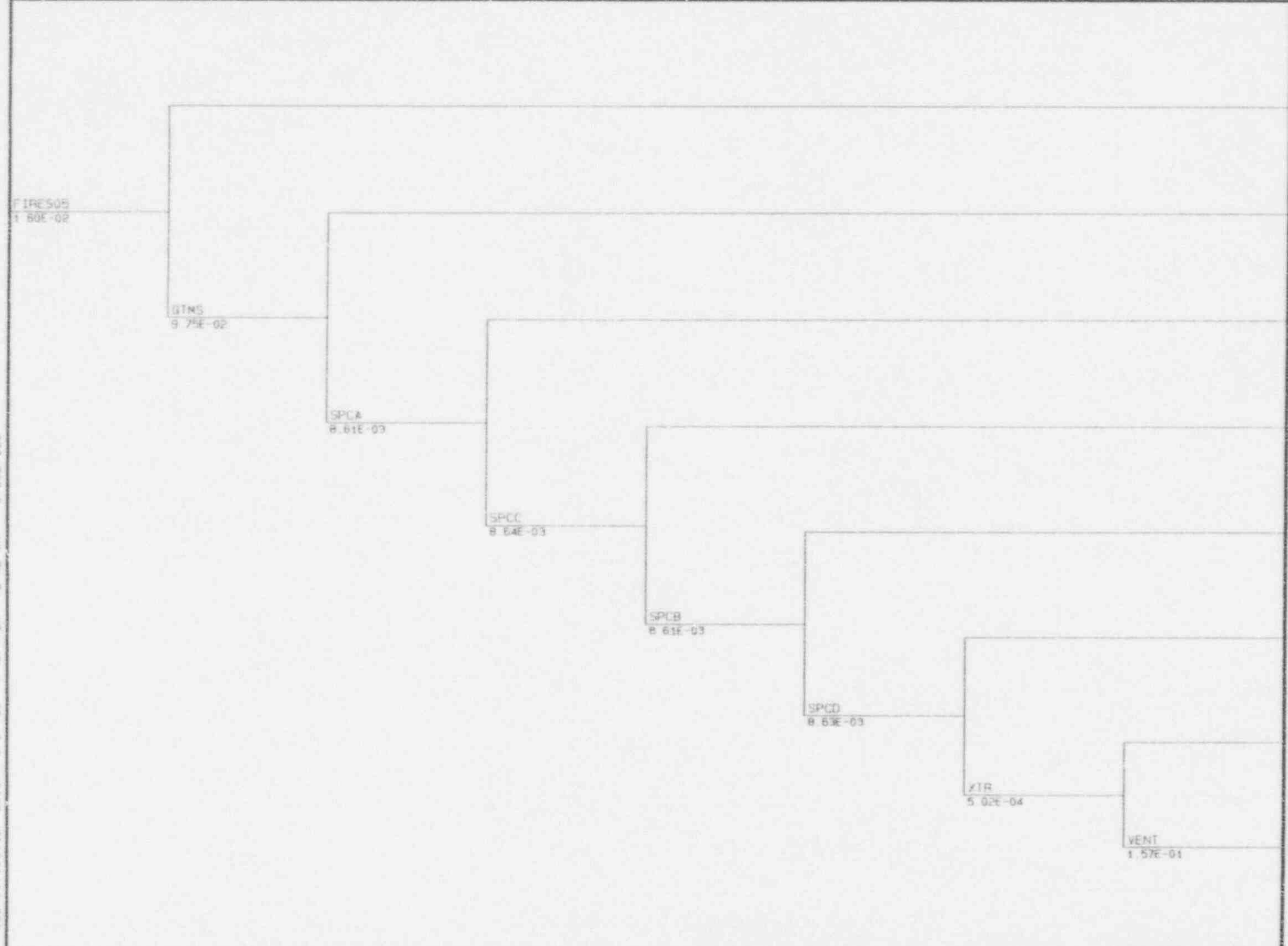
Quantification Date 6-07-96 4:37:57pm TOTAL CMF = 1.73E-009

FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM FIRES04
FEEDWATER INJECTION SUCESSFUL

TRANSFER FROM FIRES05 RCIC INJ	FEEDWATER AVAILABLE FOR HEAT REMOVAL	SUPPRESSION POOL COOLING A AVAILABLE FOR HEAT REMOVAL	SUPPRESSION POOL COOLING C AVAILABLE FOR HEAT REMOVAL	SUPRESSION POOL COOLING AVAILABLE FOR HEAT REMOVAL	SUPRESSION POOL COOLING AVAILABLE FOR HEAT REMOVAL	DEPRESSURIZATION AVAILABLE	VENTING AVAILABLE FOR HEAT REMOVAL
F5	WFW	SPA	SPC	SPB	SPD	X	WV



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S02	F5WFW	OK	
S03	F5WFWSPA	OK	
S04	F5WFWSPASPC	OK	
S05	F5WFWSPASPCSPB	OK	
S06	F5WFWSPASPCSPBSPD	TRF	1.88E-06
S07	F5WFWSPASPCSPBSPDX	OK	
S08	F5WFWSPASPCSPBSPDXWV	2A	0.00E+00

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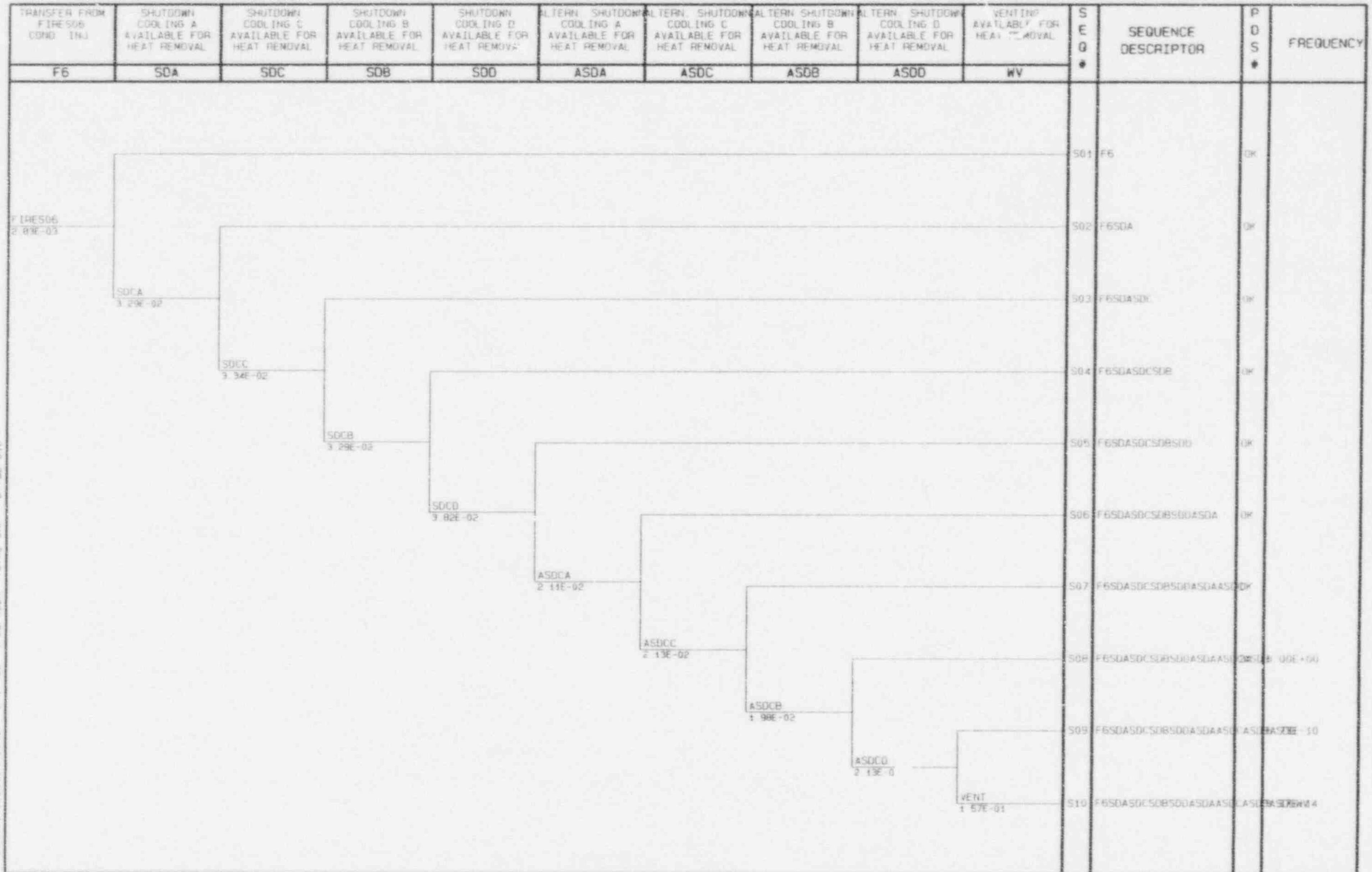
Quantification Date: 6-07-96 4:48:47pm TOTAL CMF = 0.00E+000

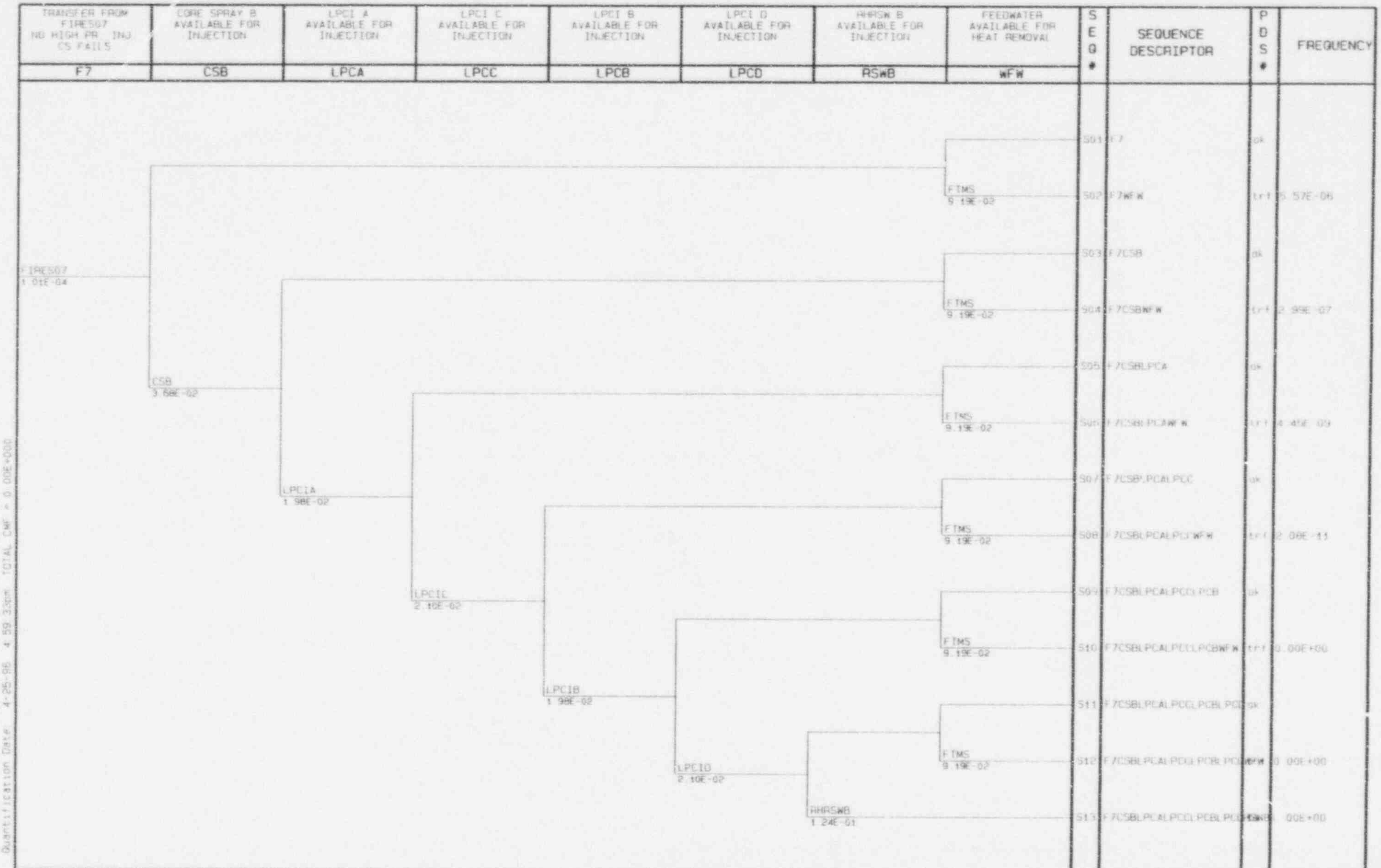
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM FIRES05
FEEDWATER INJECTION SUCESSFUL



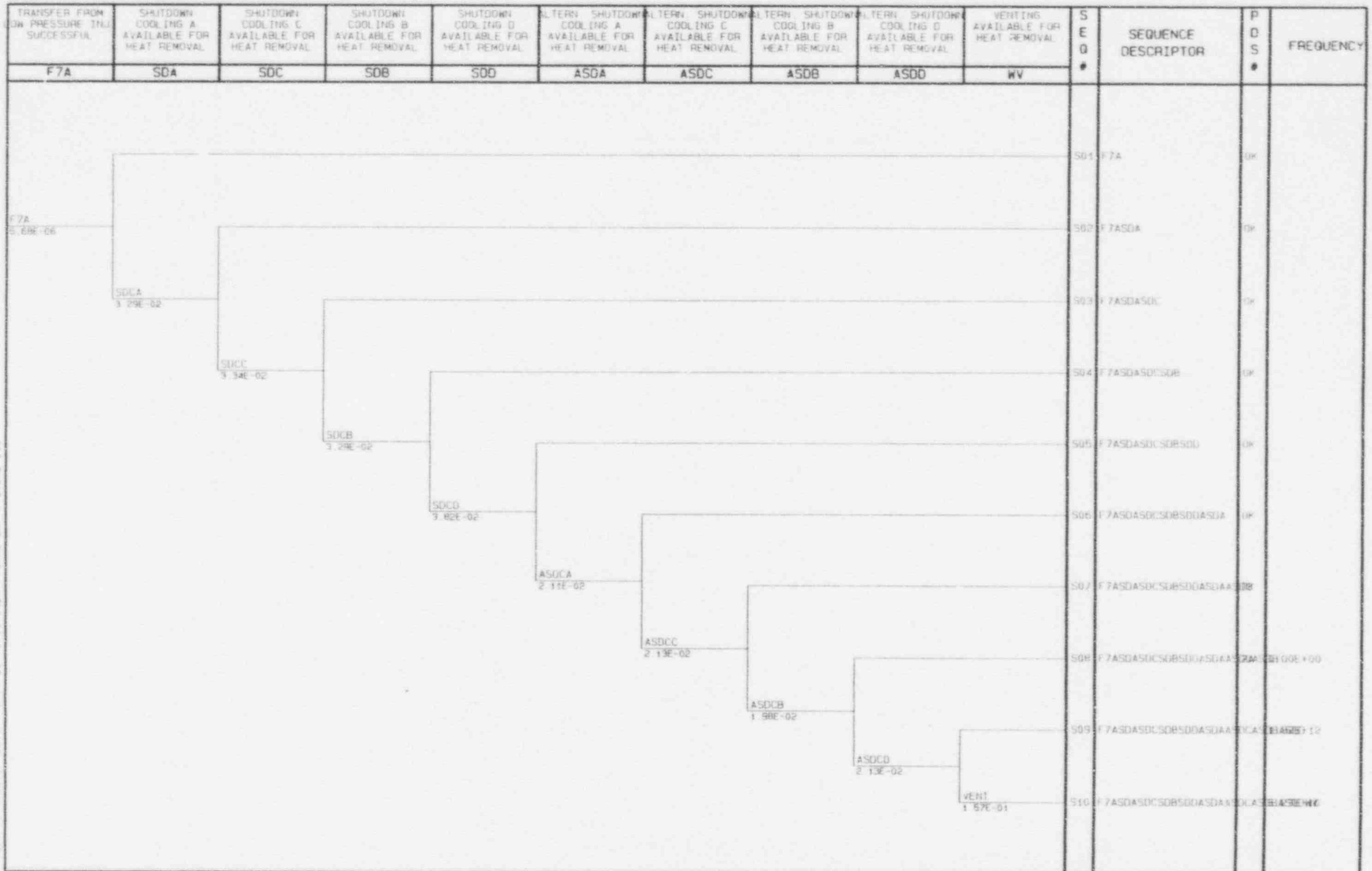
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM FIRES06
CONDENSATE FOR INJECTION



FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM FIRES07
NO HIGH PR. INJ. CSA FAILS



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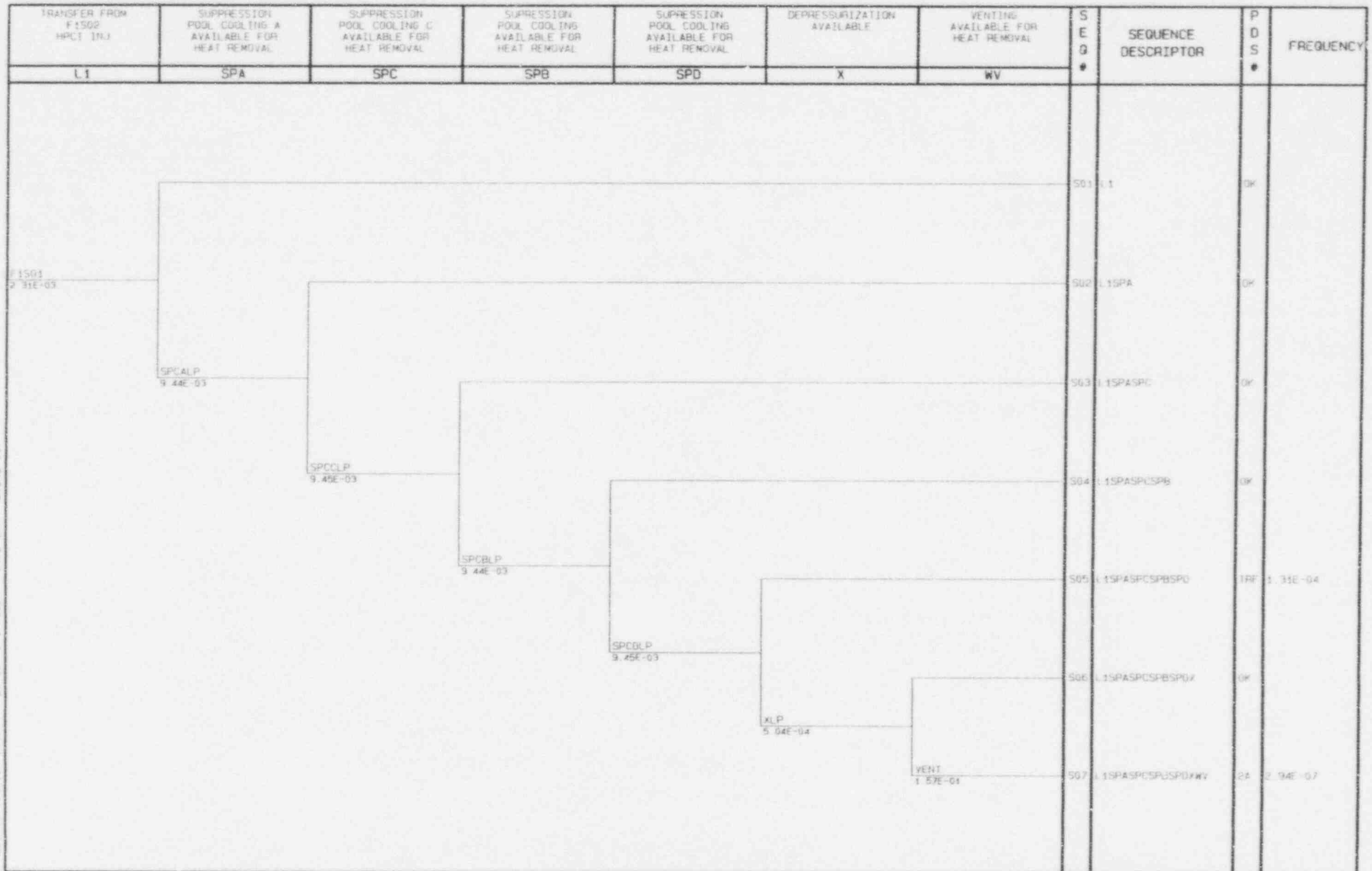
Quantification Date 8-07-96 5:03:01pm TOTAL CMF = 8.24E-010

FIRE AREA EVENT TREE
LIMERICK GENERATING STATION

LOW PRESSURE INJECTION SUCCESSFUL



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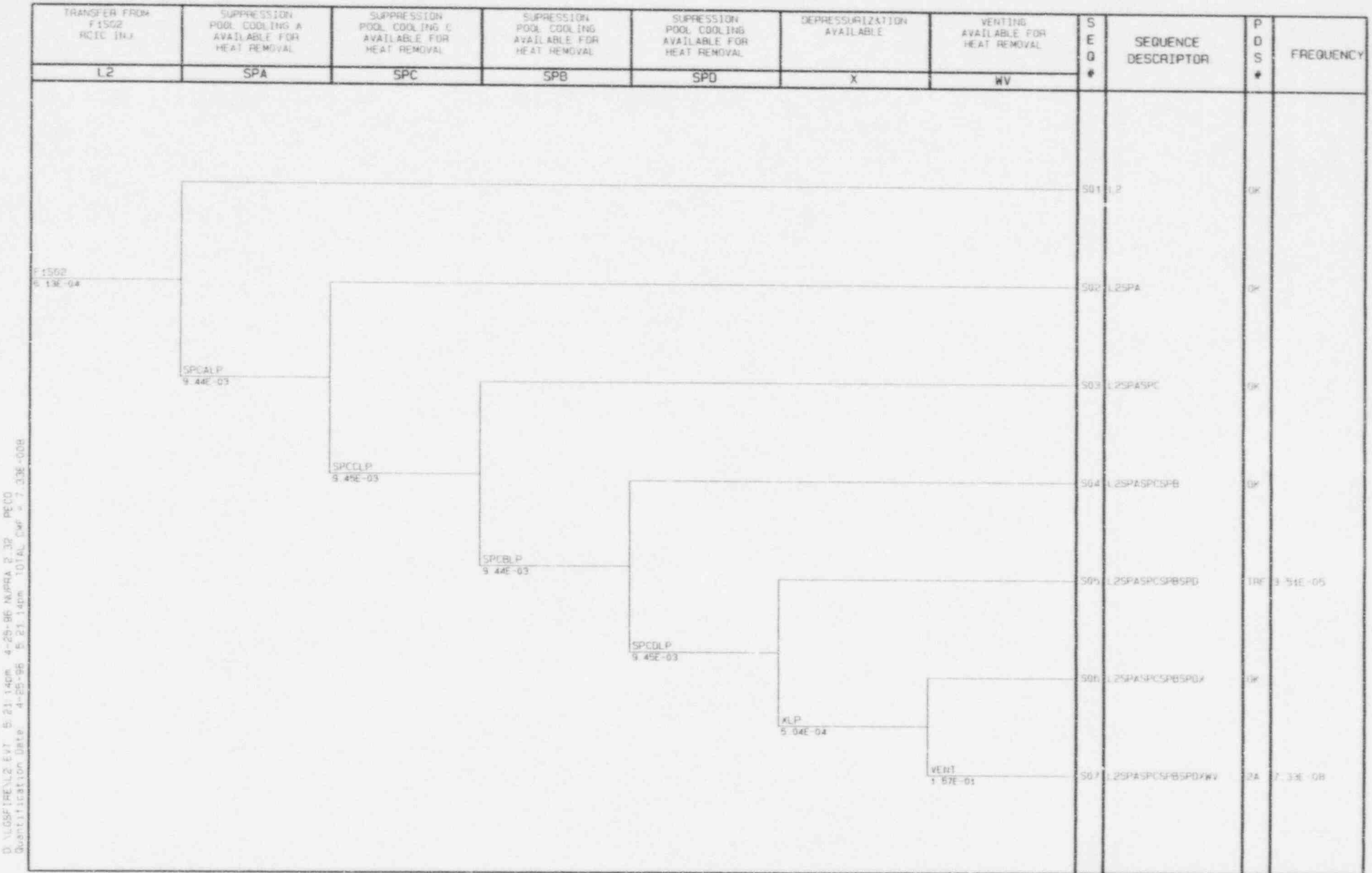
Quantification Date 4-25-96 5:17:35pm TOTAL CMF = 2.94E-007

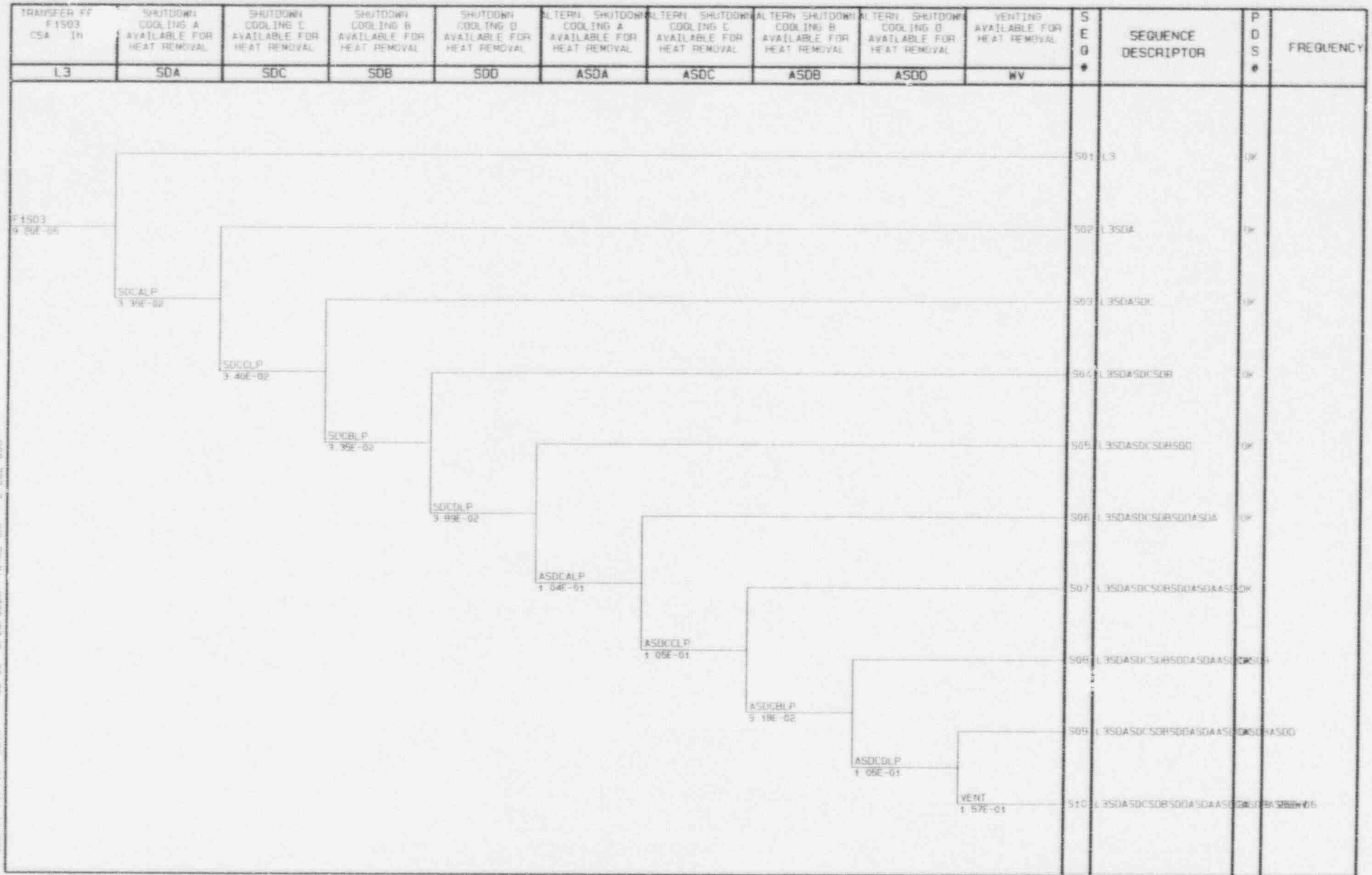
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM F1S01
HPCI INJECTION SUCCESSFUL



FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM F1S02
RCIC INJECTION SUCCESSFUL



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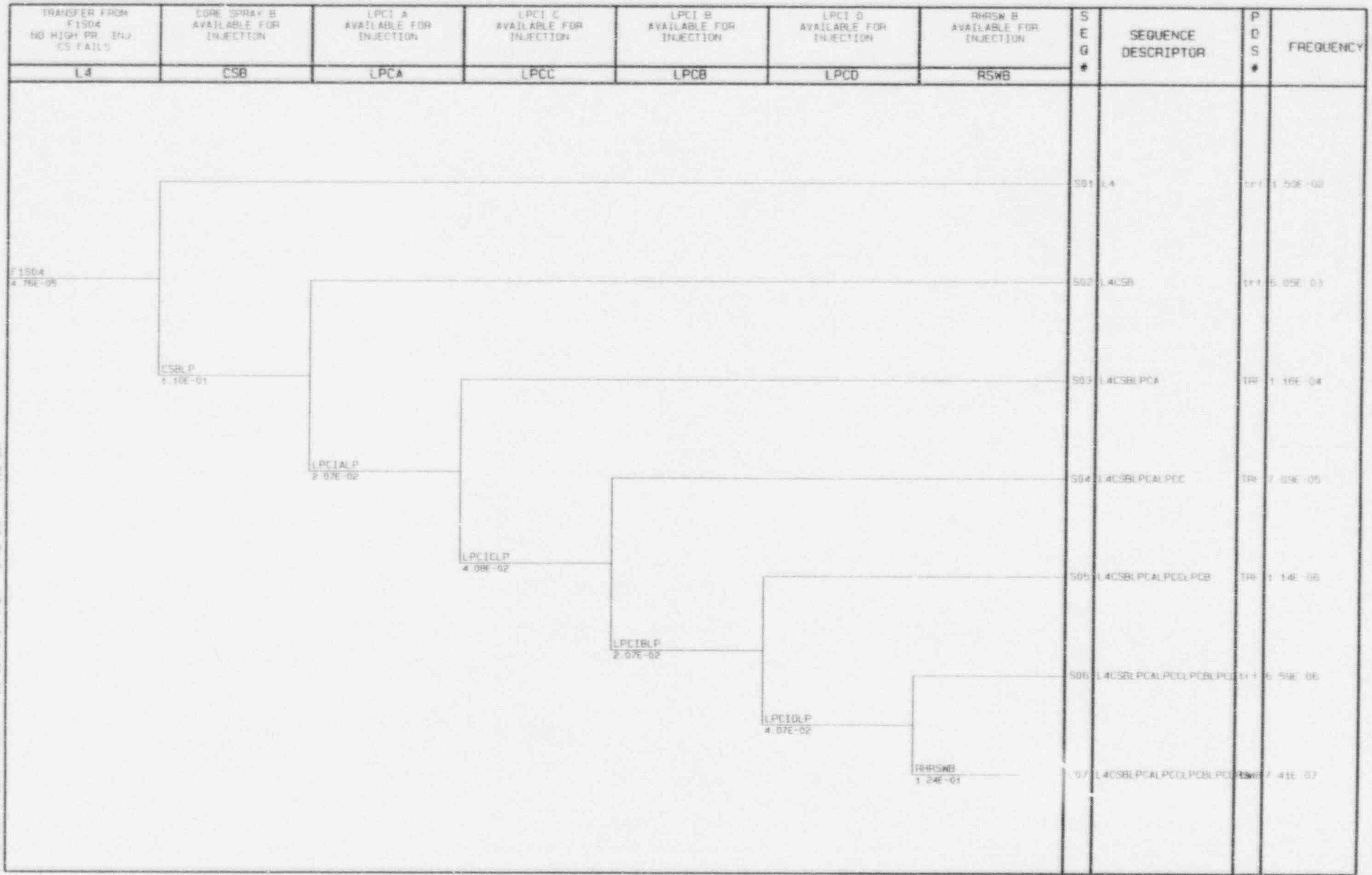
Quantification Date: 4-25-96 9:23:30pm TOTAL CMF = 2.26E-006

FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM F1S03
CORES SPRAY A FOR INJECTION



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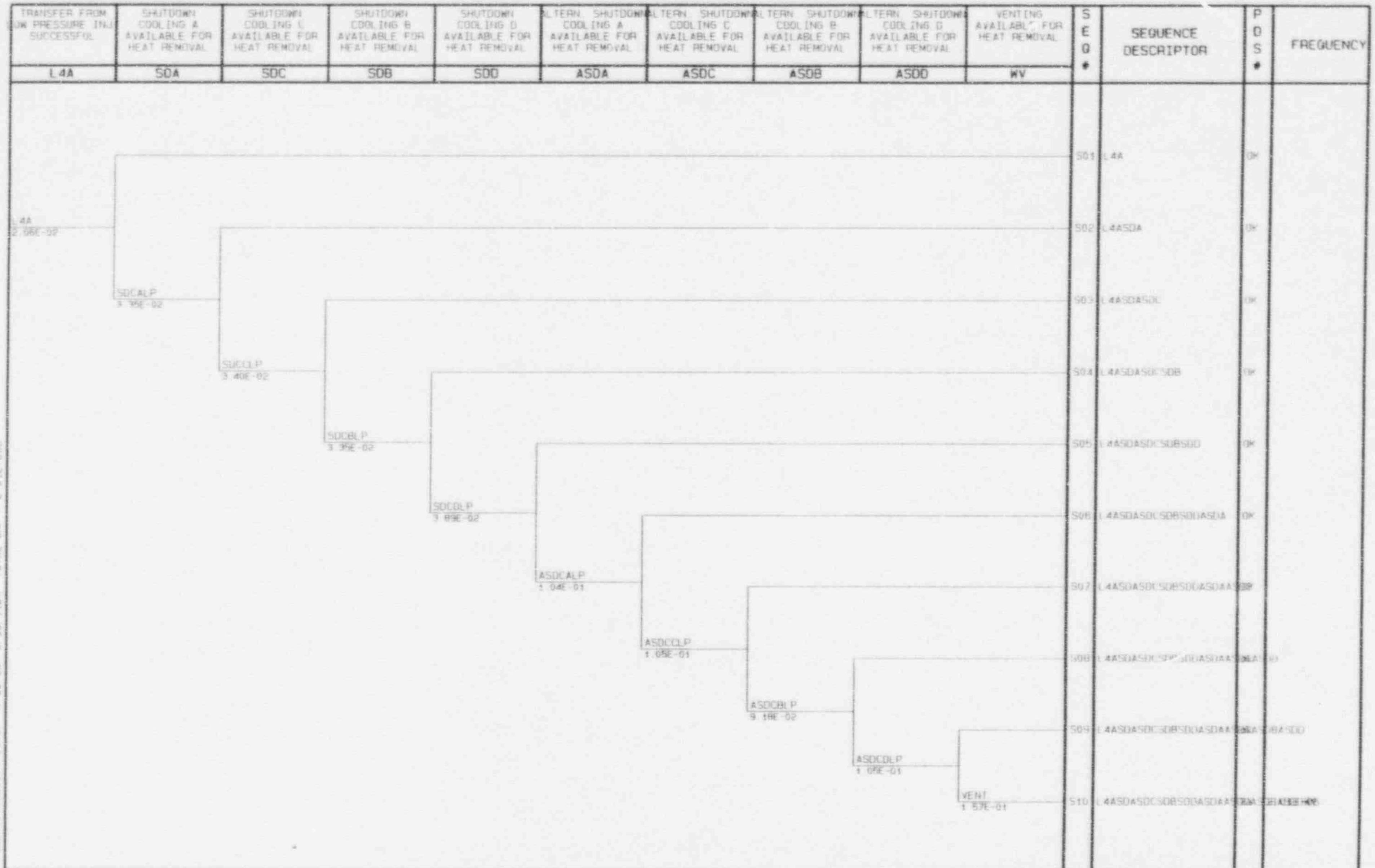
FIRE AREA EVENT TREE FOR
LIMERICK GENERATING STATION

TRANSFER FROM F1S04
NO HIGH PR. INJ. CSA FAILS



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Quantification Date: 4-25-96 5:29:11pm TOTAL CMF = 2.01E-005

FIRE AREA EVENT TREE
LIMERICK GENERATING STATION

LOW PRESSURE INJECTION SUCCESSFUL
LOSS OF OFFSITE POWER