4.0 CONTAINMENT RESPONSE (LEVEL 2) ANALYSIS

This chapter presents the analysis of the Containment Building (Containment) response to severe accidents. The accident data are treated on the bases of internal event initiators. The first seven sections are generally organized as requested in NUREG-1335. Further sections are added here to present sensitivity analyses results.

Section 4.1 identifies and highlights the component, system, and structure data that is of significance in assessing severe accident progressions.

Section 4.2 discusses the MAAP plant analytical model, the modifications of the code specific to this plant, and the selection of empirical factor and data inputs.

Section 4.3 covers the coupling of the front-end (Level 1) analysis to the back-end (Level 2) analysis, through the binning of the Level 1 accident sequences, extended to include Containment systems, into plant damage states with similar back-end characteristics.

Section 4.4 characterizes the Containment strength assessment and the magnitude of various loads necessary to fail Containment.

Section 4.5 delineates the Containment event trees which characterize the possible paths that an accident sequence may progress along, given the sets of initial conditions defined by the various plant damage states. The decomposition event trees that expand on each of the Containment event tree headings are presented. These ancillary trees include the numerical expression of the judgment of the phenomenological uncertainties ("split fractions") and it is convenient to present these split fraction assessments in this section. The methods and the results of the Containment event tree probabilistic analyses (quantification) are also in this section.

Section 4.6 describes the deterministic Containment accident progression analyses performed to support the Containment event tree development and quantification and to provide insights and information on the plant response.

Section 4.7 presents the radionuclide release source term development, analyses, and numerical results.

Section 4.8 discusses the results of a structured sensitivity analysis performed to identify the significant sensitivities to phenomenological probability variations and to assess the impact of variation in Level 1 results on the overall Level 2 results.

4.1 PLANT DATA AND PLANT DESCRIPTION

4.1.1 General Containment Building Structure

The Containment Building structure (see Figure 4.1.1-1) houses the reactor vessel, the Reactor Coolant System (RCS), and supporting primary systems. The structure consists of a 126-foot inside diameter (ID) reinforced concrete right circular cylinder with a flat base and hemispherical head.

General Description

The Containment Building structure is constructed of reinforced concrete and rests on a rock surface. There is a waterproof membrane beneath the bottom of, and outside of, the Containment walls below grade (grade is the finished ground level at the North Anna site). The inside of the structure is lined with steel liner plate to form a gas-tight barrier. The Containment structure is designed for a leakage rate not to exceed one-tenth weight percent per day at the design pressure of 44.1 psig.

The structure is designed to minimize leakage of radioactive material in the event of the most severe postulated Loss of Coolant Accident (LOCA), a double-ended rupture of a reactor coolant pump suction pipe. However, a design compromise must be made between the need for an absolute leaktight structure and the need for access to the structure's interior by people and material.

The number of points at which the gas-tight liner is penetrated is limited in order to minimize the potential for leakage. The penetrations include:

1. a personnel air lock,
2. an equipment hatch,
3. piping penetrations,
4. electrical penetrations, and
5. the fuel transfer tube.

All penetrations are leak-tight assemblies welded to the steel liner.

The personnel air lock is the normal entry and exit point for people and tools. It consists of a horizontal cylinder with gasketed doors at each end. The doors are interlocked so that only one door may be open at a time. There is a smaller air lock of similar design for emergency use, located in the equipment hatch. The equipment hatch is the largest penetration in the Containment liner. It is opened only during outages when it is necessary to move very large components into or out of Containment and when Containment integrity is not required.

There are two types of piping penetrations. For individual pipes carrying cold fluids, the penetration is formed by welding the pipe to the Containment liner. For pipes carrying hot fluids or for penetrations which provide access for multiple small pipes, the penetration is sleeved. The pipes are welded to the sleeve, and the sleeve is welded to the liner. Insulation and a liquid cooling system are provided between the pipe and the sleeve on penetrations for pipes carrying hot fluids. On each side of most piping penetrations there are isolation valves which shut automatically on rising Containment pressure.

Electrical wires penetrate Containment via several different penetration designs, the differences based primarily upon wire size. Wire sizes vary from 16 AWG thermocouple wire to 1000 MCM power cable for 4160 V motors. All electrical penetrations are specially designed to be leaktight and have provisions for testing for leakage. Separation of safety signals and fire prevention were important considerations in the location of electrical penetrations.

The fuel transfer tube is used to move new and spent fuel between the refueling cavity in Containment and the spent fuel pool in the Fuel Building. This penetration is open only during refueling outages.

A 250-ton polar crane is provided inside the Containment structure for maintenance and refueling activities. The crane is supported by a cylindrical wall, which runs 10 feet inside the circumference of the Containment wall.

The inside of the Containment structure is maintained at subatmospheric pressure during power operation. The pressure is reduced to, and maintained at, subatmospheric conditions by the Containment Vacuum and Leakage Monitoring System.

Concrete Structure

The concrete structure was constructed from the ground up, and this description follows a similar order of presentation. Refer to Figure 4.1.1-1 during the discussion. A circular excavation was made to elevation 203'-7" in order to found the structure on crystalline, metamorphic rock. The excavation methods minimized damage to the sidewalls, which were then reinforced with rock bolts, gunite, and covered with a welded wire fabric. The bottom of the excavation was covered with a 6-inch layer of porous concrete. Porous concrete is permeable to water and performs a drainage function as one part of the scheme to protect the Containment from ground-water corrosion. Porous concrete is formed by omission of the fine aggregate from standard structural concrete mix.

A flexible, polyvinyl chloride sheet with a minimum thickness of 40 mils surrounds the structure below grade. This waterproof membrane is another part of ground-water corrosion protection. The waterproof membrane was laid on top of the 6-inch layer of porous concrete; then another 4-inch layer of porous concrete was poured on top of the membrane. The sidewall of the excavation was covered with fill concrete from the porous concrete level up 10 feet to form a smooth surface. The waterproof membrane was brought up against this and held in place by 4-inch concrete blocks set dry. The 4-inch concrete blocks are also porous.

The containment foundation mat was formed in the 10-foot deep cylinder now existing in the bottom of the excavation. The containment foundation mat, walls, and dome are heavily reinforced with steel reinforcing bars (rebar) and other steel inserts. The largest and most frequently used rebar size is number 18, manufactured with controlled chemical composition and a minimum yield strength of 50,000 psi. The rebar in the bottom part of the mat is placed in a grid pattern. The top part of the mat contains rebar laid in concentric circles with radial spokes. The top pattern is arranged to permit the vertical wall rebar to extend into the mat. Structural concrete was poured and rodded over the mat rebar. During the pours, the temperature of the concrete was carefully controlled and the concrete was sampled frequently. Water was kept puddled on the surface while the concrete cured.

The containment cylinder wall was formed as rebar was placed in identical patterns near the inside and outside wall faces. Each pattern consisted of vertical and horizontal members. The two patterns were connected by rebar inclined at 45-degree diagonal angles in order to resist seismic stresses. Set at 45-degree inclines near the base mat are shear assemblies to resist the loads associated with the containment pressurization resulting from the DBA. The shear assemblies were constructed of 4-inch by 0.75-inch steel plate welded to vertical rebar. The cylinder wall concrete pours were made in 6-foot lifts, in approximately 18-inch layers so that one layer did not set before the following layer was poured. The mold for the inside of the cylinder wall was the containment steel liner (fully described in following paragraphs). After the cylinder wall pours were completed above grade and their outside molds removed, the work between the outside of the cylinder wall and the excavation could be finished. The waterproof membrane was brought around the exposed lip of the containment foundation mat and covered with a 4-inch layer of porous concrete. The waterproof membrane was then brought up around the outside of the cylinder wall to a level approximately 6 inches below finished grade level (270'-6") and fixed to a continuous Nob-lock termination strip embedded in the concrete. The outside of the waterproof membrane was covered with a 2-inch layer of Rodofoam soft grade 300, a compressible material, to provide a "rattle-space" between the Containment Structure and the rock excavation. The space remaining between the Rodofoam and the rock was filled with concrete

backfill. The finished cylinder wall is 4.5 feet thick and rises 127 feet above the top of the containment foundation mat.

The containment dome is a hemisphere with an inside radius of 63 feet. Rebar was placed in a pattern of arced spokes extending in two layers from the center of the dome outward to connect with the rebar in the cylinder wall. Structural concrete was poured for the dome in the same manner as for the mat and cylinder wall. The inside form for concrete placement was the steel liner. The concrete dome is only 2.5 feet thick rather than 4.5 feet thick as in the cylinder wall.

Steel Liner

The liner completely envelops the interior of the concrete structure to form a gas-tight membrane. It is constructed of two types of steel: ASTM SA-573, Grade B, quenched and tempered, is used for the first 28'-5" above the mat; and ASTM SA-516, Grade 60, fine-grained and normalized is used for the rest of the cylinder, mat and dome lining. The liner is anchored at close intervals to the inside of the concrete structure for support and for transfer of loads. The anchors are designed so that failure would occur at the anchor so that liner integrity would not be affected. It is important to note that the liner does not perform any direct structural function; all loads are ultimately borne by the concrete structure. The dimensions of liner thickness are as follows:

1. 0.25-inch thick on the Containment foundation mat,

2. 0.375-inch thick on the cylinder wall,

3. 0.5-inch thick on the dome, and

4. 0.75-inch thick under the in-core instrumentation area and sump area.

There are steel insert plates on the liner with additional concrete anchors which have had brackets welded to them, primarily for the support of Quench Spray System piping and spray headers/rings.

The liner was designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, with Code Addenda through summer 1969. Consequently, no welding directly to the liner is allowed without special written procedures. There are steel insert plates on the liner with additional concrete anchors which have had brackets welded to them, primarily for the support of Quench Spray System piping and spray headers/rings.

Interior Structures

The steel liner on the foundation mat is covered with a concrete surface that is generally 2 feet thick, sloped for drainage toward the sump. The purpose of the concrete mat covering is to protect the mat liner from damage which might be caused by internally generated missiles or dropped equipment. This concrete protection does not exist in two areas, beneath the in-core instrumentation and the sump, which is why the steel liner is thicker in these areas.

Rising above the concrete mat covering are a number of reinforced concrete walls, all of which perform three major functions:

1. equipment structural support;
2. biological shielding, to permit personnel entry during reactor operation; and
3. missile shielding, to protect the steel liner from puncture by internally generated missiles.

The first such wall inside the Containment Building is the crane support wall. This wall is 2 feet thick and provides biological and missile shielding from the concrete mat covering at elevation 216 feet to the bend line at elevation 342 feet. The polar crane rests on top of this wall. The annulus space between the crane support wall and the steel liner houses cable trays, piping, and ducting. The annulus includes several levels of grating that are accessible by ladders from the operating floor. Walls interior to the crane support wall are arranged to form equipment support cubicles for major components: three Steam Generator/Reactor Coolant Pump cubicles and the pressurizer cubicle. Common walls of these cubicles enclose the reactor vessel and form part of the refueling cavity. To shield the Containment dome from missiles generated by the failure of a control rod drive mechanism a 2 foot thick concrete shield plug with a steel plate on the lower side covers the reactor vessel when the plant is in operation.

The primary shield wall surrounds the neutron shield tank and the reactor vessel. The wall is 4.5 feet thick with a large opening in the bottom portion to allow personnel access to the underside of the neutron shield tank and the reactor vessel. The opening also allows passage of the thimble conduits of the In-Core Instrumentation System to the underside of the reactor vessel. The top of the primary shield wall has six penetrations for Reactor Coolant piping to pass through to the Steam Generator/Reactor Coolant Pump cubicles. The area in and around the penetrations can become very hot, so the penetrations are lined with primary shield wall sleeve coolers.

All interior concrete surfaces and the steel liner are covered with special epoxy coatings to prevent corrosion and to facilitate decontamination in the event of a radioactive spill. The refueling cavity and the fuel transfer canal are lined with 0.25-inch stainless steel plate which is not coated.

Access Penetrations

Access penetrations include the personnel air lock, the equipment hatch, and the emergency personnel escape lock.

The personnel air lock is the primary point of entry and exit for the Containment Building. Additional rebar and anchors were placed in the concrete surrounding the lock for support. The lock is welded to the steel liner.

The largest penetration to the Containment Building is the equipment hatch. This opening in the Containment wall is 14 feet, 6 inches. It is covered on the inside by a single steel closure head with a removable emergency personnel escape lock and protected on the outside by a concrete missile/radiation shield. The equipment hatch is sized to permit removal of any component from the Containment Building except the reactor vessel, reactor vessel head, or upper Steam Generator shells. The steel closure head is sealed to the Containment steel liner and to the emergency personnel escape lock by sets of double O-rings.

The emergency personnel escape lock is a 2-foot, 6-inch interior diameter air lock mounted in the equipment hatch. It is intended for emergency egress from Containment when the personnel air lock is inoperable or inaccessible. The design of the emergency personnel escape lock is similar to the personnel air lock.

Piping Penetrations

The unsleeved piping penetrations are designed for single pipes which carry cold (less than 150°F) fluids. The pipe is welded to the Containment steel liner to continue the gas-tight membrane, and it is welded to reinforcement plates embedded in the Containment concrete for structural support.

The sleeved piping penetrations are designed for multiple pipes (typically instrumentation tubing) or single pipes which carry hot (greater than 150°F) fluids. The sleeve is welded to the Containment steel liner to continue the gas-tight membrane and is welded to reinforcement plates embedded in the Containment concrete for structural support. The pipe(s) is (are) welded to the sleeve. The sleeved penetrations for pipes carrying hot fluids are insulated and cooled to reduce the damaging structural effects of elevated concrete temperature.

The fuel transfer tube penetration design is similar to a sleeved piping penetration. The purpose of the fuel transfer tube is to facilitate the movement of reactor fuel assemblies between the refueling cavity in the Containment Building and the spent fuel pit in the Fuel Building. Since each end of the sleeve is anchored to a different building, several bellows assemblies are used to allow for movement between the buildings and for thermal movement.

Important North Anna Containment and other design features are presented in Tables 4.1.1-1 and 4.1.1-2.

Electrical Penetrations

Electrical penetrations employ two basic design concepts for leaktightness: canister and compression fitting. With both designs, a sleeve is welded to the containment liner for continuation of the gastight membrane. A leak test channel is welded to the liner and sleeve. The sleeve is flanged at both ends.

Type V medium voltage penetrations for 4160 volt cables employ the canister design mounted in a 12-inch diameter sleeve. A leaktight cylinder with internal stiffeners for supporting the conductors is welded to a 12-inch flat-faced flange. Three conductors penetrate the canister flange and canister. The canister flange is bolted to the sleeve flange on the Cable Vault side and is sealed to the sleeve flange with a set of double O-rings. The canister is normally pressurized with dry nitrogen to 15 psig. A pressure gage is provided to detect leakage from the canister. A tap is provided to periodically test for leakage from the space between the O-rings. The canister is bolted to the containment side flange for structural support (not as a barrier to leakage).

All other penetrations types (type 1 instrumentation and control, type II low voltage power, type III nuclear instrumentation, and type IV thermocouples) use the compression fitting design. On these types of electrical penetrations, a bolted flat-faced flange is sealed to the 6-inch sleeve flange on the Cable Vault side with a double set of O-rings. A number of holes are drilled in the flange face and tapped with NPT threads. A compression fitting seals between the flange face and a feedthrough assembly. The feedthrough assembly is fabricated by drilling holes slightly larger than the conductor diameter in cylinders of polysulfone (a radiation-resistant plastic with a high dielectric constant) of approximately 1-inch lengths. Solid copper conductors are then inserted in the drilled polysulfone cylinders, and the entire assembly is sheathed by a stainless steel tube. The completed assembly is turned in a special lathe which seals the polysulfone cylinders to each other and to the conductors. A small hole is drilled in the stainless steel tube near one end of the feedthrough assembly. That hole lines up with a similar hole in the

compression fitting a machined canal in the flange. The machined canal from each feedthrough assembly converges on one common tapped hole containing a pressure gage and fill assembly. This space is normally filled with dry nitrogen pressurized to 15 psig, and the pressure gage is periodically monitored to detect leakage from the feedthrough assemblies.

4.1.2 Reactor Cavity Design

The reactor cavity at North Anna is isolated from the remainder of the lower Containment compartment. The North Anna cavity design is shown in Figure 4.1.2-1. It is composed of a cylindrical region directly under the reactor vessel and a connected rectangular tunnel for passage of the incore instrument tubes. The base of the North Anna cavity is at an elevation two feet below that of the Containment floor external to the cavity. The minimum depth of water in the cavity for overflow from the cavity into the lower Containment floor and sump is approximately 16.5 ft. At this depth the total water volume contained in the cavity would be approximately 8700 ft^3.

Water can enter the cavity in one of two ways. Following vessel failure any water being injected into the vessel will enter the cavity. If Quench or Recirculation Sprays are operating then that portion of the spray flow which falls into the refueling pool (estimated to be 16% of total spray flow) will flow into the cavity (after the refueling transfer canal is filled). All other spray water will flow to the Containment sump. With full operation of all spray systems the cavity will fill with water in approximately 20 minutes.

The maximum water elevation on the Containment floor (outside the cavity) considering the entire contents of the Refueling Water Storage Tank and the contents of the Primary System is approximately 6.0 ft. This is far short of the required depth of 16.5 ft. for overflow from the lower Containment/sump into the reactor cavity.

The outer end of the instrument tunnel away from the reactor vessel is not sloped as in some designs (e.g., Zion). This should result in a geometry somewhat less favorable for debris dispersion out of the cavity following vessel failure. At the outer end of the instrument tunnel the instrument tubes are directed vertically upward through a "cofferdam" raised above the adjacent tunnel ceiling penetrating the seal table into the RH pump and heat exchanger region (floor elevation = 234 feet). This is a likely pathway for debris (and water and gases) to be expelled from the cavity under high reactor vessel pressure failure conditions. Other pathways for debris to be expelled from the cavity include an (approximate) 9 ft^2 vertical ventilation duct on the roof of the instrument tunnel and an (approximate 2 ft^2) horizontal ventilation

duct at the 231 ft elevation through the cylindrical portion of the reactor cavity wall. Two other possible pathways for debris to be dispersed from the cavity involve transport through the annulus between the reactor vessel and biological shield to either the refueling pool or to the main coolant pipe penetrations through the biological shield wall.

Transport of material through the reactor vessel annulus is hindered by the presence of the massive neutron shield tanks which occupy most of the volume within the region between the reactor vessel and shield wall. In the remaining 6.5 in. gap between the reactor vessel and neutron shield tank is 3.0 in. of insulation. There is approximately 20 ft^2 of flow area around the Reactor Coolant piping penetrations in the shield wall (Hutcherson, 1989). Stone and Webster and Argonne National Laboratory have reviewed the Surry cavity design and judge that the Benelux shield cans above the vessel nozzles would very likely be crushed and impacted against the water seal ring at the top of the vessel-shield wall annulus (by the high pressure expulsion of gases, from the reactor vessel accelerating debris/water up the annulus) closing off the flow pathway for debris to the refueling pool region in the upper Containment (Hutcherson, 1989). Similar phenomena would occur at North Anna since the configuration is similar at both plants.

4.1.3 Safeguards Building

The North Anna Safeguards Building is a relatively small structure adjacent to the Containment which houses the Low Head Safety Injection pumps and the Outside Recirculation Spray pumps. It is important in that an interfacing system LOCA would occur within this building. The Safeguards Building covers an arc of the Containment wall of approximately 66 feet and is approximately 15.5 feet wide. An access door is located at elevation 273'. The LHSI and Outside Recirculation Spray pumps are located in cubicles with a floor elevation of 256.5'. Adjacent to the LHSI and spray pump cubicles is a piping and valve area. Adjacent to the Containment wall is a shaft which allows access to the safeguards valve pit (ceiling elevation 221' 4"). The volume of rooms below elevation 256.5' (the access shaft and the safeguards valve pit) is approximately 1000 ft^3. The volume of the Safeguards Building between the floor at elevation 256.5' and grade elevation at 271' is about 10,000 ft^3. Hence, the volume of water needed to fill the Safeguards Building to grade level is about a factor of 6 less than the volume of the RWST which contains about 65,000 ft^3 and is on the order of the volume of water in the RCS. Much of the Low Pressure Systems piping in the Safeguards Building is located within five feet of the floor.

Figures 4.1.3-1 and 4.1.3-2 show drawings of the North Anna Safeguards Building in relation to the Containment.

Flooding of the Safeguards Building would reach the elevation of piping penetrations to the Quench Spray Pump House (QSPH) where the penetration fire barrier would allow water into the QSPH basement. A modification was made to the piping penetration from the QSPH basement and the Auxiliary Building to prevent a flood from Safeguards or the QSPH from propagating to the Auxiliary Building.

4.2 PLANT MODELS AND METHODS FOR PHYSICAL PROCESSES

The Modular Accident Analysis Program (MAAP) was used in the North Anna IPE Containment evaluation for the accident progression analysis, to assist in quantifying the CET, and for estimating source terms. Information from prior analyses (principally NUREG-1150) was also utilized.

Some of the material presented in Sections 4.6 and 4.7 were developed as part of the Surry IPE using MAAP 3B Revision 15. This version of MAAP is described in Section 4.2 of the Surry IPE report (VEPCO, 1991) and the ancillary Surry MAAP parameter file is given in Appendix F of that report.

MAAP 3B Revisions 17.02 and 18 were used for the North Anna Level I and Level II analyses. Generic modifications were made to Revision 17 of the MAAP code to allow the correct modeling of North Anna specific ESF lineups and operational characteristics. These modifications are described in subsection 4.2.2 below. Section 4.2.1 briefly summarizes several important MAAP input modeling assumptions which were utilized in the North Anna study. The entire North Anna MAAP model (parameter file) is contained in Appendix F (Section F.5).

4.2.1 MAAP Analysis Assumption (Model Parameters)

The MAAP model parameters generally represent inputs to phenomenological models in which significant uncertainties exist. Variation in the values of these parameters can be used to assess the impact of uncertainties in important physical models. The nominal, or default, values of these parameters are generally the code developers best estimate for the value of the model parameter. These values are shown in the MAAP User's Manual (FAI, 1990a) and also contained in the MAAP User's Guide (FAI, 1990b).

For most model parameters the default values have been used for base case analysis in the North Anna IPE. The following MAAP model parameter inputs were varied from the default values.

TTRX - Time to fail reactor vessel penetration welds after contact with corium

This parameter represents the time for vessel failure to occur via failure of the lower penetration welds after these welds are contacted by corium.

The nominal default value is 60 seconds with a typical range of from 30 to 1000 seconds.

The relatively rapid default failure time appears to stem from analyses performed for the IDCOR program which indicated high heat transfer rates to the welds during the time the debris was flowing into the lower head from the (failed) support plate. Based on observations from TMI-2 it appears that substantial quantities of debris can migrate into the lower plenum without resulting in rapid vessel failure. Consequently, we have selected a time to vessel failure in the approximate middle of the indicated typical range (500 seconds). This value appears to be realistic for high pressure sequences. For low pressure sequences significantly longer times may be appropriate.

FENTR - Debris entrainment factor

This parameter is used to multiply the critical superficial gas velocity (for molten core debris or water) for entrainment out of the reactor cavity (calculated with a Kutateladze number of 3.0). This gas velocity threshold is then compared with the gas velocity in the reactor cavity/instrument tunnel following vessel failure.

Large values for this parameter make it more difficult for debris entrainment. The nominal value for this parameter is 0.33 which is based on the Zion plant cavity/tunnel configuration with the inclined slope at the exit from the tunnel. This inclined slope would facilitate dispersion of the debris from the cavity. Hence, a value less than unity (indicating a favorable design for dispersion) was selected for the representative value for a Zion like cavity. The North Anna cavity is somewhat similar in design to the Zion cavity with the important exception that there is no incline at the end of the North Anna instrument tunnel (See Figure 4.1.2-1). Consequently, entrainment of debris out of the North Anna cavity should be somewhat more difficult since the debris must be entrained directly upward. Therefore, a value for this parameter of 1.0 has been selected to represent entrainment out of the North Anna cavity.

FPRAT - In-vessel fission product release model selection parameter

Two models are available for calculating in-vessel fission product release. These are the NUREG-0772 (ORNL) model and the IDCOR/EPRI steam oxidation model. Both models have been shown to yield

release predictions which are essentially equivalent. We have chosen to use the NUREG-0772 model because of its extensive experimental basis.

FCRBUC - MAAP in-core blockage model parameter

This parameter activates/deactivates the IDCOR blockage model in the core. Selection of this model stops oxidation and gas flow through a node at the onset of melting in that node. Use of this model tends to greatly reduce hydrogen production. For the base case calculations for North Anna this model has been deactivated. Hence, predicted hydrogen production would be expected to be on the same level as predicted by the Source Term Code Package.

4.2.2 Generic MAAP Modifications

As discussed in Section 4.2 several generic modifications have been made to MAAP Revision 17. Table 4.2.1-1 contains a summary of models and features added for PWR MAAP 3.0B Revision 17.

Generalized Engineered Safety Feature (ESF) changes are the most extensive to the code. The user may select the old ESF features in the parameter file and bypass the new required inputs if you have Zion like ESF features. The North Anna plant requires the new ESF features since the old ESF features could not model these systems accurately. The new ESF features allow the user to specify pump suction and discharge locations, and different pump flow tables may be selected. Turbine driven auxiliary feedwater may be used in addition to the motor driven auxiliary pumps.

Half loop operation may be simulated. You may specify the initial conditions with much freedom: primary and secondary side pressure, temperature, and water level, whether or not the primary or secondary sides are open to containment, decay power as a function of time, etc. MAAP PWR now models separate steam, hydrogen, and air compositions in all primary system subvolumes, so the model is valid beyond core uncovery and fuel damage.

The spray model was revised to account for changes in the steam partial pressure during the fall of a spray droplet. This should greatly improve the model performance and code running time for sequences with core or containment spray.

Model changes were made for the Design Review Group (DRG) in a few instances. These include DCH, EXVIN, PLH2, and numerics. The DCH and EXVIN changes may lead to greater heat transfer and pressurization immediately following vessel failure. The DCH model now considers chemical reactions of iron, chromium, and zirconium with water, steam, and oxygen. Debris may be directed to both the

upper and lower compartments. EXVIN now allows a user-specified delay time for the interaction and allows the steaming rate to be varied.

It is not necessary to implement all the indicated modifications in order to run PWR Revision 17. For example, new ESF features may be bypassed as mentioned above, some modifications are for half-loop operation only, some are for ice condensers, etc.

These code changes have been made by Fauske and Associates, Inc. (FAI under their internal Quality Assurance procedure as required by EPRI).

4.3 BINS AND PLANT DAMAGE STATES

4.3.1 Level 1/2 Interface

This task involved reviews and analysis to assure that the Level 1 and Level 2 analyses were properly interfaced. This task included a review of the Level 1 model to assess its adequacy with respect to the Level 2 analysis.

The interface between the Level 1 Systems Analysis and the Level 2 Containment Analysis consists of a set of plant damage states (PDS). The plant damage states are defined by a set of functional characteristics for system operation which are important to accident progression, Containment failure and source term definition. Each PDS contains Level 1 sequences with sufficient similarity in system functional characteristics that the Containment accident progression for all sequences in the group can be considered to be essentially the same. Each PDS defines a unique set of conditions regarding the state of the plant and Containment Building systems and the physical state of the core, Primary Coolant System and the Containment boundary at (approximately) the time of core damage/vessel failure. The important functional characteristics for each PDS were determined by defining the critical parameters (system functions) which impact the key results. The sequence characteristics which are important were defined by the requirements of the Containment accident progression analysis. They include the type of accident initiator, the operability/non-operability of important systems, the value of important state variables (e.g., Primary System pressure) which are defined by system operation, and timing of key events.

A review of the North Anna ESF/Mitigation Systems and EOPs was conducted in order to assure that the Level 1 analyses included considerations of those systems and operator actions which are important to the Containment analyses and are required for plant damage state definition.

The important sequence characteristics from the Level 1 analyses were evaluated to assure that the event trees provided the necessary and sufficient information for the Containment analysis. Recommendations were provided to the Level 1 analysis on modifications and additional headings (events) to add to the event trees in order to assess the availability of all systems important to the Containment accident progression analysis.

The Plant Damage State Event Trees contain all the necessary information (system events) to allow all sequences to be unambiguously assigned to specific plant damage states. The plant damage states contain all the system information necessary to evaluate the Containment accident progression (evaluate the CET) with the possible exceptions of systems failures which result from the occurrence of specific physical phenomena (e.g., hydrogen burn failing Quench or Recirculation sprays) and operator, recovery or mitigation actions which occur subsequent to core damage. Since systems important to the Containment analysis have been considered in conjunction with systems important to core damage in the Level 1 analyses, dependencies between Containment systems and all other systems have been handled in a rigorous and consistent manner.

The Plant Damage State analysis includes the following considerations:

Identification of Important Plant Systems

The accident initiators, plant systems and various possible states of the Primary System and Containment, (at the time of the core damage) and station procedures were reviewed to determine their potential impact on Containment accident progression. The most important features were identified and the PDS defined to assure that these systems were considered.

Identification of Key Event Timing

The timing of key events such as system failure/recovery, and operator actions were assessed. The PDS were structured to assure that the key event timing information was appropriately considered.

Differences and Similarities Between North Anna and Surry

North Anna is generally similar to Surry in respect to severe accident response. There is one significant exception to this regarding the low pressure safety system pumps which is discussed in the next paragraph and some other less important differences that are noted below. The lineup of low pressure, high pressure, RHR, Quench and Recirculation Spray Systems is the same at North Anna as at Surry. A Casing Cooling tank at North Anna provides

extra water volume in the injection mode and provides additional head for some of the pumps in the spray system. The containment is similar. While the North Anna containment is 5.5 ft taller, it is designed by the same A/E and has approximately the same strength. The basemat aggregate is of the same type (low gas production when decomposing due to corium attack.) A comparison between North Anna and Surry for certain factors that affect severe accident progression is given in Table 4.3.1-1. The power level for North Anna is higher than for Surry. Given the same Containment volume, the RCS stored energy, Zircaloy (hydrogen) mass, and fission product mass ratioed to volume will therefore be higher. These pose more of a challenge to Containment and some increase in predicted failure frequency is possible as a result. Another difference is seen to be the Casing Cooling tank water supply to the sprays. This water appears as a larger inventory that can be sprayed into the containment as compared to Surry, but other than timings it should not affect results in a significant way.

The North Anna low pressure injection and spray pumps do not necessarily fail on high temperature. They are environmentally qualified to 300°F. When pumping hot water in the containment, (in recirculation mode without the containment heat removal function working) if the temperature gets to somewhat beyond this (say to 350°F) the containment is likely to fail from overpressure. There is some probability of the pumps continuing to work at this elevated temperature. The Level 1 analysis considers that the pumps will continue to work with a probability of 1.0 and that they will continue to run after containment failure with a probability on the order of 0.98. These sequences are referred as "core vulnerable." If the pumps continue to work then the containment is in effect vented and the heat removal is accomplished by venting steam from containment through the break. This process can go on for some extended time, particularly if the RWST has been injected into containment. In these cases (containment failed but not the pumps) the Level 1 analysis considers the sequence to be non-core melt ("OK" or success). In Surry the pumps always failed early at a much lower temperature and core melt ensued into an intact containment. In this North Anna analysis provision must be made for core melt into a failed containment. Because the frequency of these sequences is low, on the order of 2E-7/year overall, it is considered reasonable to place them in the same class as the loss of isolation sequences, which are on the order of 4E-8/year.

The internal flooding initiator frequency is much smaller for North Anna than for Surry. Hence only one set of frequencies is used for North Anna - that for all the internal initiators. By the same token, however, the station blackout (SBO) type sequences become relatively more important and it was deemed necessary to use a somewhat different structure for the Plant Damage Diagram for North Anna as compared to Surry. In particular, the SBO type sequences are first classified as either of a small/medium LOCA type (for example due to a stuck-open relief valve) or else as a transient

type, and then considered as an SBO type. This results in the classification of North Anna sequences by accident progression type rather than by Level 1 initiator type. For example, therefore, there is not a one-to-one match between SBO frequencies in the North Anna Plant Damage States and the Level 1 SBO initiator frequency and due attention must be paid to the meaning of the Plant Damage Diagram definitions.

4.3.2 Plant Damage State Grouping Criteria

Plant Damage State Definition

The entry points to the Containment Building event trees are plant damage states. The plant damage states are groupings of the (extended) core melt (Level 1 sequences) into similarity bins or states. The goal of the grouping process is to reduce the number of required Containment analyses to a tractable number while continuing to distinguish the more important differences among the sequences which are likely to influence the Containment accident progression. The plant damage state characteristics are defined by selecting a set of key systems operation related parameters which are considered to be important to: accident progression in the Containment; the time, mode and location of Containment failure; and the radionuclide source term. The parameters that are used to define the plant damage states include the functional status of important systems, state variables which are determined by systems operation (e.g., Reactor Coolant System pressure), accident initiator type and timing of key events (e.g., power recovery).

Nine criteria were selected for use in defining the NAPS plant damage states. A description of these criteria and the bases for their selection are discussed in detail below. Generally, these criteria were selected because they have a controlling influence on determining key accident progression characteristics; the time, mode and location of containment failure and the radionuclide source term to the environment.

The plant damage state logic diagram is a tool used to perform the classification by combining the various grouping parameters into unique plant damage states. An initial logic diagram was constructed with the selected nine criteria as decision branches to aid in the assembly of specific plant damage state characteristics from the matrix of all possible combinations allowed by these nine grouping parameters.

Using nine parameters in a grouping logic structure with (only) binary choices at each decision point would result in 2^9 (512) groups which is clearly intractable. However, by arranging the logic diagram in such a way that the most important parameters are considered before parameters of lesser importance, and eliminating

decision points by allowing only one decision branch results in the collapse of the number of plant damage states to a reasonable number while still preserving the most important differences among the various sequences. The reasons for suppressing branching on a decision branch are somewhat judgmental and involve the following considerations:

(1) Is this branch necessary to distinguish an important difference among the sequences?

(2) Is the frequency of sequences following this pathway likely to be sufficiently large to warrant additional plant damage states? and

(3) Can a conservative choice be made which allows for branch suppression which is not likely to significantly impact the overall results. For example, on the North Anna PDS Logic Diagram shown on Figure 4.3.3-1, branching is suppressed under the Station Blackout heading for large LOCA type sequences based on the relative frequency of these LOCAs with and without AC power.

The initial logic diagram for the North Anna plant damage state groups is shown in Figure 4.3.2-1. The basis for the frequencies shown is described in the next Section. The associated rules for assigning the Level 1 sequences to Plant Damage States are given in Table 4.3.2-1. These rules are shown in NUCAP+ (Fulford, 1991) format and the diagram is drawn with the assistance of NUCAP+. The basis for the assignment rules for each criterion is given in detail below. The endpoints of the logic diagram represent individual plant damage states and the pathway through the diagram (i.e., the set of decision paths taken at each decision branch) define the attributes for each plant damage state. Fifty eight (58) individual plant damage states were defined for the NAPS. As described in the next section, the diagram was then simplified, on frequency grounds, to twenty five (25) states, using the same nine criteria and deleting zero frequency of occurrence states and transferring states with a frequency of less than 10-8 to a similar but higher frequency state.

The rationale for selection of each of the PDS grouping criteria (or parameters) is discussed below. In scanning the rules, the following brief list of NUCAP+ rule syntax and semantics may be useful:

"==" Denotes an equality test. If the item in front is equal to the item after, then the result is TRUE, otherwise it is FALSE.

"!=" Denotes an inequality test and returns the opposite results from the equality test.

":"	The first item in the test for Level 1 classifications is denoted as "A:ccccc" where 'A:' means a sequence front line function success or failure as recorded in the .SEQ date file of the extended Level 1 core melt sequences and 'ccccc' is the name of the function. The second in the test is either SUCCESS or FAILURE.
"*"	Denotes a Boolean AND conjunction of the tests before and after it. Both tests have to return TRUE for the ANDed tests to be TRUE.
"IF .."	A statement line beginning with 'IF' is evaluated to see if the tests on the line '..' are TRUE. A IF statement line may have more than one AND in it. Subsequent statement rule lines without an intervening THEN line are de facto OR statements.
"THEN aaa"	When a complete statement (line) is evaluated as TRUE then the first subsequent THEN gives the assigned attribute 'aaa' for this RULE. A DEFAULT statement is always TRUE and the attribute is given on the same line. After an attribute is assigned no further processing of the rule is performed, so that the ordering of rule statements is important.

Containment Bypass (CONBYPASS)

This parameter is used to divide the Level 1 sequences into by-pass and non-bypass groups. Furthermore, the containment bypass sequences are subdivided into interfacing system LOCAs ("EVENT V") and steam generator tube ruptures ("SGTR") with a direct pathway from the RCS to the environment - e.g., failed open secondary SRVs. The containment bypass sequences are distinctly different from non-bypass sequences in that there exists a direct flow pathway from the primary system to outside the containment boundary which bypasses the main containment region. Hence, radionuclides (released from the core/primary system) are not attenuated by the natural processes and engineered safety systems in containment. Consequently, bypass sequences can result in relatively large source term releases early in time. The interfacing system LOCA and SGTR bypass sequences are separated into different groups because the radionuclide release pathways for these two groups of sequences are distinctly different.

For the SGTR sequences the pathway includes the reactor coolant system (RCS), steam generator (SG) secondary side, secondary steam line and safety/relief valves. For the interfacing system LOCA sequences and pathway is RCS, low pressure injection (LPI) system piping and the safeguards building (where the break location may be submerged). Strictly speaking a containment event tree is not

required for these sequences since containment phenomena are largely irrelevant or unimportant. For the interfacing system LOCA sequence however, a "CET" which considers important safeguards building phenomena (such as whether or not the break location is submerged) is necessary to assess the effectiveness of the safeguards building in attenuating radionuclides.

PDS Sequence Classification Rule for CONBYPASS

```
IF A:VX == FAILURE;
THEN EVENT V;
IF A:T7 == FAILURE * A:P  == FAILURE;
IF A:T7 == FAILURE * A:SGI== FAILURE;
IF A:T7 == FAILURE * A:O  == FAILURE;
IF A:T7 == FAILURE * A:D1 == FAILURE  * A:L == FAILURE;
THEN SGTR;
DEFAULT NO BYPASS;
```

Core damage sequences initiated by a SGTR (PDS tree initiating event T7) and with failure to isolate the ruptured steam generator (failure in heading SGI or O), with failure of auxiliary feedwater (event L) and concurrent loss of injection (D1), and with failure of feed and bleed (event P) will follow the PDS "CONTAINMENT BYPASS" heading "SGTR" branch. All core damage sequences initiated by an interfacing systems LOCA (initiating event V) will follow the PDS containment bypass "EVENT V" branch. All other core damage sequences will follow the "NO BYPASS" branch. Note that only one PDS group exists for the SGTR sequences and one group for the event V sequences. This implies that all SGTR and V sequences in these groups will be treated identically using a representative sequence without differentiating other PDS grouping attributes (e.g., high or low RCS pressure). The discussions below on the remaining eight criteria are thus only applicable to the non-bypass sequences.

Containment Status Before Core Melt (CONISOLAT)

This parameter segregates the Level 1 sequences into groups based on the status of the containment leak boundary at the time of core damage. The "not isolated" branch includes sequences with containment failure prior to core damage (the so-called "core vulnerable" sequences referred to above) as well as the sequences with an un-isolated containment. With the containment not isolated, early and relatively large releases of radionuclides from the plant are possible. If the containment is not isolated the most important additional system consideration from the standpoint of the radionuclide source term is whether the Quench or Recirculation sprays function. Consequently, for sequences which are not isolated this is the only other grouping parameter which is considered. Hence, all sequences with containment isolation failure are grouped into one of two PDS groups.

PDS Sequence Classification Rules for CONISOLAT

```
IF A:H1 == FAILURE;
IF A:H2 == FAILURE;
THEN ISOLATED;
IF A:IS == FAILURE;
IF A:D3 == SUCCESS * A:Rs == FAILURE;
IF A:D3 == SUCCESS * A:Ch == FAILURE;
IF A:H2 == SUCCESS * A:Rs == FAILURE;
IF A:H2 == SUCCESS * A:Ch == FAILURE;
IF A:H1 == SUCCESS * A:Rs == FAILURE;
IF A:H1 == SUCCESS * A:Ch == FAILURE;
THEN NOT ISOLATED;
IF A:IS != FAILURE;
THEN ISOLATED;
```

Since containment isolation failure is not strongly dependent on other systems considered in the PDS event trees it was determined by the Level 1 analysts that containment isolation failure could be evaluated independently from the PDS trees. Furthermore, since the NAPS containment is subatmospheric the probability that the containment is not isolated is very low. The probability of containment isolation failure has been assigned a constant value for all sequences. Two pseudo-sequences (initiator IS) were added to the Level 1 core melt sequence data file (.SEQ). One sequence accounts for the non-isolated sequences with sprays operational, and the other for the non-isolated sequences with spray failure (Rs). In particular all sequences with initiator IS are classified as NOT ISOLATED.

The sequences with containment failure before core melt require that injection/recirculation be successful and that containment heat removal be failed. All sequences with low or high head pressure recirculation failure (H1, H2) are classified into the ISOLATED (intact) state. All sequences with injection/recirculation success (D3,H1,H2 success) and containment heat removal failure either because of spray (Rs) failure or heat removal function (Ch) failure are classified into the NOT ISOLATED group.

All other sequences are classed as ISOLATED.

Transient or LOCA Type (TRANLOCA)

This parameter is used to separate transient sequences from LOCA type sequences and to further subdivide large LOCA sequences from the small/intermediate LOCAs. The major reasons for the use of this parameter for grouping are: 1) to aid in the subsequent classification of sequences by RCS pressure, 2) to distinguish sequences with distinctly different key event timing, and 3) for

radionuclide release and transport behavior differences. Small and intermediate LOCAs have been combined since their containment accident progression is expected to be similar.

PDS Sequence Classification Rules for TRANLOCA

```
IF A:A   == FAILURE;
IF A:RX  == FAILURE;
THEN LARGE LOCA;
IF A:S1  == FAILURE;
IF A:S2  == FAILURE;
IF A:T7 == FAILURE;
IF A:Q == FAILURE;
IF A:P == SUCCESS;
IF A:S1c == FAILURE;
IF A:T4 == FAILURE * A:O == FAILURE;
IF A:T6 == FAILURE * A:O == FAILURE;
IF A:T8 == FAILURE * A:O == FAILURE;
IF A:T1Tr == FAILURE * A:O == FAILURE;
IF A:T2Tr == FAILURE * A:O == FAILURE;
IF A:T2ATr == FAILURE * A:O == FAILURE;
IF A:T3Tr == FAILURE * A:O == FAILURE;
IF A:T9ATr == FAILURE * A:O == FAILURE;
IF A:T9BTr == FAILURE * A:O == FAILURE;
THEN SMALL/MED LOCA;
DEFAULT TRANSIENT;
```

All core damage sequences initiated by a large break LOCA (PDS tree initiating event A) or a vessel rupture (RX) will follow the PDS diagram "LARGE LOCA" branch. All core damage sequences initiated by a small LOCA (initiating event S2), or by a medium LOCA (initiating event S1) will follow the "SMALL/MED LOCA" branch. All core damage sequences that develop a permanent RCS pressure boundary failure will also follow the "SMALL/MED LOCA" branch. These include those with RCS boundary not intact (Q failure), seal LOCAs (S1c failure), and all SGTRs (initiator T7) that result in successful steam generator isolation but result in core damage (these are all the T7 sequences not already classed as SGTR by the rule for bypass sequences given above). Core melt sequences with feed and bleed success (P success) imply that the primary relief valves are open and the system can stay at low pressures and hence are SMALL/MED LOCA type. Reactor coolant pump seal failures (failure of function O) for the transients with loss of seal cooling (T4), loss of service water (T6), and loss of emergency switchgear room cooling (T8) are classified into the SMALL/MED LOCA type also.

All remaining sequences are classified as TRANSIENT type.

Station Blackout Type (SBO)

This parameter is used to distinguish sequences with effective total loss of AC power from other sequences. It is selected as a grouping parameter for several reasons. First, total loss of AC power results in a sequence without any containment safeguards (sprays or containment heat removal). Second, past studies at the similar Surry plant (NRC, 1990, NUREG-1150) and other plants indicate that station blackout sequences will be important contributors to core damage and offsite risks. Third, power recovery subsequent to core damage allows for: 1) the possible restoration of in-vessel injection which may terminate the accident and prevent vessel failure or 2) restoration of Quench or Recirculation sprays and containment heat removal in sufficient time to prevent containment failure or mitigate the source term.

PDS Sequence Classification Rules for SBO

```
IF A:T1A   == FAILURE;
IF A:T8    == FAILURE * A:RC1 != SUCCESS;
IF A:T6    == FAILURE * A:RC1 != SUCCESS;
IF A:T1Tr  == FAILURE * A:RC1 != SUCCESS;
IF A:T2Tr  == FAILURE * A:RC1 != SUCCESS;
IF A:T2ATr == FAILURE * A:RC1 != SUCCESS;
IF A:T3Tr  == FAILURE * A:RC1 != SUCCESS;
IF A:T9ATr == FAILURE * A:RC1 != SUCCESS;
IF A:T9BTr == FAILURE * A:RC1 != SUCCESS;
THEN YES;
DEFAULT NO;
```

All transient core damage sequences initiated by station blackout (initiator T1A) follow the PDS diagram "STATION BLACKOUT" heading "YES" branch. Transients with loss of switchgear room cooling (T8) and transients with loss of service water (T6), both without power recovery before core damage (RC1 is not success) will also follow the "YES" branch. All other transient core damage sequences will follow the "NO" branch for this heading. Loss of switchgear room cooling leads to loss of all AC power and hence T8 sequences are grouped with station blackout sequences.

Power Recover (POWRECOV)

This parameter is used to identify station blackout type sequences with recovery of offsite AC power (subsequent to core damage) within a time period judged to be prior to either vessel failure and/or containment failure. Note that recovery of the diesel generators is not considered in the PDS event trees and that power recovery is defined solely as offsite power recovery. Three possible branch pathways are evaluated: 1) "PRIOR RV FAILURE", 2) "PRIOR CONTAINMENT FAILURE" and 3) "NO POWER RECOVERY". Power recovery subsequent to core damage allows for: 1) the possible

restoration of in-vessel injection which may terminate the accident and prevent vessel failure or 2) restoration of Quench or Recirculation sprays and containment heat removal in sufficient time to prevent containment failure or mitigate the source term. This parameter is also used to identify loss of switchgear room cooling transients with recovery of room cooling.

MAAP calculations indicate that there will be many hours between core damage and the time when the containment integrity is first threatened from long term steam/non-condensible gas pressurization. We have conservatively chosen to use the 5% failure pressure to assess time available for power recovery prior to containment failure. For the purpose of estimating recovery probabilities in the Level 1 PDS event tree analysis 1 hour and 18 hours were used as the representative time periods for power recovery prior to vessel failure and prior to containment failure.

PDS Sequence Classification Rules for POWRECOV

```
IF A:B    == SUCCESS;
IF A:B1   == SUCCESS;
IF A:RC2  == SUCCESS;
THEN PRIOR RV FAIL;
IF A:B2   == SUCCESS;
IF A:RC3  == SUCCESS;
THEN PRIOR CONT FAIL;
DEFAULT NO POWER REC;
```

All station blackout type sequences with AC power recovery prior to core damage (event B success) or with power recovery prior to vessel failure (event B1 success or RC2 success) follow the branch "PRIOR RV FAIL". SBO sequences with power recovery prior to containment failure (event B2 success or RC3 success) are follow the "PRIOR CONT FAIL" branch. All other SBO type sequences are classified into the "NO POWER REC" state.

Recirculation Sprays (RECSPRAYS)

Operation of the Quench/Recirculation sprays provide several important functions which impact containment accident progression, containment loading and the radionuclide source term. First, operation of the recirculation sprays is necessary for long term containment heat removal. Second, with or without the containment heat removal function available, operation of the sprays will attenuate fission products released to the containment atmosphere and greatly reduce the source term. The sprays also provide a source of cooling water to debris in the reactor cavity or on the lower containment floor enhancing the possibility of cooling the debris and preventing debris concrete attack and the release of radionuclides and non-condensible and combustible gases. To be considered successful for the purpose of this grouping the sprays

must operate during periods of time when fission product release is occurring and when containment heat removal is required. This implies that long term operation of at least one train of the inside or outside recirculation spray systems is required. Successful operation of (only) the Quench (injection) Spray System does not meet the above requirements for successful spray operation. Sequences with only Quench spray injection system operation will be grouped into PDS groups with sequences with no sprays.

PDS Sequence Classification Rules for RECSPRAYS

```
IF A:Rs     == SUCCESS;
IF A:SPRAY == SUCCESS;
THEN YES;
DEFAULT NO;
```

All core damage sequences with success for PDS event tree function "RECIRCULATION SPRAY AVAILABLE" (event Rs) will follow the "YES" branch for the PDS grouping diagram CONTAINMENT RECIRCULATION SPRAYS heading. All other core damage sequences will follow the "NO" branch.

Longer term failure of the Quench or Recirculation Spray Systems due to specific energetic containment events (e.g., hydrogen burns) or due to the general containment environment is evaluated in the containment event tree.

Containment Heat Removal (CNHEATREM)

Operation of containment heat removal (i.e., operation of at least one train of recirculation sprays with a functional heat exchanger) is necessary to prevent long term containment overpressure failure from steam generation and high containment temperature. Successful operation of containment heat removal requires that heat removal be established prior to the containment reaching a pressure where containment integrity is threatened (taken to be 93 psig). This particular timing requirement generally affects SBO type sequences where success of offsite power recovery (prior to containment failure) is based on the time period from onset of core damage to the time when the containment integrity is initially threatened.

PDS Sequence Classification Rules for CNHEATRM

```
IF A:Ch   == SUCCESS;
THEN YES;
DEFAULT NO;
```

All core damage sequences with success for PDS event tree function "CONTAINMENT HEAT REMOVAL AVAILABLE" (event Ch) will follow the "YES" branch for the PDS grouping diagram heading "CONTAINMENT HEAT

REMOVAL". All other core damage sequences will follow the "NO" branch. Since recirculation spray operation is required for containment heat removal to be successful this heading is not branched for sequences without recirculation sprays and is set to "NO".

RCS Pressure During Core Damage/At Vessel Failure (RCSPRESS)

The reactor coolant system pressure during core damage and at the time of vessel failure can have a major impact on several potentially important containment events. High RCS pressures during core heatup and core damage facilitate natural circulation heat transfer from the core to the hot leg which increases the potential for temperature induced hot leg, surge line or steam generator tube failure. Elevated pressures at the time of vessel rupture may result in entrainment of the core debris out of the reactor cavity and increase the potential for debris fragmentation and dispersal into the main containment gas volume thus increasing the potential for direct containment heating (DCH). RCS pressure during core degradation can also impact the timing and magnitude of fission product release from the primary system. Four pressure regimes have been identified as being significant. These are:

Pressure Regime	Pressure Range (psig)
LO	< 200
LO HI	200 - 2000
HIGH	2000 - 2335
HI HI	> 2335

The reasons for this selection of pressure regimes for use as PDS grouping characteristics is discussed below. Energetic dispersal of the debris out of the reactor cavity following vessel failure is not expected for RCS pressures below about 200 psi, whereas for RCS pressures above 200 psi debris entrainment and dispersal out of the cavity and DCH are potentially important processes. 200 psi was the threshold pressure for these phenomena in NUREG/CR-4551. 2000 psi was judged by the NUREG-1150 In-Vessel Expert Panel as the lowest pressure where induced hot leg or surge line creep rupture failure was credible (though unlikely). At very high RCS pressures in the range of the pressurizer relief/safety value setpoints (> 2335 psia) the NUREG-1150 experts panel judged that induced hot leg or surge line failure was likely and that induced steam generator rupture was possible (though highly unlikely).

<u>PDS Sequence Classification Rules for RCSPRESS</u>

(no rule necessary)

All LARGE LOCA type sequences are classified as LO LO type. All SMALL/MED LOCA type sequences are classified as LO HI type. This stems from the definition of the small LOCA primarily as being

insufficient in size to remove decay heat. Various MAAP runs indicate that for the SMALL/MED LOCA type the pressure regime is generally below 2000 psi. The pressure can be expected however to be above 200 psi after core slump because of steam generation in the lower head plenum. That is, SMALL/MED LOCA (LO HI) sequences may result in a DCH event but will not induce a hot leg or surge line failure. The major effect of such a failure, which might occur if the pressure were HIGH, would be to lower the pressure into the LO HI or LO LO range prior to vessel breach. So, even if certain of the SMALL/MED LOCA type sequences in fact were to result in a HIGH pressure regime, the end result is essentially the same. Because of the classification of transients with an RCS boundary failure into the LOCA class, the remaining transients must be of the HI HI type, that is, steam must be escaping from the RCS only through a properly cycling primary relief or safety valve.

As discussed in the preceding paragraph, all sequences are assigned into pressure regimes based only on whether they are LARGE LOCA type, SMALL/MED LOCA type, or TRANSIENT type. Therefore there is no further classification needed and no PDS selection rule is needed for this criterion. It is also worthwhile noting that no sequences are classified into the HIGH regime, for the reasons noted above.

Status of In-Vessel Injection

The status of in-vessel injection at the time of core damage is important for several reasons. If in-vessel injection is available during the period of core uncovery, core damage may be limited and vessel failure prevented. This situation would be the case for a large break LOCA sequence with operable LPI but with all accumulators failed. For this sequence the Level 1 success criteria indicate core damage occurs. MAAP calculations confirm that core damage is likely, however, these calculations also indicate that the amount of fuel damage would be limited (peak fuel temperature of approximately 2700°F) and termination of the accident progression in-vessel would be highly likely. (See MAAP Level 2 Calculation No. 5 in NAF Analysis File 326MAF.7.) If the RCS pressure is elevated above the LPI injection threshold pressure (approximately 165 psig) but the system is available (deadheaded) it could provide in-vessel injection if the RCS is depressurized prior to RV failure (such as by an induced hot-leg rupture). In addition, with the LPI operating an additional source of cooling water is available to the cavity debris following vessel failure. (This additional source of water to the cavity was not considered in the containment analysis since it is expected that the number of core melt sequences with operational LPIS after vessel failure and failed recirculation sprays will be small.) The in-vessel injection systems may also be available following off-site power recovery for station blackout sequences. The four possible branches for this heading are thus:

ON	(available and operating)
LPI DEADHEAD	(available but cannot inject because of high RCS pressure)
RECOVERED	(injection recovered subsequent to core damage but prior to the anticipated time of RV failure)
FAILED	(never available)

PDS Sequence Classification Rules for INVESSINJ

```
IF A:H1 == FAILURE;
IF A:H2 == FAILURE;
THEN FAILED;
IF A:A == FAILURE * A:H1 == SUCCESS * A:Dh == SUCCESS;
IF A:A == FAILURE * A:D3 == SUCCESS * A:Rs == SUCCESS;
IF A:RX== FAILURE * A:D3 == SUCCESS * A:Rs == SUCCESS;
IF A:RX== FAILURE * A:Qs == SUCCESS * A:Rs == SUCCESS;
IF A:S1 == FAILURE * A:H1 == SUCCESS;
IF A:S1 == FAILURE * A:H2 == SUCCESS;
IF A:S2 == FAILURE * A:H2 == SUCCESS;
THEN ON;
IF A:A == FAILURE * A:H1 == SUCCESS * A:Dh == FAILURE;
IF P:SBO==NO *P:RCSPRESS!=LO LO * A:H1==SUCCESS;
IF A:S2 == FAILURE * A:H1 == SUCCESS;
THEN LPI DEADHEAD;
IF P:POWRECOV==PRIOR RV FAIL*P:RCSPRESS != LO LO*A:H1==S;
THEN RECOVERED;
DEFAULT FAILED;
```

This criteria is based on the status of the SI systems in recirculation. If there is a failure in recirculation mode (events H1 or H2 as failures) then the systems are considered FAILED. LARGE LOCA type core damage sequences with success for PDS event tree function "LOW HEAD SAFETY INJECTION AVAILABLE LATE" (event D3) or "LOW HEAD RECIRCULATION AVAILABLE LATE" (event H1) are considered "ON"except that the LARGE LOCA sequences with failure to switch to hot leg recirculation (event Dh is a failure) is assigned to LPI DEADHEAD, as the reason for core melt is plugging in the upper head. Other LOCA-initiated sequences (S1,S2) are considered ON if H1 or H2 (high head recirculation) is a success. For non-station blackout types (SBO is NO) with non LO LO pressures and with low head recirculation success (H1 success), and particularly for S1 sequences with H1 success, the systems are considered LPI DEADHEADed. For station blackout type sequences with power recovery prior to reactor vessel failure (PDS state PRIOR RV FAIL for the power recovery criterion), the LPI system available (H1 success), and pressure higher than LO LO, the systems are

considered RECOVERED. All other sequences are considered to have invessel injection FAILED.

4.3.3 PDS Logic Diagram and PDS Characteristics

The quantified Plant Damage State Logic diagram for North Anna is shown in Figure 4.3.3-1. The endpoints of the logic diagram represent the significant individual plant damage states and the pathway through the diagram (i.e., the set of decision paths taken at each decision branch) define the attributes for each plant damage state. Twenty five individual plant damage states are defined. This diagram is a simplified derivative of the initial diagram. All Level 1 sequences are represented in this diagram, the difference being that some very low frequency states have been rebinned and zero frequency states deleted. The frequencies shown at intermediate branch nodes on this diagram are merely the sum of the frequency of the branches stemming from this node. They are developed by simply combining branches starting from the "end" or rightmost end of the diagram and working back to the starting point recording the intermediate sums.

The PDS diagram was constructed as follows:

- The Plant Damage State Logic Diagram shown in Figure 4.3.2-1 was quantified. The PDS assignment rules shown in Table 4.3.2-1 were used to assign the sequences to the PDSs of this diagram.

- The frequencies of all the sequences for each of the states were summed. These are shown in the rightmost column of the diagram for each sequence. These frequencies were then used to guide the simplification of the PDS diagram.

- All states with zero frequency were deleted. The 26 states so eliminated are designated with a "D" as the state attribute under the last diagram heading DISPOSTN.

- All states with a frequency of occurrence of less than 1.E-8 were binned with another state of higher frequency that was judged sufficiently similar. The binning target state of each of these very low frequency states is shown as T:## under DISPOSTN, where ## is the receiving state. Then these seven very low frequency states were eliminated as distinct states.

- The Level 1 sequences were then reassigned to the remaining twenty five states of the reduced diagram and the PDS frequencies summed. The PDS state numbers shown on the Level 1 Accident Event Trees (Chapter 3) are the ones assigned in this step.

Plant Damage States 1 and 2 contain sequences with the Containment not intact at core melt ("core vulnerable" sequences) and sequences with failure of Containment isolation. PDS 3 through 7 represent station blackout (SBO) type sequences without a significant breach of the primary system pressure boundary prior to core damage. PDS 3 represents all SBO type sequences with power recovery prior to RV failure. PDS 4 through 6 represent SBO sequences with power recovery after vessel failure but prior to the time when the Containment failure would be expected from long term overpressurization. PDS 7 represents SBO type sequences with no power recovery. PDS 8 through 11 contain all transient initiated sequences other than those of the station blackout type. PDS 12 and 13 represent large break LOCAs including vessel ruptures. PDS 14 through 23 represent small and intermediate break LOCA type sequences, which includes the small and medium LOCA initiated sequences, the seal LOCAs including induced seal failures, and stuck open valve sequences. PDSs 14 through 19 represent the SBO subset of this type, with the subdivisions based on power recovery as for the non-LOCA SBO states. PDS 20 through 23 represent the small/medium LOCA types with AC power available. PDS 24 contains the Containment bypass interfacing system LOCA (Event V) sequences and PDS 25 contains the SGTR sequences.

4.3.4 Plant Damage State Frequencies and Dominant Sequences

Figure 4.3.3-1 presents the North Anna PDS results. Plant damage states ranked by frequency are listed in Table 4.3.4-1 and are illustrated in Figure 4.3.4-1. Table 4.3.4-2 shows the Level 1 sequences that contribute to each plant damage state at approximately the 1E-8 level or higher.

Inspection of individual plant damage state attributes yields the following results. Containment bypass (SGTR and Event V) sequences in PDS 24 and 25 make up approximately 13% of the total plant damage state frequency, and Containment not isolated/not intact sequences in PDS 1 and 2 0.22%.

The observations made in the next paragraph are for the Containment intact/isolated/not bypassed sequences only (PDSs 3 through 23) which are the remaining 87% of the core damage frequency.

Transient type sequences contribute 30% to the core damage frequency, large LOCA types 6%, and small/medium LOCA types make up the remaining 51%. Station blackout (SBO) type sequences account for 34% of the total. For SBO type sequences 24% have power recovery subsequent to core damage but prior to the estimated time for vessel failure, 71% have power recovery after vessel failure but prior to the estimated time when long term Containment over-pressure failure might first occur and for 5% of the SBO sequences power is never recovered. The percentage of sequences with Recirculation Sprays available and with Containment heat removal

operable is 77%. Hence sprays and Containment heat removal are available for the majority of sequences.

RCS pressure during core damage and at the time of reactor vessel failure is an important parameter for assessing Containment accident progression. The overall breakdown for fraction of all plant damage state sequences in each pressure regime is shown below (for the containment intact/not bypassed sequences).

High High	($P \geq 2335$ psig)	47%
High	($2000 < P < 2335$)	0%
Intermediate	($200 < P < 2000$)	45%
Low	($P < 200$)	8%

4.4 CONTAINMENT BUILDING FAILURE CHARACTERIZATION

The North Anna Containment Building consists of a 126-foot inside diameter reinforced concrete right circular cylinder with a flat base and hemispherical dome. The cylindrical portion of the Containment sits on a basemat that is 10 ft. thick. The wall of the cylinder is about 4.5 ft. thick and rises 127 feet above the top of the Containment foundation mat. The dome has an inside radius of 63 ft. and is about 2.5 ft. thick. The inner surface of the Containment is a liner of welded steel plate, which forms the pressure boundary. Figure 4.1.1-1 shows a section through the North Anna Containment Building. The volume is approximately 1,825,000 ft^3, and the design pressure is 45 psig. Due to conservatism in design and construction, most estimates of the failure pressure are between two and three times the design pressure. The mean of the aggregate distribution for the failure pressure of the North Anna Containment Building is estimated to be approximately 128 psig. The size and strength of the Containment Building means that it can absorb a great deal of energy without failing.

When the reactor is operating, the pressure inside the Containment is kept at approximately 10 psia, which is about 5 psia below ambient atmospheric pressure. The implication of this is that the existence of pre-existing leaks of a size that would be significant for the Containment analysis is negligible. The vacuum pumps that keep the Containment atmosphere below ambient pressure are limited in their capacity, so significant leakage past the equipment hatch, personnel airlocks or through other penetrations would be quickly discovered. For a leak opening with an effective leak area greater than about 0.25 square inches the vacuum pumps would be unable to keep the pressure at 10 psia. The Technical Specifications prevent plant operation much above this pressure, so the rise in Containment pressure would force the plant to shut down until 10 psia could be maintained in the Containment. The fact that the Containment is maintained below ambient pressure also means that

very few lines are normally open into the Containment during normal operation; thus, the probability of isolation failure is low.

The North Anna IPE relies heavily on the extensive work done for the NUREG-1150 project. Part of that work was to define the Containment failure modes and the Containment over-pressure fragility curve for the Surry plant. A containment failure analysis comparison (Virginia Power, 1992) shows that the containments at North Anna are similar to those at Surry and concludes that the Surry containment failure criteria can be applied to the North Anna containments.

4.4.1 Containment Building Failure Modes

The Containment Building failure modes identified for North Anna are based on NUREG-1150 results. Several failure modes were considered by the experts during the NUREG-1150 elicitation process. NUREG-1335 gives a list of potential Containment failure modes and mechanisms and states that all of these failure modes and mechanisms were considered in NUREG-1150. The following text discusses each of these items.

Direct Bypass

Direct bypass of the Containment is considered in the NUREG-1150 analysis and in the North Anna IPE. In each analysis the bypass sequences include both V-sequence and Steam Generator tube rupture (SGTR) sequences that are not isolated.

Failure to Isolate

The failure to isolate Containment leads to direct release of radioactivity and is of obvious importance. In NUREG/CR-4550 the probability of failure of Containment isolation was determined on the basis of analytical significance rather than fault tree analysis. A leak size greater than 0.1 ft^2 is required to prevent Containment overpressurization (from long term steam generation). A review of the leakage paths reveals that the design of the North Anna Containment Building precludes operation with a leak anywhere near the above size. Therefore, it was concluded in NUREG/CR-4550 that the probability of pre-existing leakage of sufficient size to impact Containment pressurization was negligible.

The NUREG/CR-4550 analysts did not consider failure of Containment isolation from a source term point of view, however. A leak in Containment either at the time of the accident or resulting from the failure of the isolation paths to close may result in a significant release pathway especially if the path is in direct contact with the Containment atmosphere. The North Anna IPE has

considered this issue. A base failure probability has been assumed that corresponds to the failure of a normally open line in contact with the Containment atmosphere. It has been further assumed that the normal isolation valve arrangement is two valves in series per line. Finally, there are four lines open to atmosphere. Given a generic failure-to-close probability of 1.1E-2/valve the overall failure probability for these lines is 4.3E-4. This failure probability was combined with the contribution to core damage frequency for all non-bypass sequences to obtain the core damage frequency of non-isolated sequences. This frequency was divided into the frequency of non-isolated sequences with the recirculation sprays operating and that without the sprays operating for use in the plant damage state grouping logic.

Vapor Explosions

NUREG-1150 considered steam explosions originating in-vessel(the classic alpha-mode failure) or ex-vessel. Alpha-mode failures were considered by the Steam Explosion Review Group (NRC, 1985). Details of their results are presented below. Ex-vessel steam explosions were dismissed for the Surry plant in NUREG-1150 because steam explosions in the cavity would not directly contact structures that are both vulnerable and essential to the containment function. Based on the NUREG-1150 results, Containment failure resulting directly from ex-vessel steam explosions was not considered in the North Anna IPE.

In the NUREG-1150 Surry study, about half of the mean frequency of early Containment failure conditional upon core damage is associated with the alpha-mode scenario, this total mean frequency is less than 10^{-2}. The estimates for probability of alpha mode containment failures given in NUREG-1150 were also used in the North Anna IPE.

Combustion Processes

The combustion of hydrogen prior to reactor vessel breach was treated in NUREG-1150 as an expert elicitation issue. However, it was decided that hydrogen combustion is of much greater concern for lower capacity containments [Boiling Water Reactor (BWR) plants and ice condenser PWR plants] than it is for large high capacity containments such as Surry or North Anna. In the words of NUREG-1150: "... the importance of early hydrogen combustion to the uncertainty in reactor risk for these plants is minor in comparison to that observed in the Grand Gulf and Sequoyah analyses."

Nonetheless, hydrogen combustion was considered in the Surry NUREG-1150 accident progression analysis. Both early and late combustion were considered. Since the Surry Containment Building was found to be robust by the structural experts, the possibility of Containment

failure prior to reactor vessel failure is so remote as to be considered negligible and was not included in the NUREG/CR-4551 Surry Containment Building event analysis. The failure of containment due to a hydrogen burn at the time of reactor vessel failure or subsequent to vessel failure was considered likely enough to be included. In the North Anna IPE Containment analyses the impact of hydrogen combustion on Containment overpressurization was considered at vessel failure and late in the accident sequence after vessel failure. However, the more likely late Containment failure modes are basemat melt-through or gradual steam overpressurization.

Steam Overpressurization

Gradual pressurization of the Containment Building would result from the protracted generation of steam and non-condensible gases from the interaction of molten core material with water on the Containment floor or with the concrete basemat. This pressurization process could last from several hours to several days, depending upon accident-specific factors such as the availability of water in the Containment and the operability of engineered safety features.

Gradual Containment pressurization by steam production and from the non-condensible gases generated during debris concrete attack was considered in the North Anna IPE.

Core-Concrete Interaction (Basemat Melt-through)

The North Anna design is such that water fills the reactor cavity if the sprays operate. Also, the design of the sump is such that the Containment floor is covered with water when the RWST empties. However, if the Recirculation Sprays do not function in the long term (and ex-vessel cooling is not available) or the debris is not in a coolable configuration then basemat melt-through may occur. Therefore this phenomenon was considered in the North Anna IPE Containment analysis.

Blowdown Forces (Vessel Thrust Force)

Failure of the Containment Building as a result of gross displacement of the reactor vessel (above the shield wall) was considered in the NUREG/CR-4551 accident progression analysis. However, the assigned probability for this event was sufficiently small that it made a negligible contribution to the probability of early containment failure. This mode of containment failure was not considered in the North Anna IPE Containment analysis.

Liner Melt-Through (Direct Contact of Containment Shell with Fuel Debris)

This issue is of primary concern to BWR plants because of the drywell design. This mode of failure was not considered in the North Anna IPE since the pathways for debris transport out of the reactor cavity are to interior Containment Building compartments and away from the Containment wall.

Failure of Containment Building Penetrations

Failure of Containment Building penetrations (electrical, fluid, equipment hatch, personnel hatch, etc.) was evaluated in the NUREG-1150 analysis and was judged to be significantly less important than over-pressure failure of the cylinder wall. Based on the NUREG-1150 results this failure mode was not explicitly included in the North Anna IPE.

4.4.2 Containment Building Over-pressure Fragility

The Level 2 analysis considers the possibility of the Containment Building failing under various accident scenarios. In order to be comprehensive, failures resulting over the spectrum of possible pressures must be considered. The NUREG-1150 work characterized Containment failure using four parameters: likelihood of failure as a function of Containment pressure, failure size, location of failure and timing. Likelihood of failure is the primary parameter of interest in the study. Failure size is important because the larger the hole the faster the release of radionuclides following an accident. The location of the failure is important because the retention of radioactive materials can be dependent on this parameter. The longer the materials can be retained inside the Containment before escaping the larger the reduction in source term to the environment since the radionuclides are removed from the Containment atmosphere by natural processes and ESFs. For a similar reason timing is also important.

The Containment failure pressure is a question in the NUREG/CR-4551 Accident Progression Event Tree (APET). The question was answered using expert elicitation. A panel of structural experts were asked to provide a distribution for the failure pressure of the Surry Containment Building, and identification of the failure modes. Since the probability of global detonations was judged to be quite small, only static loads were treated in the NUREG-1150 structural analysis. The experts used available structural calculations for Surry as well as those for plants with similar containment designs such as Indian Point. All of the important failure locations identified for Surry provided direct pathways to the outside environment. Yielding of one of the steel hoop bars that reinforce the vertical concrete wall was identified as a likely mode of

failure by all the experts. All experts identified the intersection of the wall with the dome as a likely location for failure. The experts identified leakage due to the formation of a tear in the steel liner as the most likely failure mode.

The distributions of failure pressure determined by the NUREG-1150 experts are presented on Figure 4.4.2-1. The aggregate distribution, also presented in the figure, was determined by weighing equally the individual distributions of the structural experts. The mean and the median failure pressures from the experts aggregate distribution are about 128 psig.

The 5th-95th percentile range of potential failure pressures extends from approximately 93 psig to 149 psig. Leakage was assessed as the most likely mode of failure for breaches occurring below 135 psig, while ruptures were the most likely modes of failure for failure pressures in the 135-150 psig range. The dominant mode of failure above 150 psig was assessed to be catastrophic rupture.

NUREG-1150 characterized the above failure sizes as follows:

> A leak was defined as a containment breach that would arrest a gradual pressure buildup, but would not result in containment depressurization in less than 2 hours. The typical leak size was evaluated for all plants to be of the order of 0.1 ft^2.
>
> A rupture was defined as a containment breach that would arrest a gradual pressure buildup and would depressurize the containment within 2 hours. For all plants, a rupture was evaluated to correspond to a hole size in excess of approximately 1.0 ft^2.
>
> A catastrophic rupture was defined as the loss of a substantial portion of the containment boundary with possible disruption of the Piping Systems that penetrate or are attached to the containment wall.

The above failure modes, failure pressures and failure sizes were adopted for the North Anna IPE analysis.

4.5 CONTAINMENT BUILDING EVENT TREES

A containment event tree (CET) is a logic model to delineate the possible paths that an accident sequence may progress along given an initial set of conditions defined by a plant damage state. The headings in the NAPS CETs consist of only the important "events" which can lead to significantly different outcomes in the sequence progression where the major outcomes of interest relate to timing and mode of containment failure and the atmospheric release of

radionuclides (the source terms). The events in the containment event tree generally are chosen to: 1) represent the uncertainties in physical phenomena (e.g., direct containment heating, containment loading); 2) assess operator recovery and mitigation actions, 3) assess consequential failure of important systems given the occurrence of specific physical phenomena (e.g., H_2 burns) or as a result of the general severe accident environment.

The number of headings that are required for a containment event tree to depict the important accident progression possibilities and to define the spectrum of possible outcomes need not be large. Additional event detail required for the quantification of CET events have been relegated to decomposition event trees (DETs). These quantification aids are further discussed below.

4.5.1 Containment Event Tree Development

Containment event trees have to be developed for each plant damage state. The top events in the CET consist of phenomenological events or processes and consequential systems failures resulting from physical phenomena or the accident environment which are considered to be important to the definition of the source term and the time, mode, and location of containment failure. The severe accident phenomena and containment events specified in Generic Letter 88-20 have been evaluated for inclusion in the CET. Also considered were the detailed set of events developed for NUREG-1150 and for NUREG/CR-4551. A review of past PRAs and IDCOR results was performed to identify events which should be included in the CET.

Specific events to be included in each CET were determined to a large extent by the characteristics of the sequences in each plant damage state with which a particular CET is associated. Additional events were identified based on a review of the specific design and operational characteristics of NAPS.

In the CET, events that occur nearly simultaneously and/or have effects that are interrelated are combined into single events. For example events "Direct Containment Heating" and "Mode of Early Containment Failure" were combined since they relate to, or contribute to, over-pressure loading of the containment at the time of vessel failure.

Event timing was a key factor in organizing the events on the CET. The accident progression was divided into distinct time periods for which different phenomenological processes are important and for which different recovery and mitigation actions may be effective. The general time periods considered were:

1. prior to Reactor Vessel failure
2. at or within a few hours of the time of RV failure
3. late - many hours after RV failure

A general containment event tree structure was used to assess containment accident progression for the majority of the plant damage states. Exceptions include the PDS associated with containment bypass sequences (steam generator tube rupture sequences or interfacing systems LOCA sequences) and loss of isolation sequences. For these later classes of plant damage states, special CETs were developed. Although the general CET structure was applicable for most of the PDS, the quantification of the CET was different as a result of differing PDS characteristics. The general CET structure is shown in Figure 4.5.1-1. The events selected for incorporation into the general CET are those judged to be the most important for assessing the NAPS containment accident progression, containment failure and source term. These events are grouped on the tree into the three principle time periods of interest for the analysis shown above.

The following discussion summarizes the events included in the general NAPS CET.

Mode of Induced Primary System Failure

This question asks whether the elevated temperatures and pressures within the Reactor Coolant System following core uncovery can result in failure of the RCS pressure boundary outside of the vessel prior to reactor vessel lower head failure. Three branch possibilities are considered:

1. no induced RCS failure
2. rupture of a hot leg (or the pressurizer surge line)
3. Steam Generator tube rupture(s)

Induced RCS pressure boundary failure is only likely to be important for sequences where the RCS pressure remains elevated during core uncovery and core heatup, since the high pressure conditions enhance natural convection heat redistribution from the core to the hot leg and Steam Generators and the high pressure conditions may lead to failure of these components at elevated temperatures. Each of the possible branch pathways for this event has an important impact on accident progression. Hot leg failures are likely to be of sufficient size (large break LOCA) to cause depressurization of the RCS prior to vessel failure and consequently to greatly reduce the probability that energetic events at vessel failure (e.g., DCH or H_2 burning) will cause Containment failure. Failure of one or more Steam Generator tubes can result in a bypass of Containment if a secondary relief/safety valve opens or if there is significant leakage past the MSIVs. However, unless the number of induced Steam Generator tube failures

is large (> 10), the primary system would not be expected to depressurize prior to reactor vessel failure.

Debris Cooled In-Vessel

Given that core uncovery and some core damage has occurred, this question considers whether the damaged core can be cooled in-vessel and gross damage and vessel failure prevented. For there to be any possibility that the core be cooled in-vessel, a supply of water to the vessel in excess of that required to remove decay heat must be supplied. This requires an absolute minimum of several hundred gpm injection flow. At this minimum flow level the probability of successfully cooling the damaged core in-vessel will be low, even given a core debris configuration favorable to cooling. At substantially higher injection flow rates (several thousand gpm) the probability of cooling the debris under less favorable debris configurations (e.g., at later times with greater amounts of core damage, core slumping and/or core melting) is enhanced.

The plant damage state entry conditions define whether low pressure (or high pressure) injection flow is (or can be) provided. The types of core damage sequences with coolant injection to the vessel following core damage initiation can be divided in two major classes. The first class of sequences are those where the injection flows are insufficient to prevent core damage as defined by the Level 1 analysis success criteria of limiting peak core temperatures to less than 2200°F (1200°C). An example of this type of sequence is a large break LOCA with successful low pressure injection but with failure of the Accumulators to inject. The second class of sequences are those where there is no coolant injection prior to core uncovery and incipient core damage but where some form of injection is recovered prior to vessel failure. This second class of sequences would include station blackout with late recovery of power and high pressure sequences with failure of high pressure injection followed by late depressurization (either by operator action or as a result of induced hot leg or surge line rupture) followed by successful LPI. The possible branch pathways for this event are:

1) debris cooled in-vessel (no vessel failure), and
2) debris not cooled in-vessel.

If the debris is cooled in-vessel Containment failure is extremely unlikely since only limited hydrogen production would be expected, steam generation will be limited, and DCH is not a possible threat. Furthermore, radionuclide release from the debris will be limited and longer-term revaporization of radionuclides deposited on RCS surfaces will be largely avoided. Hence, because the Containment does not fail and because of the limited radionuclide release, the environmental source terms for core damage sequences successfully

terminated in-vessel are expected to be very small. The sequences of this type are very similar to the TMI-2 accident.

No Alpha-Mode Containment Failure

Postulated alpha mode Containment failures result from large coherent in-vessel steam explosions which fail the reactor vessel and generate a missile (from part of the reactor vessel upper head) with sufficient mass and energy to fail Containment. There is a substantial body of evidence to suggest that in-vessel steam explosions do not represent a credible threat to early Containment failure (i.e., the probability of early Containment failure resulting from in-vessel steam explosions is negligibly small). This opinion appears to be shared by the authors of Appendix 1 to Generic Letter 88-20 (NRC, 1988). However, since in-vessel steam explosions were considered in the NUREG-1150 Containment analysis for Surry (Breeding, 1990, NUREG/CR-4551) and because this event, if it should occur, can result in large and early environmental releases, this event has been included in the North Anna IPE CET.

Experimental evidence and calculations have shown that steam explosions are much less likely at elevated pressures than at low pressure, consequently the probability of an alpha mode Containment failure should be significantly less for high pressure sequences than for low pressure sequences.

The branches for this event are:

1. No alpha-mode Containment failure
2. Alpha mode Containment failure occurs.

Mode of Early Containment Failure

This question determines whether the Containment fails early in time, and if Containment fails, what the mode of Containment failure is. Early Containment failure is defined as shortly before, at, or soon after reactor vessel failure. Early Containment failure can potentially result from a combination of energetic processes and events which may occur at reactor vessel breach. These processes and events include blowdown of the primary system, direct Containment heating (DCH), hydrogen combustion and rapid steam generation in the cavity.

The ultimate Containment strength and the likely failure modes for the North Anna Containment were evaluated by comparing the North Anna Containment characteristics to those for Surry. It was concluded that the characteristics are sufficiently similar that the Surry Containment fragility curve can be used for the North Anna study (Virginia Power, 1992). Therefore, a median failure pressure of 128 psig can be approximated for NAPS. Three failure

modes corresponding to the following rupture sizes were considered possible:

1) a catastrophic rupture (nominal leak size approximately 7.0 ft^2)
2) a rupture (leak size approximately 1 ft^2 or larger)
3) a leak (typical leak size 0.1 ft^2)

The major difference between a rupture and a leak is that a rupture is capable of arresting a gradual pressure rise in Containment and in depressurizing the Containment in less than 2 hours. A leak, would also arrest a gradual pressure buildup but would not result in Containment depressurization within 2 hours. The catastrophic rupture considers a sufficiently energetic event that piping systems which are attached to, or penetrate the Containment wall may be disrupted. The Containment fragility curve 5th to 95th percentile range of potential failure pressures extends from approximately 93 psig to 149 psig. For Surry, the NUREG-1150 (NRC, 1989) Structural Expert Panel estimated that if Containment failure occurred below 135 psig that leakage was the most likely failure mode. Ruptures were the most likely failure mode for failure pressures in the range 135-150 psig. For failure pressures in excess of 150 psig, catastrophic rupture was estimated to be the likely failure mode. These failure ranges are also assumed to be applicable for North Anna.

It should be noted that a fast pressure rise such as from DCH or a hydrogen burn will not be arrested by a small leak. Hence, for these loading conditions, a small leak, if it occurs, may progress to a rupture.

The branches for this event are:

1. No Early Containment Failure
2. Leak
3. Rupture
4. Catastrophic Rupture

No Early Recirculation Spray Failure

The failure of the Recirculation Spray System is included on the Containment event tree because this system provides the heat removal function for the Containment. Hence, without the Recirculation Sprays the pressurization of the Containment will continue unabated once the heat sinks absorb all the energy possible. In addition, operation of the Spray System provides an effective mechanism for fission product mitigation. Early is defined, as in previous event headings, to be before, at, or just after vessel failure.

Initial "failure" of the Recirculation Sprays (as defined by the plant damage state conditions) is most likely to be from station blackout. Since there are four Recirculation Spray pumps and only one is required for successful operation, random system failures are not likely to defeat this function. Other potential failure modes that may lead to failure of the Recirculation Sprays include spray failure as a result energetic Containment failure, a massive blockage of the Containment Sump screens by the core debris or environmental conditions inside the Containment or in the Safeguards Building that result in failure of the Recirculation Spray pumps or pump motors.

The effects of local hydrogen combustion on equipment is accounted for in the quantification of the effect of the containment environment on the spray pumps. Because of the diversity and redundancy of these systems, localized hydrogen burns are not expected to have a significant impact on the failure probability. In addition, since only the catastrophic rupture failure mode of the containment is judged capable of failing the spray nozzle headers, header failure due to hydrogen combustion effects is considered bounded by the spray motor failures.

The two branches for this heading are simply failure or no failure. Failure includes both the spray function and the heat removal function.

Debris Cooled Ex-vessel

This question concerns long term Containment loadings resulting from core debris concrete attack. Debris concrete attack results in concrete degradation and ablation, production of non-condensible and combustible gases, additional heat generation from chemical reactions, changes in the corium mass chemical composition and releases of radionuclides and aerosols.

If the debris is cooled then its only subsequent challenge to the Containment is the continued addition of the decay heat to the cooling water and hence to Containment.

Physically, the debris is not cooled if the debris surfaces that are exposed to the heat-removing medium are not large enough with respect to the heat generating volume to prevent high temperatures being attained. High surface-to-volume ratios imply debris being spread thinly over a large surface area. Additionally, sufficient cooling water must be present. (However, if spread thinly enough, water may not be necessary).

For the North Anna Station it has been determined that the Containment concrete aggregate is basaltic which, unlike limestone aggregate concrete, will not produce much combustible carbon monoxide or non-combustible carbon dioxide upon decomposition. The

concrete does contain the normal amounts of bound and unbound water so that sparging/aerosolization of the debris, metal oxidation, and hydrogen and steam production will occur.

Another important factor is the geometry of the cavity and instrument tube tunnel. Water will enter the cavity only if the Containment injection and/or Recirculation Sprays are operating (or if low pressure injection is operating after vessel failure). For high pressure vessel breach sequences a considerable fraction of the debris may be transported out of the cavity, relocating to the RHR cubicle floor and beyond.

The branches for this event are:

1. Debris Cooled Ex-vessel
2. Debris Not Cooled Ex-vessel

Mode Of Late Containment Failure

The CET heading, mode of late Containment failure, is similar to the heading for early Containment failure. The obvious difference is that the accident has been in progress for a significant amount of time. The time frame for late Containment failure begins many hours after the vessel has failed and continues indefinitely.

The structural analysis discussed in the section entitled mode of early Containment failure is also applicable in this section. The primary cause for failure of the Containment late in time would be from steam overpressurization, resulting from loss of the Recirculation Sprays or Containment heat removal. The possibility of late failure due to a late hydrogen burn is also considered. The branches for this event are:

1. No Late Containment Failure
2. Leak
3. Rupture
4. Catastrophic Rupture

No Late Recirculation Spray Failure

The Recirculation Sprays are the only source of Containment heat removal. Additionally, these sprays provide the source of cooling for the sump water necessary to protect the Low Head Safety Injection pumps from overtemperature failure. The Recirculation Spray System, including the Service Water supply to the Recirculation Spray heat exchangers, is therefore vitally important for preventing both core damage and Containment overpressurization in the late phases of an accident.

The late failure of the Recirculation Sprays may be caused by a catastrophic failure of the Containment or by environmental conditions inside Containment and the Safeguards Building. Harsh environmental conditions can result from high radiation, high humidity, high temperatures, or from effects of local hydrogen burns.

The two branches for this heading are simply failure or no failure. Failure includes both the spray function and the heat removal function.

Containment Long Term Failure

The long term failure of the Containment can result from one of several scenarios. If the Recirculation Sprays fail late in the accident then the Containment will fail from overpressurization at some point in time. If Containment heat removal functions the Containment will not fail due to gradual steam overpressurization. The Containment can also fail due to basemat melt-through (even if the sprays function) if the molten debris is not coolable. Note that if the Containment heat removal function is not available we assume that over-pressure failure will occur and neglect basemat melt-through since the offsite consequences of basemat melt-through would be small compared to over-pressure failure. The two branches for this heading are No Late Late Containment Failure and Basemat Melt-through.

North Anna Containment Event Trees

Figure 4.5.1-1 shows the general North Anna Containment event tree which can be used for plant damage states 3 through 23. Figures 4.5.1-2 and -3 show the CETs which are used for loss of Containment isolation sequences PDSs 1 and 2. For these sequences the most important question is whether or not the accident progression is terminated in-vessel. Figure 4.5.1-4 shows the CET used for interfacing systems LOCA (event V) sequences (PDS 24). For these sequences the most important question is whether the break location is submerged in the safeguards building. Figure 4.5.1-5 shows the CET for unisolated SGTR sequences (PDS 25). For these sequences it is assumed that there exists a release pathway from the secondary system directly to the environment and that the tube break location is uncovered. Hence, the most important factors which impact radionuclide release have been determined and no events are evaluated on this CET.

4.5.2 Methods for Containment Event Tree Quantification

The purpose of the CET quantification is to assess the relative likelihood or probability of each distinct containment end state

conditional on the plant damage state associated with the CET. This is accomplished by assigning a probability (branch fraction) to each branch in the CET and propagating (combining) the probabilities for each pathway leading to a distinct containment end state.

As discussed previously, the events in the CET may represent phenomenological processes, operator actions or system failures resulting from the severe accident phenomena and conditions. These events are different in character and the quantification process must recognize these differences.

Events associated with physical phenomena generally represent uncertainties regarding the effect the phenomenological event will have on the accident progression. The probability assigned to each branch pathway for these events are the analyst's degree-of-belief, for a given set of accident conditions, that the specific event outcome will occur. These subjective probabilities represent the uncertainty as to which is the physically correct outcome.

Conversely, an event associated with an operator action is similar to the system-based events modeled in the Level 1 event trees. In this case the event branch probabilities can be taken to represent the random or stochastic nature of the event.

To aid in the quantification of a CET event it is often helpful to logically decompose the event into "sub-events" which contribute to the event. For certain events sufficient information may be available (e.g., from past studies) to allow a direct assignment of branch probabilities without further in-depth analyses. This situation may be the case for an event which has been evaluated for a similar plant under similar conditions in a recent PRA or in NUREG-1150.

However, if the CET event cannot be readily quantified by reference to past studies then more in-depth analyses is required. The analysis proceeds by identifying the sub-events (or conditions) which can influence the CET event outcome. For example, if the CET event to be quantified is "Debris Cooled Ex-Vessel" the sub-events which contribute to this event may be identified by asking the following set of questions:

Is there cooling water supplied to the debris?
What is the debris depth?
What was the reactor vessel (RV) failure mode?
What is the vessel pressure at vessel failure?
Where has the debris relocated to?
How much has the debris spread?
What is the debris particle size?

Some of these sub-events may be conditional on the plant damage state characteristics and some may be conditional on prior CET

event branch decisions. A decomposition event tree (DET) or fault tree is often useful for decomposing and evaluating CET branch probabilities.

4.5.2.1 Decomposition Event Trees - General Discussion

Decomposition event trees (DETs) were developed to support the NAPS CET quantification. As mentioned above, a DET is a subordinate tree to the CET and is used to decompose a particular CET event into a more detailed set of events or factors that are useful in quantifying the CET event. DET factors often include dependencies on the occurrence of specific prior events, either in the original plant damage sequence or in the Containment sequence up to that point.

For each CET event heading, there is only one associated DET and all branch points for a CET event will utilize the same DET. Each CET branch point is associated with a unique set of conditions (attributes) that has been determined by the previous CET events or by its plant damage state attributes (for example, at one branch point, the grouping might involve sequences with high RCS pressures and with recirculation sprays operable; while at a second branch point, the grouping might include sequences with low RCS pressures and with recirculation sprays available; etc.) Therefore, even though the same DET is used for all branch points in a CET event, the paths through the DET would be different for each CET branch point. The DET pathway followed by a particular CET branch point will depend on the unique attributes of that branch point. The choice of branch pathway is discussed below.

There are two types of branching allowed for any DET event. The first type, called a sorting event, assigns one branch the value of one and all other branches a value of zero. The sorting event branching is based on a set of rules which determine the branch pathway based on the values for key plant damage state attributes and prior event decisions in the CET. A rule is indicated on the DET diagrams by a left pointing arrow (<--) under each branch in place of a split fraction.

The rules are logic expressions which result in the assignment to a particular event branch when the logic expression is true. Once a rule segment is evaluated as "true", processing stops and no other rule segments are evaluated. An example of a simple rule is presented later in this discussion.

The rule can be evaluated using information from PDS characteristics or from prior CET branching decisions. The rule can be simple or complex, but it must be structured so at least one of the rule segments for the DET heading will be evaluated as true.

The second type of event branching in the DETs are split fractions. For these events a probability is assigned to each of the event branches by the analyst. The probabilities for all the branches in each event should sum to unity.

The sources of "data" for quantification of the split fractions includes:

A. Results of Past Studies

B. Plant Specific Calculations

C. Separate Effects Calculations

D. Engineering Assessment/Judgment

E. System Failure Rate Date

F. Human Error Rate Data

Generally data sources A/B/C/D apply to phenomenological type events and A/D/E/F apply to system/operator action types of events.

The last event in the DET is the same event heading as in the CET. Each possible branch pathway shown in the CET for this event must also exist in the last event of the DET. After the DET is quantified the endpoint probabilities for similar branches in the last event are summed and these summed probabilities are passed back into the CET as the CET branch probabilities.

The basic considerations in the construction of a DET are that: 1) the DET endpoint outcomes match the CET event being decomposed; 2) the selected sub-events can be quantified with available data or analyses, and; 3) all dependencies in the sub-events on plant damage state conditions and prior CET branch point decisions are rigorously treated.

4.5.2.2 Quantification of a DET - An Example

After determination of the event type and deciding whether to decompose the event, the next step is to quantify each DET branch point.

As an example of this process the following discussion describes the quantification for the DET for CET Event "Debris Cooled Ex-vessel." This DET is shown in Figure 4.5.2-6.

The first event on the DET "RCS Pressure at Reactor Vessel Failure" is a sorting type event which assesses whether the vessel pressure at the time of vessel failure was elevated (above 200 psig) or not elevated. If the vessel pressure is elevated then debris

entrainment out of the reactor cavity must be considered. This event is uniquely determined by the plant damage state characteristics and by prior event decisions in the CET. The DET branch pathway to be followed is determined by a set of rules. The rules for this event are summarized below:

If P: RCSPRESS == LO LO;	(If the PDS attribute for "RCS pressure at vessel failure" is LO LO)
Then LO LO;	(Then follow the LO LO branch in the DET)
If C: RCSFAIL == HOT LEG FAILURE	(If for CET event "Induced RCS Failure", the Hot Leg Failure Branch was taken)
Then LO LO;	(Then the RCS is depressurized before vessel failure and follow the LOW branch)
Otherwise NOT LO LO;	(Otherwise the RCS pressure is elevated above 200 psig. So follow the branch for RCS pressure > 200 psig)

The next event asks a phenomenological question - is (a majority of) the debris dispersed out of the reactor cavity? The answer is dependent on the design of the reactor cavity and the pathway out of the cavity. Prior studies indicate that debris dispersal out of the cavity is highly dependent on the vessel pressure at failure and to a lesser extent on the failure mode of the vessel. For low pressure sequences (i.e., the LOW branch for the previous event) little or no debris would be expected to entrain out of the cavity whereas for high pressure sequences some entrainment would be likely (Probability=.9).

The third heading in the DET assesses the depth of the debris. As the debris pool depth increases, the probability of cooling the debris decreases. If the RCS pressure was high at vessel failure and the debris is disbursed out of the cavity then the debris would likely spread over a relatively large area in the lower containment and the debris pool would almost certainly be shallow (< 25 cm deep). For low pressure sequences the debris will remain largely contained within the cylindrical portion of the cavity (deep pool).

The fourth event in the DET "Cooling water to Debris Ex-vessel" is a sorting event which is determined uniquely by plant damage state conditions and is evaluated using a "rule". If cooling water is being supplied to the debris ex-vessel then debris cooling is

possible. If water is not being supplied to the debris then the debris pool is not coolable. The rules for this event are summarized below:

IF C:RS-EARLY == NO FAILURE;	(If the NO FAILURE branch in the CET event "No Early Recirculation Spray Failure" was taken)
THEN YES;	(Then follow the YES branch in the DET because water is available to cool the debris)
IF C:RS-EARLY == FAILURE	(If for CET event "No Early Recirculation Spray Failure" the FAILURE branch was taken)
THEN NO;	(Then follow the NO branch in the DET)

Finally, the fifth event in this DET, "Debris Cooled Ex-vessel" assesses the probability that the debris pool is cooled given the set of prior conditions defined by each pathway through the DET. The probabilities from all the COOLED branches in this event are then summed and passed back to the COOLED branch in the CET event EXVCOOL. Similarly, the summed probabilities from the NOT COOLED branches in the DET are passed back to the CET. The results from the quantification of this DET can now be used in the next CET or DET event.

4.5.2.3 Description of North Anna Decomposition Event Trees

The general Containment event tree (CET) has nine headings which are quantified using decomposition event trees (DETs). DETs are used to avoid having a large number of headings on the CET. This section discusses each North Anna DET in order to provide an indication of how the quantification is accomplished. The details of the quantification are provided in Appendix F. The DETs are presented in Figures 4.5.2-1 through 4.5.2-9. The DETs for the special CETs for loss of isolation sequences (Plant Damage States 1 and 2) are shown in Figures 4.5.2-10 and 4.5.2-11. The DET for the special CET for Event V Containment bypass sequences (Plant Damage State 24) is shown in Figure 4.5.2-12.

DET for Mode of Induced Primary System Failure (Figure 4.5.2-1)

Induced Primary System failure was investigated extensively in the NUREG-1150 study by the In-vessel Expert's Panel. This panel judged that the RCS pressure has a strong influence on the

likelihood of the event. Therefore this is the first heading used in the DET.

The four pressure regimes considered in the North Anna Level 2 analysis are:

Lo Lo Pressure (< 200 psig)
Lo Hi Pressure (200 - 2000 psig)
High Pressure (2000 - 2335 psig)
Hi Hi Pressure (> 2335 psig)

The appropriate pressure regime is determined (by rule) from the plant damage state characteristics.

The NUREG-1150 In-vessel Expert's Panel did not consider temperature induced SGTR or hot-leg failure to be credible events for sequences with pressures below about 2000 psia.

For very high RCS pressures (equal to or greater than the pressurizer PORV setpoint pressure - 2335 psig) the NUREG-1150 In-vessel Expert's Panel estimated that temperature induced SGTR would be highly unlikely if there were no defective tubes in the SGs. Since there are likely to be a number of defective tubes they estimated that temperature induced SGTR would be very unlikely [P(SGTR) = .018]. The expert panel estimated that under these conditions hot leg or surge line failure would be likely [P(Hot Leg Failure) = .72].

For high pressure sequences (RCS pressure less than pressurizer PORV setpoint pressure - 2335 psig and above 2000 psig) the NUREG-1150 In-vessel Expert's Panel estimated that temperature induced hot leg or surge line failure would be unlikely [P(Hot Leg Failure) = .034]. The In-vessel Expert's Panel estimated that temperature induced SGTRs were impossible at pressures below the pressurizer PORV setpoint pressure [P(SGTR) = 0.].

DET for Debris Cooled In-Vessel (Figure 4.5.2-2)

If core damage occurs it may be possible to terminate the accident progression in-vessel and prevent vessel failure. In order to determine if in-vessel cooling is possible a source of in-vessel cooling must be available, (e.g., low pressure injection and recirculation). This question is the first heading in the DET.

For cases where low pressure injection (LPI) is available but the Primary System pressure is elevated above the shutoff head of the LPI System (LPI Deadheaded) (determined by plant damage state characteristic- Status of In-Vessel Injection) initiation of low pressure injection can occur if the RCS pressure is reduced to below the shutoff head of the LPI. Induced hot leg or surge line failure will result in a large break in the RCS which will rapidly

reduce the RCS pressure to below 200 psia allowing for low pressure injection. Rupture of one or two Steam Generator tubes late in time would not be expected to depressurize the RCS to below the LPI shutoff pressure.

For loss of AC power sequences the potential exists for recovery of AC power prior to reactor vessel failure. If power is restored in sufficient time, in-vessel debris cooling and prevention of reactor vessel failure is possible. Since the Level 1 PDS event tree analysis considered power recovery in the time period prior to core uncovery, the recovery period considered here is from the end of the power recovery period considered in the Level 1 analysis up to vessel failure.

For the limited number of core damage sequences where the Low Pressure Injection System is operating and injecting water into the vessel (e.g., large LOCA with failure of accumulators to inject) NUREG/CR-4551 estimated the in-vessel cooling and prevention of vessel failure is very likely [P(Cooled) = .95].

For high pressure core damage sequences with the Low Pressure Injection System available (but dead headed) where an induced hot leg failure occurs during core damage in-vessel cooling was estimated in NUREG/CR-4551 to be likely [P(Cooled) = .9].

For SBO core damage sequences with power recovery prior to vessel failure the probability of in-vessel cooling [P(Cooled) = .7] was estimated, again using NUREG/CR-4551 results.

For all other sequence the probability of in-vessel cooling was estimated to be zero.

DET for No Alpha Mode Containment Failure (Figure 4.5.2-3)

The process where a large mass of debris falls into water in the lower reactor vessel plenum and causes a large steam explosion which fails the upper head of the reactor vessel and generates a missile which fails the Containment is considered based on the fact that it has been included in past PRAs. Experimental evidence indicates that if the RCS pressure is elevated (above several hundred psi) the probability of a large steam explosion is substantially reduced. Therefore, the question of what the RCS pressure is at core damage is the first heading in this DET.

The second heading provides the numerical values for the branch split fractions for low pressure (<200 psig) [P(Alpha CF = .008] and elevated pressure sequences [P(Alpha CF = .0008)]. These values are taken from NUREG/CR-4551.

DET for Mode of Early Containment Failure (Figure 4.5.2-4)

The Mode of Early Containment Failure DET assesses the mechanisms which may cause (or contribute to) Containment over-pressure failure at the time of reactor vessel rupture. The mechanisms which are considered in the DET include 1) the pre-existing Containment pressure before RV failure, 2) the Containment pressure rise due to blowdown of the RCS at vessel failure, 3) the amount of hydrogen produced prior to vessel failure, 4) the fraction of core mass involved in direct Containment heating and 5) the extent of hydrogen combustion at RV failure. The expected pressure for each pathway through the DET is shown under the Summary Event for total Containment peak pressure. The last event in the tree evaluates this pressure against the Containment fragility curve to assess the probability of Containment failure. Furthermore, given that Containment failure has occurred for this indicated pressure the probability of each mode (size) of Containment failure is assessed.

Three pressure ranges (Low, Intermediate and High) were considered for the Containment pressure prior to RV failure. The low pressure regime represents all sequences with successful operation of the Recirculation Sprays and Containment heat removal. The high pressure regime represents large break LOCA sequences that are without Quench or Recirculation Sprays and heat removal and sequences where the RCS is depressurized at the time of vessel failure as a result of an induced Primary System failure (hot leg or surge line failure) that are also without Containment Sprays and Containment heat removal. The intermediate regime is typical of all other sequence types where the RCS is not depressurized prior to vessel failure and where Containment heat removal is not available.

Two regimes (Low, High) were considered for the Containment pressure rise due to blowdown of the reactor vessel at vessel failure. The low pressure rise branch represents all sequences with low RCS pressures (below 200 psig) at vessel failure (including sequences with induced Hot Leg System failure). The high pressure branch represents all other sequences (RCS pressure greater than 200 psig).

Two discretized regimes (greater than, and less than, 40% core inventory of zircaloy oxidized) have been selected to represent the uncertainty in the magnitude of in-vessel hydrogen production. This breakdown is the same as that chosen for this event in the NUREG/CR-4551 Surry analysis.

A number of sequences were analyzed with the MAAP code to assess the extent of in-vessel hydrogen production. For a number of cases the MAAP code was run with the in-vessel core node blockage model both turned on, and turned off. These results indicate that use of the MAAP blockage model will generally result in predicted in-vessel Zr oxidation fractions of less than 40%. Turning off the

MAAP blockage model results in Zr oxidation fractions greater than 40%. The only exceptions to this trend are for large LOCAs where the amount of in-vessel oxidation was predicted to be less than 40% for all cases (with or without the MAAP blockage model on).

There remains substantial disagreement within the technical community regarding the impact of blockage on the magnitude of in-vessel Zr oxidation. This uncertainty was considered in the assignment of the branch split fractions.

The fraction of core debris mass that participates in a DCH event is one of the most important parameters impacting the peak Containment pressure associated with vessel failure. Three discretized levels have been selected to represent the uncertainty in the amount of core debris which fully participates in a DCH event at vessel failure. This breakdown is based on sensitivity studies performed with the CONTAIN code investigating DCH events at the Surry plant (Williams et.al., 1987 NUREG/CR-4896). In this study the peak pressure following reactor vessel failure was calculated for various parameters considered to be important to the calculated DCH pressure. The parameters which were varied in this study included:

a. Extent of hydrogen burning

b. Rate of debris removal from the Containment atmosphere (trapping rate)

c. Debris particle size

d. Amount of In-vessel Zr oxidation

e. Rate of blowdown from the RCS

f. Effect of water

g. Amount of debris participating in the DCH event

h. Debris chemical reaction rate

i. Gas - structure heat transfer rate

The results from this study indicated that peak Containment pressure was not greatly sensitive to debris particle size, chemical reaction rate in the debris particles, extent of in-vessel Zr oxidation, gas - structure heat transfer rates or rate of debris removal from the Containment atmosphere. The peak DCH pressure was found to be sensitive to the fraction of debris participating in the DCH event, the extent of hydrogen combustion associated with the event, the extent and timing of co-entrained water and the blowdown rate from the RCS. The fraction of debris participating in the DCH event has been explicitly included in the DET as has

been the extent of hydrogen combustion (see discussion below). The extent and timing of co-entrained water has not been considered in the tree for the following reasons. Because of the North Anna cavity design one of two distinctly different situations will generally occur for a severe accident sequence. For sequences with the Quench/Recirculation Sprays operating the cavity will fill with water (>300,000 kg of water). For all other sequences the cavity will remain essentially dry prior to reactor vessel failure. The sensitivity study shows minimum values for the peak Containment pressure for a dry cavity and for cases with a large mass of water co-dispersed with the debris. Consequently, we have not explicitly included mass of co-dispersed water as an event in the tree. The rate of blowdown of the Primary System was also shown to have a significant impact on the peak Containment pressure. This parameter was not explicitly included in the tree since a very rapid blow down (approximately 10 seconds or less) was required to significantly increase the predicted peak Containment pressure. This rapid of a blowdown was judged to be very unlikely with the expected mode of vessel failure (i.e., limited area failure of a lower head penetration).

A conservative estimate of the expected fraction of core debris in the vessel likely to be available to participate in a DCH event would be 17% (NRC, 1990). This represented an estimate of the amount of the core debris which would be present as liquid in the lower head of the vessel and available to participate in a DCH event at vessel failure. This analysis also suggested that 30% would be an upper limit on this parameter. The median value of the aggregate distribution for fraction of core inventory released at vessel failure from the NUREG-1150 In-Vessel Experts Panel is .28. The North Anna cavity design does not include an inclined tunnel (such as Zion) for the instrument tubes and hence debris entrainment out of the reactor cavity is likely to be somewhat restricted leading to less than 100% entrainment from the cavity. Given elevated reactor vessel pressure at vessel failure it is assumed in this analysis that a DCH event of some magnitude occurs. Based on the above results it appears likely for high pressure sequences that the fraction of core inventory of debris participating in a DCH event will be in the 0-35% range (nominal value 25%), and that it is unlikely that the value would be in the 35-60% (nominal value 50%) range and very unlikely to be above 60% (nominal value 75%).

The extent of hydrogen combustion at vessel failure is another important parameter impacting the peak Containment pressure associated with vessel failure. Given that a DCH event has occurred it is probable that some hydrogen combustion will occur since the DCH event can act as an ignition source or may even cause catalytic recombination. For DCH events two possible outcomes are considered - a hydrogen burn limited by the local flammability conditions of the Containment (Standard Hydrogen Burn) or hydrogen combustion limited only by the availability of hydrogen or oxygen

in a region without regard to the region flammability conditions (Unconditional Hydrogen Burn - UCHB). These two cases were selected since they were the parametric variations evaluated in the DCH sensitivity study (Williams et.al., 1987). The UCHB is considered to be a very conservative assumption since it allows hydrogen combustion under conditions where the hydrogen is clearly non-flammable. For sequences without a DCH event the two possible outcomes are No Hydrogen Burn or the Standard Hydrogen Burn. For these sequences the important uncertainty is whether an effective ignition source is present at the time of vessel failure.

The Containment pressure summary event is used to summarize the expected Containment pressure for each DET event sequence pathway. The pressure evaluated for each pathway is then used to evaluate the probability of Containment failure and the mode of Containment failure in the next event. Sequences with sprays on are estimated to have a peak Containment pressure 25 psi lower than an equivalent sequence without Quench or Recirculation sprays at vessel failure (Pratt and Bari, 1981, NUREG/CR-2228). Note also that with spray operation the Containment pressure will be at approximately 12 psi prior to vessel failure versus 28 to 37 psi for sequences without spray operation - a difference in base Containment pressure of from 16 to 25 psi even before the transient pressurization at vessel failure.

A study of the North Anna Containment strength shows that this containment is comparable to the Surry Containment. Therefore, the Containment fragility curve developed by the NUREG-1150 experts' panel can be used to evaluate the probability of Containment failure for each DET pathway. The mode of Containment failure was assessed as follows. The NUREG-1150 experts panel judged that 1) a Leak type failure was most likely for failure pressures less than 135 psig (150 psia), 2) a Rupture type failure was most likely for failure pressures between 135 and 150 psig (150 - 165 psia) and 3) a Catastrophic Rupture was most likely for failure pressures greater than 150 psig (165 psia). Note that in the analysis of early Containment failure that we assume that the peak calculated Containment pressure represents the Containment failure pressure. We assume that Containment failures which may occur at pressures lower than the calculated peak pressure do not limit the peak pressure. Furthermore we assume that the Containment failure mode is solely determined by the peak pressure regardless of which pressure Containment failure first occurs at. This implies, for example, that if a Leak type failure were to occur at a lower pressure it could evolve into a Rupture if the Containment pressure rises sufficiently to enter the Rupture pressure regime. The conditional probabilities for each mode of Containment failure were taken from NUREG/CR-4551 and were used to generate the branch probabilities for each mode of Containment failure shown in Figure 4.5.2-4.

The probability of failure by each failure mode for each calculated peak Containment pressure in the DET is evaluated by first assessing the probability of Containment failure from the Containment fragility curve and then multiplying by the conditional probability of failure for each failure mode.

DET for No Early Recirculation Spray Failure (Figure 4.5.2-5)

The headings on this DET include: 1) Recirculation Spray Availability Initially, 2) No Alpha Mode Containment Failure, 3) Mode of Early Containment Failure, 4) Containment Failure Causes Spray Failure, 5) RCS Pressure at Vessel Breach, 6) Excessive Debris in Sump Causes Spray Failure, 7) Containment Fails Into Safeguards Building, 8) Environmental Conditions in Safeguards Building Fails Outside Recirculation Spray (ORS) Pump Motors, 9) Environmental Conditions in Containment Fails Inside Recirculation Spray (IRS) Pump Motors and 10) No Early Recirculation Spray Failure.

The first event assesses whether or not Recirculation Sprays were available (not failed mechanically and AC power available or recovered) prior to reactor vessel failure.

For the second event we assume that steam explosions sufficiently energetic to fail both the reactor vessel and Containment will also fail the Outside and Inside Recirculation Sprays. NUREG/CR-4551 estimated that in-vessel steam explosions which fail the reactor vessel and the Containment (Alpha mode Containment failure) would also fail the sprays.

Events 3 and 4 assess the probability of spray failure resulting from Containment failure.

The Sandia structural engineers who were consulted by the NUREG/CR-4551 authors indicated that the probability of spray failure as a result of Containment failure was "incredible" for all Surry Containment over-pressure failure modes except catastrophic rupture. For the case of catastrophic rupture they indicated that spray failure was unlikely [P(Spray Failure)= .1]. We interpret "incredible" to mean impossible (probability = 0) for No Containment failure, Leak type failures and for Rupture type failures.

Events 5 and 6 evaluate the probability of spray failure as a result of debris entrainment into the sumps. Two failure mechanisms are postulated which could result in Recirculation Spray pump failure from core debris expelled from the cavity; (1) large core debris particles and/or insulation or other loose debris) collecting on the fine mesh screens and blocking flow to the pumps and (2) passage of smaller debris particles through the fine mesh screens and into the pump suction which could damage the pumps.

The fine mesh screens surrounding the sump and individual spray pumps are sized such that particles larger than the smallest Recirculation Spray nozzle opening would not pass through the screens. The total surface area of the outer set of screens is 168 square feet. A uniform debris layer thickness of 1/2 inch would require 1950 kg of core debris which represents about 2% of the core debris expelled from the vessel at vessel failure.

The sump at the North Anna units are directly opposite Containment from the incore instrument tunnel cavity exit in the RH "cubicle" which is the most likely pathway of debris being entrained from the cavity. Hence, debris that is expelled from the cavity will need to be transported a substantial distance across the lower Containment floor to reach the sump. In addition to the instrument tunnel, other debris paths out of the cavity include paths to the operating floor at elevation 291 ft. or through the annulus grating at elevation 241 ft. and over to the containment sumps. In both these cases, the debris would have to migrate a very tortuous and complex path through grating and other containment structures before it could reach the containment sump. Based on the above discussion it appears very unlikely that sufficient amounts of cores debris will be entrained from the cavity and transported across Containment to either block the sump or damage both trains of Inside and Outside Recirculation Spray pumps. Based on the above consideration a probability of .01 was estimated for high pressure sequences (>200 psig) and a probability of zero for low pressure sequences since little debris is expected to be entrained out of the cavity for these latter sequences.

The Safeguards Building houses the Outside Recirculation Spray pumps and the LPI pumps. Containment failure into this relatively small building may result in environmental conditions which could fail the ORS pumps. The Safeguards Building is adjacent to the Containment wall. In event 7, the probability of failure into the Safeguards Building is estimated based on the fractional area of the Containment wall in contact with the Safeguards Building and assuming that the probability of failure is uniform across the Containment cylindrical shell.

Given that the Containment Building has failed into the Safeguards Building the eighth event question assesses whether the environmental conditions (temperature, pressure, humidity, radiation) in the Safeguards Building will cause failure of the Outside Recirculation Spray pump motors. The outside spray pump motors appear to be similar in design to the Inside Recirculation Spray pump motors, however they are qualified for less severe environmental conditions. The table below indicates the environmental conditions for which the Inside and Outside Recirc pump motors are qualified.

	Ambient Temperature (°F)	Humidity	Radiation (rads)
IRS	280 for 30 min 150 for 7 days	100% RH at 150°F for 7 days	1.0E8
ORS	246	100% RH	1.0E7

Based on the available information on the design and environment qualifications for the Outside Recirculation Spray pumps the probability of outside spray failure was estimated as follows:

For sequences with leak type failures, (nominal leak area = 0.1 ft^2) because of the limited leak rate from Containment Building and the physical separation between each Outside Recirculation Spray pump it was judged unlikely [P(Spray Failure = 0.1)] that both trains of the Outside Recirculation Sprays would fail.

For the larger leak sizes ($\geq$ 1.0 ft^2) and leakage rates expected for rupture or catastrophic rupture type Containment failures the probability of failing both trains of the Outside Recirculation Sprays was judged to be indeterminant [P(Spray Failure) = .5] given that the ORS have failed we are now interested in determining whether the inside Recirc Sprays will fail due to environmental conditions in the Containment. The ninth question assesses whether the environmental conditions (temperature, pressure, humidity, radiation or local hydrogen burns) in the Containment early in the accident sequence will cause failure of the Inside Recirculation Spray pumps. The inside spray pumps are qualified for the severe accident environment shown above. Over-pressure failure of Containment implies that the temperature of the Containment atmosphere was elevated to a minimum of 350°F and possibly much higher if the Containment atmosphere was superheated (for example by a DCH event or hydrogen combustion event). These early pressure transients arising from vessel blowdown, DCH, hydrogen burns etc. are likely to be short lived as Containment heat sinks and the spray systems act to cool the Containment atmosphere. Consequently, it is reasonable to assume that the sealed IRS pump motors can withstand transient ambient temperatures well in excess of their peak qualification temperature for a short period of time. For example, these motors are qualified for ambient temperatures of 430°F for 2 minutes (following a Main Steam line break into Containment). Consequently, it is believed that the failure of both IRS pumps will be very unlikely due to environmental conditions for pressure transients which do not result in Containment failure and unlikely [P(Spray Failure = 0.1)] for transients which do result in Containment failure.

Because of the diversity and redundancy in the spray systems, local hydrogen burns are not expected to fail the spray or Containment

heat removal functions. Therefore, hydrogen combustion effects are considered bounded by the environmental conditions discussed above and are not treated explicitly.

The last event has no branching. It is used solely to summarize the outcomes of each pathway in the appropriate form to transmit information back to the CET.

DET for Debris Cooled Ex-Vessel (Figure 4.5.2-6)

The first event on this DET "RCS Pressure at Reactor Vessel Failure" assesses whether the vessel pressure at the time of vessel failure was elevated (above 200 psig). If the vessel pressure is elevated then debris entrainment out of the reactor cavity must be considered. This event outcome is determined by the plant damage state characteristics and by the branch taken under the first CET question (i.e., whether an induced hot leg failure has occurred).

The next event asks a phenomenological question - is (a majority of) the debris dispersed out of the reactor cavity? The answer is dependent on the design of the reactor cavity and the pathway out of the cavity. Prior studies indicate that debris dispersal out of the cavity is dependent on the vessel pressure at failure and on the cavity design. For low pressure sequences little or no debris would be expected to entrain out of the cavity whereas for high pressure sequences substantial entrainment was judged to be likely (Probability = 0.9).

The third heading in the DET assesses the depth of the debris. As the debris pool depth increases the probability of cooling the debris decreases. If the RCS pressure was high at vessel failure and the debris is disbursed out of the cavity then the debris would likely spread over a relatively large area in the lower Containment and the debris pool would almost certainly be either very shallow (<10 cm deep) or shallow (10 to 25 cm deep). In this case it was judged that the debris pool would most likely be very shallow [P(Very Shallow) = 0.9]. If vessel failure occurred at high pressure but the majority of the debris remained in the cavity it was judged that it would be likely that the debris would spread to cover most of the cavity and instrument tunnel floor which results in a shallow pool [P(Shallow = 0.9)]. For low pressure sequences it was indeterminant whether the debris would spread out over the entire cavity and tunnel floor (shallow pool) or if the pool would remain largely contained within the cylindrical portion of the cavity (deep pool) [P(Shallow) = P(Deep) = 0.5].

The fourth event in the DET "Cooling Water to Debris Ex-Vessel" is an event which is determined uniquely by plant damage state conditions. If cooling water is being supplied to the debris ex-vessel then debris cooling is possible. Water will be supplied to the debris if the Recirculation Sprays are operating. If water is

not being supplied to the debris then (unless the debris pool is very shallow) the debris pool is not coolable.

Finally, the fifth event in this DET, "Debris Cooled Ex-Vessel" assess the probability that the debris pool is cooled given the set of prior conditions defined by each pathway through this DET. For sequences with water being supplied to the debris the following conditions are considered. For deep pools (depths greater than 25 cm) it is indeterminant whether the debris pools are coolable given a supply of coolant water [P(Cooled) = P(Not Cooled) = 0.5]. For shallow debris pools (10 to 25 cm deep) it is likely that the debris is coolable [P(Cooled) = 0.9] and for very shallow debris pools (less than 10 cm deep) it is essentially certain that the pool is coolable. For very shallow pools it is likely [P(Cooled) = 0.9] that the debris will be able to transfer sufficient energy by radiation and convection to cool the debris to below the concrete melting temperature without cooling water flow to the debris.

DET for Mode of Late Containment Failure (Figure 4.5.2-7)

A key element of this DET is the availability of the Recirculation Sprays and CHR (and hence AC power). The first two headings deal with power availability/recovery before and after vessel failure. The third heading asks if the Recirculation Sprays are available "early" in the sequence. That is, prior to the time when the Containment would be threatened by gradual steam overpressurization. This requires that the sprays are not mechanically failed and that AC power is (or becomes) available. If the sprays are not operating then Containment failure will occur eventually since decay heat removal from Containment is unavailable. If the sprays and CHR are operating only an energetic reaction such as hydrogen combustion can cause an over-pressure failure of Containment.

In NUREG/CR-4551 it was judged that the only time that a hydrogen burn of sufficient magnitude to challenge the integrity of the Containment might occur would be during rapid deinerting. The North Anna Containment is sufficiently robust that a hydrogen burn at relatively low hydrogen concentrations will not challenge the Containment. Furthermore, it is very unlikely that a large hydrogen concentration could accumulate in a deinerted Containment because of the plethora of ignition sources that would be expected to be available. For example, for the Containment to remain deinerted for long periods of time following core damage the Recirculation Sprays and Containment heat removal must be available or else steam generation would soon cause the Containment to reach an inert condition. The availability of AC power and the operation of electrical equipment inside Containment would almost certainly assure that ignition sources would be available to prevent accumulation of very high hydrogen concentrations. Consequently,

it is judged that the only time that a high hydrogen concentration could occur in conjunction with a deinerted Containment would be late in accident sequences without sprays/Containment heat removal where sprays/CHR are recovered. This situation might occur, for example, for a SBO accident with late power recovery.

A bounding calculation was performed to estimate the probability of late Containment failure from a hydrogen burn. It is assumed that a hydrogen burn occurs when the steam concentration is 50% which consumes all the available oxygen in Containment. This burn would result in a maximum peak pressure of 103 psia. Given this pressure the Containment failure probability would be .03. This is a very conservative calculation. Consumption of all oxygen requires an amount of hydrogen equivalent to greater than 130% reaction of the core inventory of Zirconium. Furthermore, it is assumed that all oxygen is consumed and it is assumed that the burn occurs when the Containment is at its highest pressure where a hydrogen burn could occur (i.e., Containment atmosphere has just deinerted).

If an early hydrogen burn at vessel failure has occurred then insufficient oxygen is available to react with hydrogen to threaten Containment. It is assumed that all sequences with high pressure melt ejection will result in sufficient hydrogen combustion at vessel failure to render the late hydrogen burn threat negligible. Hence, if the RCS pressure is elevated at vessel failure (DET event 4) then it is assumed that a high pressure melt ejection and hydrogen burn have occurred at vessel failure and a late hydrogen burn of sufficient magnitude to fail Containment is not credible.

For sequences with no debris concrete attack (i.e., debris cooled ex-vessel), insufficient additional hydrogen will be produced over that generated in-vessel to establish a potential concentration that could threaten Containment integrity. This is evaluated in the fifth DET event.

For sequences without Recirculation Sprays and Containment heat removal, eventual over-pressure Containment failure was assumed to occur. With sprays and CHR Containment steam, over-pressure failure is prohibited. Non-condensible gas generation alone will not result in over-pressure failure of the North Anna Containment.

The Mode of Late Containment Failure was assessed in a manner similar to that described under the DET for Mode of Early Containment Failure. However, unlike the early Containment failure analysis where Containment loading was due to fairly rapid transient events (e.g., DCH, vessel blowdown, H2 combustion) for late over-pressure failure the loading rate is relatively slow and we assume that any Containment failure mode results in a hole size which is sufficiently large to at least terminate the pressure rise in Containment. The NUREG-1150 Surry Containment over-pressure fragility curve and the NUREG/CR-4551 conditional probabilities for

each failure mode have been used to develop the branch probabilities shown in Figure 4.5.2-7.

DET for No Late Recirculation Spray Failure (Figure 4.5.2-8)

The structure of this DET is similar to the DET for "No Early Recirculation Spray Failure". The first heading in this DET asks if the Recirculation Sprays have already failed or if power is not available. The second heading asks the mode of late Containment failure.

NUREG/CR-4551 indicated that the probability of spray failure as a result of Containment failure was "incredible" for all Containment failure modes except catastrophic rupture. For the case of catastrophic rupture they indicated that spray failure was unlikely (probability = .9). We interpret "incredible" to mean impossible (probability=0) for No Containment failure, Leak type failures, and Rupture type failures. These probabilities were assigned to the branches in the third heading.

The next heading asks if CHR is available. If CHR is not available then failure of the Recirculation Spray pumps due to high temperature and humidity may occur.

The Safeguards Building houses the Outside Recirculation Spray pumps and the LPI pumps. Containment failure into this relatively small building may result in environmental conditions which could fail the ORS pumps. The Safeguards Building is adjacent to the Containment wall. For additional details regarding this event see discussion under DET for "No Early Recirculation Spray Failure".

Given that the Containment has failed into the Safeguards Building the next question assesses whether the environmental conditions (temperature, pressure, humidity, radiation) in the Safeguards Building will cause failure of the Outside Recirculation Spray pump motors. The outside spray pump motors appear to be similar in design to the Inside Recirculation Spray pump motors, however they are qualified for less severe environmental conditions. The environmental conditions for which the Inside and Outside Recirc pump motors are qualified are given in the discussion for the DET for "No Early Recirculation Spray Failure".

If the Outside Recirculation Spray pumps do not fail as a result of the failure of the Containment into the Safeguards Building, they may fail due to the environmental conditions (temperature, pressure, humidity, radiation) in the Containment late in the accident sequence. The only significant failure mechanism is for the pump seals to fail as a result of the temperature of the water passing through the pumps.

The outside spray pumps are qualified for the severe accident environment shown above. At the median Containment failure pressure, the temperature of the Containment atmosphere will be approximately 350°F and possibly much higher if the Containment atmosphere is superheated. Unlike the early pressure transients arising from vessel blowdown, DCH, hydrogen burns, etc. which are likely to be short lived, long term Containment overpressurization will expose equipment in Containment to elevated temperatures for many hours or days. The ORS pump seals may be exposed to temperatures and pressures well in excess of the peak qualification temperature for a long period of time since Containment heat removal is not available.

Since only the pumps are exposed to high temperature fluid, it is considered unlikely [P(Failure) = 0.9] that the outside recirc pumps will fail within the 24 hour mission time being used for this project.

The next question determines whether the Inside Recirculation Sprays will fail due to environmental conditions in the Containment. This question assesses whether the environmental conditions (temperature, pressure, humidity, radiation) in the Containment late in the accident sequence will cause failure of the Inside Recirculation Spray pumps. The inside spray pumps are qualified for the severe accident environment shown under the heading DET for Early Recirculation Spray Failure.

As discussed above, at the median Containment failure pressure, the temperature of the Containment atmosphere will be approximately 350°F and possibly much higher. The IRS pump seals and motors may be exposed to temperatures and pressures well in excess of their peak qualification temperature for a long period of time. It is judged that the failure of both IRS pumps is likely [P(Failure = 0.9] if no Containment heat removal is available since the pump motors will be exposed to elevated Containment temperatures for long periods of time.

DET for Containment Failure Long Term (Figure 4.5.2-9)

This tree is entered only for sequences with operable Recirculation Sprays and Containment heat removal <u>and</u> with the debris **NOT** cooled ex-vessel. CHR operation assures that Containment over-pressure failure from gradual steam generation will not occur. Hence, only basemat melt-through is possible as a long term Containment failure mechanism. In addition, operation of the Recirculation Sprays indicates that water is being supplied to the debris. However, the debris is not in a coolable configuration for these sequences.

The first event on this DET evaluates the vessel pressure at vessel failure. This parameter impacts the extent of debris-entrainment from the cavity and the debris configuration. LO LO sequences are

those with pressures less than 200 psia. The second event evaluates the probability of basemat melt-through.

For noncoolable debris pools with an overlying water layer NUREG/CR-4551 provides the following estimates for the probability of basemat melt-through:

P(BMT) = .25	for core concrete interactions (CCI) involving a large fraction of the core debris
P(BMT) = .05	for intermediate CCI

The assumption is made that "deep pools" used in this analysis (see DET for Debris Cooled Ex-vessel) can be equated with the NUREG/CR-4551 large CCI category and "shallow pools" with the intermediate CCI category.

For low pressure sequences there is a high probability of the core debris remaining in the cavity in relatively deep pools. Evaluating the DET for CET Event Debris Cooled Ex-Vessel for low pressure sequences (LO LO branch) with in-vessel cooling available indicates that uncooled shallow pools have a conditional probability of .05 and uncooled deep pools a conditional probability of .25. Combining these results with the NUREG/CR-4551 probabilities shown above results in a probability of basemat melt-through of 0.2 for low pressure sequences with a cooling water supply to the debris (which are uncoolable).

For high pressure sequences there is a high probability of the core debris entraining out of the cavity and spreading over large areas of the Containment floor resulting in relatively shallow pools. A calculation similar to the one described above results in a probability of basemat melt-through for high pressure sequences of 0.1.

Special CETs Associated With PDS 1 and 2 - Loss of Containment Isolation Sequences

DET for Debris Cooled In-Vessel (Figures 4.5.2-10 and 11)

It is assumed that failure to isolate the Containment will have no impact on whether or not the core debris is coolable in-vessel. Hence the probability that sequences with loss of isolation are coolable in-vessel should be the same as for successfully isolated sequences. The probabilities given below are the frequency weighted average for all isolated non-bypass sequences (PDS 3 through 23).

For all sequences with operable Containment Recirculation Sprays (and CHR) the frequency weighted average probability for debris cooling in-vessel is 0.25.

For sequences without sprays and CHR long term cooling of the debris in-vessel is not considered since eventual failure of the Low Pressure Recirculation System will occur due to increasing sump water temperatures.

Special CET Associated With PDS 44 - Interfacing Systems LOCAs

DET for Event V Safeguards Building FP Retention Effectiveness (Figure 4.5.2-12)

The Safeguards Building houses the LPI and Recirculation Spray pumps. This building is relatively small and the possibility exists for the building to fill with water and cover the break following an interfacing systems LOCA in this building. Section 4.1.3 discusses the building design and the equipment layout. Analyses performed for NUREG/CR-4551 resulted in the assignment of a probability of 0.85 for the break location being submerged. Since the Safeguards Building at Surry and North Anna both have similar layouts and piping configurations, the probability of a break being submerged is also taken to be 0.85 at North Anna.

4.5.3 Results of CET Quantification

The six plant damage states with the highest frequency are PDS 21, 4, 20, 25, 14 and 12. These six PDS categories account for over 78% of the total core damage frequency. The quantified CETs associated with each of these PDS are shown in Figures 4.5.3-1 through 4.5.3-6. Appendix F, Section F.4, contains the quantified CETs for all plant damage states.

CET Results for Dominant Plant Damage States

PDS 21

PDS 21 contains small and intermediate LOCAs with power available and with operable Recirculation Sprays and Containment heat removal. The quantified CET for PDS 21 is shown in Figure 4.5.3-1. Because of the LOCA the estimated RCS pressure is in the intermediate range (200-2000 psig). Pressures in this regime are not expected to result in induced RCS hot leg failure (and depressurization below 200 psig) or in induced SGTR.

The probability of in-vessel cooling is considered to be negligible since the dominant sequences contributing to this PDS involve failure of high and low head injection (S2D1D3), failure of high head ECCS and failure to depressurize (S1OH2, S2D1Y, S2D1Y) or failure of low head recirculation (S2H1). These sequences are considered unrecoverable in the time available between core damage and vessel breach. Consequently in-vessel injection is assumed never available for sequence in this PDS.

Since the RCS pressure always remains above 200 psig for sequences in this PDS the probability of an alpha mode Containment failure (CF) is small. Also, since the pressure does remain elevated the possibility of an early over-pressure failure at vessel breach is non-zero (due to DCH and associated mechanisms). However, the probability is very small since the Recirculation Sprays and Containment heat removal (CHR) are operating. With sprays and CHR the Containment pressure prior to vessel breach will be subatmospheric and the peak pressure from DCH and/or hydrogen burns will be limited. There is a very small probability of spray (and CHR) failure subsequent to vessel rupture due to debris blocking the sump screens or failing the sprays pumps (or as a result of Containment failure for sequence pathways with early Containment failure).

For pathways where the sprays remain functional following vessel failure there is a high probability that the debris will be coolable ex-vessel since substantial quantities of debris would be expected to be entrained out of the cavity (or spread over the entire cavity floor) because of the elevated RCS pressures. With the sprays and CHR functional throughout the accident duration late over-pressure failure is precluded and either no Containment failure or basemat melt-through (for debris not cooled ex-vessel) will result. The CET endpoint probabilities for the PDS 21 CET are shown in Figure 4.5.3-1 along with their source term category assignments.

PDS 4

PDS 4 contains station blackout type sequences with recovery of power (sprays and CHR) subsequent to reactor vessel failure but prior to the anticipated time of Containment over-pressure failure. The quantified CET for this PDS is shown in Figure 4.5.3-2.

Because of the very high RCS pressures for this PDS there is a high probability (.72) of induced RCS failure and a small, but still significant, probability of induced SGTR (.018). Since AC power is not available early, there is no possibility of cooling the debris in-vessel. For sequence pathways with induced hot leg failure, the RCS pressure is assumed to fall below 200 psig. This increases the probability of an in-vessel steam explosion failing the vessel and Containment (alpha mode failure). However, with the RCS

depressurized the probability of early Containment failure by overpressurization (DCH, hydrogen burns, etc.) at vessel failure is reduced to insignificant levels.

Since for PDS 4 power may not be recovered until many hours after vessel failure the CET indicates the sprays are failed early. In addition, it is assumed that since spray operation is delayed until many hours after vessel failure that the debris is not cooled ex-vessel. The major effect of late recovery of AC power, sprays and CHR is to greatly reduce the probability of late over-pressure Containment failure. There also exists a small probability that a hydrogen burn capable of failing Containment will occur during deinerting of the Containment following spray initiation. Cooling of the debris ex-vessel is assumed to not be possible, consequently eventual basemat melt-through may occur. Figure 4.5.3-2 shows the sequence endpoint probabilities for this CET and the associated source term categories.

PDS 20

PDS 20 contains small and intermediate LOCA initiated sequences. Figure 4.5.3-3 shows the quantified CET for this PDS. Sprays and Containment heat removal are available. Sequences in PDS 20 have Low Pressure Injection available, but injection is dead headed because the RCS pressure for these sequences is above the shutoff head of 200 psig. However, since the RCS pressure is below 2000 psig, the probability of induced primary system failure is negligible. Hence, in these sequences, RCS pressure will never by low enough for LPI. Therefore, the CET pathways and results for this PDS are similar to that for PDS 21.

PDS 25

PDS 25 contains Containment bypass SGTR core damage sequences. Since Containment mitigation features and Containment accident phenomena are largely irrelevant for these sequences a detailed CET is not required. The CET for this plant damage state shown on Figure 4.5.3-4 simply transfers the PDS 25 SGTR sequences to the unique source term category (24) for SGTR sequences.

PDS 14

PDS 14 consists of sequences where power is lost either because of a loss of offsite power or because of a loss of Emergency Switchgear Room cooling. The RCS boundary does not remain intact because of the failure of operator cooldown or because of the failure of seal cooling resulting in a seal LOCA. Power is recovered prior to vessel failure; and, therefore, high and/or low pressure injection is available to cool the debris in-vessel, thus

preventing vessel rupture. For pathways where the debris is not cooled in-vessel, the vessel will fail and debris will be dispersed out of the cavity because of the relatively high RCS pressures. Because of the debris spread, the debris will be coolable. Recirculation sprays and Containment heat removal will be available before vessel rupture (after power recovery).

Figure 4.5.3-5 shows the CET endpoint probabilities and source term category assignment for PDS 14.

PDS 12

PDS 12 consists of large break LOCA and Reactor Vessel rupture sequences where Recirculation Sprays and Containment heat removal are available. In addition, in-vessel injection is available and although this flow is insufficient to prevent core damage, it is sufficient to remove decay heat and therefore reduce the likelihood of vessel failure.

As a result of the initiator, the RCS pressure is assumed to be below 200 psig. Since Recirculation Sprays and Containment heat removal are operating, the probability of overpressure failure of the Containment is negligible except for the alpha mode failure.

Since high and/or low pressure injection is available, it is likely (95% probability) that the core debris will be cooled in-vessel and, as a result, the vessel will remain intact. For pathways where the debris is not cooled in-vessel, the vessel will fail and the debris will pour out into the cavity at relatively low pressures. Entrainment out of the cavity will be minimal, and the debris in the cavity will either be shallow (10 to 25 cm) or deep (greater than 25 cm) depending on the degree of spreading. Even in the presence of water in the cavity for these sequences, there is a probability (30%) that this debris is not coolable because of the debris depth. Since there is no overpressure containment failure for these sequences, pathways where there is no ex-vessel debris cooling can lead to a basemat melt-through as a long term Containment failure mechanism.

The CET endpoint probabilities and source term category assignment for this PDS are shown in Figure 4.5.3-6.

4.6 ACCIDENT PROGRESSION ANALYSIS

4.6.1 Summary of Sequences Analyzed

This section contains a description of the deterministic Containment accident progression analyses. All aspects of the

probabilistic Containment event analysis, including a discussion of CET quantification, are contained in Section 4.5 above.

To support the development and quantification of the Containment event tree an assessment of the physical progression of a spectrum of accident sequences was performed. This effort provided critical information and insights into:

- timing of key events
- Containment loads
  - pressure
  - temperature
  - pressure rise rates
- debris relocation and cooling
- mitigation effectiveness of ESFs
- generation, and combustion of hydrogen

Plant-specific analysis of accident progression with the deterministic code MAAP (Section 4.2) were closely coupled to development of the CET. Accident progression analyses were also utilized in the quantification of the CETs. Results from prior studies, sensitivity studies with deterministic models (MAAP), and separate effects analysis and judgment are all used in assessing the relative probabilities of the various possible accident progression pathways modeled in the CET discussed above in Section 4.5.

A limited set of North Anna plant specific MAAP calculations were made to supplement the Surry IPE MAAP calculations and both sets of MAAP results were used for the North Anna IPE. It was judged that the Surry calculations would be applicable for the North Anna IPE, since the two plants are very similar. In addition, the worst case Surry MAAP calculations were verified by using the North Anna MAAP model. It is to be noted therefore that the Tables and Figures relating to MAAP calculations for timing of certain key events and peak values of plant parameters including source term release fractions were mostly calculated using the Surry model.

A number (23) of MAAP calculations were performed to determine the range and variation in Containment response to be expected for a variety of accident scenarios. A brief description of each of these accident progression cases is given in Table 4.6.1-1. Table 4.6.1-2 presents a summary of the timing of key events, and Table 4.6.1-3 presents selected calculated parameters for these cases. Subsets of the accident progression studies were utilized for various specific aspects of the CET development and quantification and these are discussed in Section 4.6.2. The other calculations were performed primarily to gain general insights and are not

discussed further here beyond their inclusion in the tables. (It should be noted that these runs were generally terminated when the item of interest had been determined. Also, in some cases, arbitrary modeling assumptions were imposed to achieve specific sequence circumstances. These factors should be kept in mind when interpreting the results.)

MAAP calculations were also performed to derive release fractions for various source term categories (cf. Section 4.7). These source term cases were run to well past Containment failure and as such represent the entire accident progression from initiating event to release completion. The (eleven) source term MAAP cases are summarized in Tables 4.7.3-1, -2, and -3. Three of these cases were selected, based upon their expected importance to the source term and/or the contribution of the initiating event to core damage, for detailed discussion of the accident progression. The cases selected are an SBO with late Containment failure, an interfacing systems LOCA, and a Steam Generator tube rupture with stuck open secondary side relief valve. The discussion is given below in Section 4.6.3.

4.6.2 Accident Progression Analysis Results

As noted previously (above and Section 4.5.1), MAAP runs were performed to gain further understanding of the accident processes. Four specific topics are discussed below, based on selected sets of the cases of Table 4.6.1-1.

RCS Pressure at Vessel Rupture

The Reactor Coolant System (RCS) pressure during core damage and at the time of vessel failure can have a major impact on several potentially important Containment events. High RCS pressures during core heatup and core damage facilitate natural circulation heat transfer to the hot leg which increases the potential for temperature-induced hot leg, surge line or Steam Generator tube rupture failure. Elevated pressures at the time of vessel rupture may result in entrainment of the core debris out of the reactor cavity and increase the potential for debris fragmentation, dispersal, and direct Containment heating.

Large LOCAs (29 inch diameter breaks) result in rapid depressurization of the RCS to below 200 psia. Accumulator contents are released immediately after the break occurs. Recovery of in-vessel injection after the onset of core damage will not result in the sustained production of steam sufficient to maintain any significant pressure at vessel failure because of the existence of the large break area.

Medium and small LOCA events may, however, have elevated RCS pressures at various times because of the reduced leakage area. Figures 4.6.2-1 to 4.6.2-3 are the calculated RCS pressure traces prior to vessel failure for a 6 inch (case 1), 4 inch (case 2), and 2 inch (case 3), equivalent diameter LOCA, respectively. In all cases there was no in-vessel injection. In each case there is a significant RCS pressure rise occurring after the core support plate fails and hot molten core debris falls into the residual water in the lower head plenum. The values calculated are shown below.

Break Diameter	Peak Pressure After Core Slump Into Lower Plenum
29" (x2)	~ 0 psig
6"	83 psig (.51 MPa)
4"	265 psig (1.93 MPa)
2"	1250 psig (8.6 MPa)

There exists some uncertainty regarding the exact mechanics and dynamics of the phenomena occurring during transfer of molten material into the pool of water in the lower head. Allowing for this, the above results show that there is a reasonable probability, even at the 6" break sizes, that the RCS pressure will be elevated above 200 psig at or before vessel failure. As the pressure traces illustrate, there is similarly some probability that the RCS pressure might be below 200 psig. Very small breaks (less than 2") may attain pressure above 2000 psig threshold. This is shown in Figure 4.6.2-4, which is the calculated RCS pressure for a SBO event, where the pressure peaks above 2350 psig (1.6 MPa). Therefore for transients type events, which include very small LOCAs, the RCS pressure will depend also on whether the pressurizer PORV's are opened.

Power Recovery

This parameter is used to identify station blackout sequences with recovery of offsite AC power, subsequent to core damage but within a time period judged to be prior to either vessel failure and/or Containment failure. Note that recovery of the diesel generators is not considered in the PDS event trees and that power recovery is defined solely as offsite power recovery. Three possible branch pathways are evaluated; prior to RV failure, prior to Containment failure and no power recovery. Power recovery subsequent to core damage allows for either the possible restoration of in-vessel injection which may terminate the accident and prevent vessel failure, or later restoration of Quench or Recirculation sprays and Containment heat removal in sufficient time to prevent Containment failure and mitigate the releases. For events initiated by loss of switchgear room cooling, the recovery of room cooling is treated as analogous to the recovery of power.

In this type of severe accident, there will be many hours between core damage and the time when the Containment integrity is first threatened from long term steam/non-condensible gas pressurization. The Surry Containment fragility curve developed for NUREG-1150 has a median Containment failure pressure of 128 psig (143 psia). However, to allow for uncertainties, the 5% failure pressure (93 psig), was utilized to assess the time available for power recovery prior to Containment failure.

MAAP calculations were performed for a matrix of station blackout sequences that considered operation/non-operation of the steam Turbine-Driven Auxiliary Feedwater System and the occurrence/non-occurrence of a large seal-LOCA.

Figures 4.6.2-1 to 4.6.2-13 show the calculated Containment pressures. The pertinent results are summarized below.

Case	Steam-Driven AFW Turbine Operate?	Seal LOCA Occurs?	Time Period From Initial Core Damage to Vessel Failure (hours)	Time Period From Core Damage to 5 Percentile CF Pressure (hours)
7	Yes	No	1.1	28
8	Yes	Yes	1.5	38
9	No	Yes	1.8	21
10	No	No	2.4	17

The fourth column of the table is the time period from 30 minutes before core damage to 30 minutes before vessel failure (30 minutes being the allowance for the time to refill the Service Water canal). The fifth column is the time interval calculated between the time of initial core damage (less 30 minutes) and the time the Containment pressure reaches the fifth percentile failure level (less 30 minutes).

Representative time periods were selected from the above results for use in the plant damage state selection rules (see Section 4.3).

Containment Pressure Just Prior to Reactor Vessel Failure

Early over-pressure Containment failure can potentially result from the energetic processes which may occur at reactor vessel failure (pressure rise due to RCS blowdown, DCH, hydrogen burns). The pre-existing Containment pressure just prior to vessel failure is also an important factor influencing the peak transient pressure following vessel failure.

The following cases have been analyzed with the MAAP code to determine the Containment pressure just prior to vessel failure. NUREG/CR-4551 estimates are also presented.

Sequence Description	MAAP Case No.	Containment Pressure MAAP (psia) SPS / NAPS	NUREG/CR-4551 Estimate (psia)
Large Break LOCA No SI, No Sprays No Cont Heat Rem.	14	30.4 / 39	37
Short-Term SBO	9	28.3 / -	26
Short-Term SBO	10	29.0 / 25	26
2" LOCA with Sprays and Cont. Heat Removal	3	11.5 / -	16
2" LOCA, No SI, No Sprays, No Cont. Heat Removal	16	41.0 / 23	
3" LOCA, No SI, Normal Sprays	34 (level 1)	- / 12	
3" LOCA, 1 of 2 SI, 1 of 2 IRS and 1 of 2 ORS	6 (level 1)	- / 14	

The calculated Containment pressures traces are shown in Figures 4.6.2-7 through 4.6.2-12 for cases 9, 10, 1, 3, 14 and 16, respectively.

Based on these results, three pressure ranges were selected to describe the expected pressure regimes for event tree quantification purposes (see Section 4.5.2). The three pressure ranges cover the expected pressure regimes for the spectrum of Surry accident sequences. The low pressure regime represents all sequences with successful operation of the Recirculation Sprays and Containment heat removal. The high pressure regime represents large break LOCA sequences and sequences where the RCS is depressurized at the time of vessel failure by an induced Primary System failure and that are without Quench or Recirculation sprays and Containment heat removal. The intermediate regime is typical of all other sequence types where the RCS is not depressurized prior to vessel failure and where Containment heat removal is not available.

Containment Pressure Rise due to RCS Blowdown at RV Failure

The following cases have been analyzed with the MAAP code to determine the Containment pressure rise at vessel failure due to blowdown of the RCS.

		MAAP Calculated Values	
Case Description	Case Number	Containment Pressure Rise (psi) SPS / NAPS	RCS Pressure At Vessel Failure (psia) SPS / NAPS
Large Break LOCA No Cont. Sprays No Cont Heat Rem.	14	0 / 0	<< 200/<< 200
6" LOCA No Cont. Sprays No Cont Heat Rem.	15	1.7 / 3	< 200/< 200
Short-Term SBO	9	13.4 / -	2494/-
Short-Term SBO	10	14.2 / 20	2422/2400
Long-Term SBO	8	21.8 / 21	580/580
2" LOCA No Cont. Sprays No Cont. Heat Rem.	16	20.0 / 13	766/760

Containment pressures for these cases are shown in Figures 4.6.2-6 to 8 and 4.6.2-11 to 13.

It is seen that sequences with low RCS pressure (< 200 psia) at RV failure have a low pressure rise (2 psi) and all other sequences have a high pressure rise (≥ 8 psi) due to blowdown of the RCS.

Amount of Hydrogen Produced In-Vessel

A number of cases were analyzed with the MAAP code to assess the extent of in-vessel hydrogen production. For a number of cases the code was run with the MAAP in-vessel core node blockage model either turned on and turned off. The results of these runs are summarized below:

Case Description	Case Numbers	MAAP Calculated Fraction of Core Inventory Zr Oxidized (Blockage On/Off)
Large Break LOCA	20 / 14	21 / 31 %
6" LOCA	- / 15	- / 28 %
2" LOCA	21 / 16	32 / 55 %
Short-Term SBO	13 / 09	32 / 63 %
Short-Term SBO	10 / -	- / 55 %
Long-Term SBO	08 / -	- / 49 %

The above results indicate that use of the MAAP blockage model will generally result in predicted in-vessel Zr oxidation fractions of less than 40%. Turning off the MAAP blockage model results in Zr oxidation fractions greater than 40%. The only exceptions to this trend is for large and intermediate LOCAs where the amount of in-vessel oxidation was predicted to be less than 40% for all cases (with or without the MAAP blockage model on).

There remains substantial disagreement within the technical community regarding the impact of blockage on the magnitude of in-vessel Zr oxidation. It is interesting to note that for the seven cases considered by the NUREG-1150 expert panel, the mean values of the aggregate distribution for fraction of Zr oxidized in-vessel ranged from .32 to .52 which generally corresponds with the MAAP calculated values above except for the large break LOCA with blockage case.

4.6.3 Selected Accident Sequence Progressions

These sequences were analyzed for the purpose of determining representative sets of radionuclide release fractions for specific source term categories (those results are presented in Section 4.7.3 below). A discussion of the Containment processes is presented here to provide insight into the phenomena and mechanisms that are occurring during the progress of the accident. The plant modeling utilized for MAAP has been presented in Section 4.2 and the plant damage state and Containment event modeling have been discussed in Sections 4.4 and 4.5 above.

Long Term SBO with Late Containment Over-pressure Rupture

This is MAAP Case 28 run for source term category (STC) 15. It is a station blackout without power recovery. The only source of water to the core are the three accumulators. One turbine-driven AFW pump operates. The AFW flow is throttled to maintain Steam Generator level until the batteries deplete and then is left constant at the last average throttled flow rate. There is no Safety Injection, Quench Spray, or Recirculation Spray flow, and hence no long-term Containment heat removal. The MAAP-produced summary of event occurrences is Table 4.6.3-1. The run was terminated at 100 hours, well after Containment failure.

The water level in the Steam Generators is shown in Figure 4.6.3-1. Prior to 14400s (4 hr) the AFW flow has been throttled to just match the boiloff due to decay heat removal from the RCS. After that, the level rises because the (constant) AFW flow is more than the (decreasing) decay heat boiloff rate. At 46730s (13.2 hr) the condensate storage tank (CST) water is depleted and the Steam Generators then boil dry at 71630s (19.9 hr). This eliminates heat removal from the RCS and Containment which then results in core melt, vessel rupture, and a late large Containment rupture due to over-pressure.

The pressure in the Primary System (Figure 4.6.3-2) gradually falls to 7.3 MPa (1060 psia) to match the secondary side relief value set point by 37900s (10.5 hr). Following loss of heat removal, the pressure climbs to 17.2 MPa (2470 psia), limited by steam flow out of the primary relief valves. The RCS water temperature (Figure 4.6.3-3 rises from 568K (562°F) to about 620K (656°F) at this time. The water level in the RCS begins to drop when flow occurs out of the relief value and the cover uncovers at 77470s (21.5 hr) (Figure 4.6.3-4) and then melts. The core support plate is failed by the molten debris at 89770s (24.9 hr) and the vessel lower head fails at 90270s (25 hr).

The molten core debris is ejected out the vessel failure breach into the Cavity/Incore Instrument Tunnel volume located below the vessel. At the time of vessel breach, the RCS is at an elevated pressure (at least at the pressurizer relief value setting). The quantity of escaping steam that follows the debris ejection is calculated to be at sufficient velocity to sweep much of the debris out into the other lower Containment regions (below the refueling deck and inside the crane wall - Volume B in the MAAP model). This displaced debris is assumed to quickly solidify due to its dispersed form. As the residual in-vessel core material subsequently melts, it is assumed to drop into the cavity by gravity and remain there.

The total amount of hydrogen produced in-vessel (Figure 4.6.3-5) is about 470 kg (1034 lb). It does not ignite because of steam inerting.

The Containment pressure (Figure 4.6.3-6) first rises after the Quench Tank disk ruptures at 73150s (20.3 hr), and then rises again rapidly from .18 MPa (26 psia) to .32 Pa (46.4 psia) at vessel rupture. Due to continued steam production from decay heat, the pressure then climbs steadily to .986 MPa (143 psia) at 238400s (66.2 hr) when a 7 ft^2 rupture occurs in the Containment which rapidly results in Containment depressurization. Enough debris is deposited in the cavity for it to dry out at 271860s (75.5 hr).

Most (20.1 kg) of the iodine in the core (CsI: see Table 4.7.3-5 for a description of the released radionuclide species) is released to Containment from the fuel during the initial in-vessel melt process, with 2.7 kg CsI released after vessel rupture (Figure 4.6.3-7). A further 2 kg is released from the debris ex-vessel constituting essentially complete release of the initial inventory. The CsOH (cesium) release follows essentially the same pattern (Figure 4.6.3-8). The La_2O_3 releases occur primarily as hot corium drops into the cavity, when a total of 0.08 kg of the initial inventory of 410 kg is emitted (Figure 4.6.3-9) SrO (strontium) releases are shown in Figure 4.6.3-10 and are intermediate between the volatile and non-volatile species releases.

Interfacing LOCA (V Sequence)

This is MAAP Case 33 run for STC 23. It is a Containment bypass sequence. Consistent with this source term category definition, the low pressure piping break location is assumed as one that is located above the flooded Safeguards Building volume at core melt. A 2.57 in diameter break is assumed from the RCS to the environment outside the Containment. The 2.57 in diameter figure corresponds to the inside diameter of the venturi in the LHSI line. The core melts because of the loss of Primary System and injection water out through the break. No fission product holdup or retention capability is assigned to the Safeguards Buildings as the break occurs at an elevation above the flooding line. The MAAP summary of events is given in Table 4.6.3-2.

When the break occurs, the charging pumps and the low head pumps come on. The charging pumps maintain volume in the RCS while the LPI pumps discharge directly through the break into the Safeguards Building. The Safeguards Building is open to the basement of the Auxiliary Building at the 16 ft elevation. Water exiting the Safeguards Building will eventually enter the Auxiliary Building where the Charging Pumps are located. Depending on the circumstances of the break, one or both sets of the pumps may flood out, but the general result is the same - core melt occurs because of failure to successfully achieve the recirculation mode. For this run the Charging Pumps were assumed to continue to operate. The result is that the RWST is depleted at 14620s (4.1 hr). The water in the vessel then begins to drop (Figure 4.6.3-11) with core uncovery at 17930s (5 hr). The RCS pressure and water temperature

both drop continuously from the break initiation time until vessel rupture except for a temporary increase at the time of loss of makeup (Figures 4.6.3-12 and -13). The main coolant pumps trip off at 15570 (4.3 hr) because of voiding in the Primary System. The fuel reaches 2200°F at about 21000s (5.8 hr) (Figure 4.6.3-14). When the RCS pressure has dropped sufficiently, accumulator injection occurs with their contents being depleted by 51308s (14.2 hr). The RCS pressure then falls more rapidly until the core support plate fails at 55180s (15.3 hr). The drop of the molten core debris into the lower plenum causes the RCS pressure to spike at 6.2 MPa (900 psia). The vessel is breached by the molten debris at 55680s (15.5 hr) and the remaining RCS water is vented to the Containment. The corium is distributed between the cavity and the lower compartments. The processes at vessel breach are similar to those for the SBO case discussed previously.

There is little water in the Containment as the RCS, RWST, and accumulator contents escaped through the break. The remaining water and steam are sufficient such that the Containment pressure rises from subatmospheric to atmospheric (Figure 4.6.3-15), and little further outflow from the break occurs.

The amount of hydrogen produced in-vessel prior to rupture is about 350 kg (760 lb). The corium slowly reheats ex-vessel, and at run termination, 86000s (24 hr), the corium temperature in the lower compartment is 1560K (2350°F) and concrete attack has begun. The material is spread thinly and the rates are quite low. At some later time, a release of low-volatile materials will start again, but the calculations were not run out that far.

Steam Generator Tube Rupture (SGTR)

This is MAAP Case 40 for STC 24. It is a two-tube SGTR with a stuck-open secondary relief valve. The core melts because of the loss of RCS inventory and Safety Injection flow out through the break. The MAAP summary of events is Table 4.6.3-3. The run was terminated at 172000s (48 hr).

The reactor trips, the MSIVs close, and the AFW comes on 190s (3.1 min) after the break of 6.6E-3 ft^2 (two 0.775 in diameter tubes). Level is maintained in the unbroken SGs while the broken SG floods due to the inflow from the RCS (Figure 4.6.3-16).

The AFW flow to the broken SG is turned off at 900s (15 min). The secondary side relief valve is opened at 1800s (30 min) with at least one relief valve associated with the broken SG assumed to stick open.

The Charging Pumps come on at 200s (0.05 hr) and deplete the RWST at 24180s (6.7 hr) discharging the water into the RCS, out the break, and then out the secondary relief valve. The Primary System

pressure (Figure 4.6.3-17) has decreased to about 6.2 MPa (900 psia) at this time. The pressure continues to drop so that the accumulators inject and are depleted at 27050s (7.5 hr). The RCS pressure then holds more or less constant at about 1.5 MPa (210 psia). The remaining RCS water temperature reheats to about 470K (382°F) (Figure 4.6.3-18).

The water level in the broken SG begins to fall at about 42300s (11.8 hr) and it dries out at 65930s (18.3 hr). The core uncovers soon thereafter at 67120s (18.6 hr) (Figure 4.6.3-19). The core temperature reaches 2200°F at about 70000s (19.4 hr), melting ensues, and the core support plate fails at 78220s (21.9 hr) resulting in vessel rupture at 78320s (22 hr). From this point on, the accident progresses more or less as in the V sequence with core-concrete attack beginning at about 95000s (26.4 hr). It is worthy of note that the broken SG gas temperature (Figure 4.6.3-20) rises to about 880K (1120°F) as the core melts, it having boiled dry previously.

The CsI, CsOH, SrO, and La_2O_3 radionuclide releases to Containment are shown in Figures 4.6.3-21 to -24 respectively. Most of the volatile release occurs before vessel rupture.

4.7 SOURCE TERM CHARACTERIZATION

The end points of the Containment Event Trees (CET) represent the outcomes of possible in-Containment accident progression sequences. These endpoints represent complete severe accident sequences from initiating event to release of radionuclides to the environment. The Level 1 system information is passed through to the Containment evaluation in discrete plant damage states. An atmospheric source term may be associated with each of these Containment sequences. Because of the large number of CET sequences and because of similarities in the sequence characteristics, however, it is neither necessary nor practical to develop a source term estimate for each Containment sequence. Sequences with similar characteristics are therefore grouped into source term (release) categories (STCs) to reduce the required source term assessment effort.

4.7.1 Source Term Category (STC) Grouping Parameters

Source Term Category Definition

The source term categories(STCs) are groupings of the (extended) containment accident sequences. These sequences combine the Level 1 accident sequences, as grouped into plant damage states (see Section 4.3 above), and the post-core damage accident progression portrayed by the containment event trees (CETs). The goal of this

grouping process is to reduce the number of required source term release analyses to a tractable number while continuing to distinguish the more important differences among the sequences which are likely to influence the source terms. The source term category characteristics are defined by selecting a set of accident progression and plant damage state parameters which are considered to be important to: accident progression in the Containment; the time, mode and location of Containment failure; and the radionuclide source term. The parameters that are used to define the plant damage states include the functional status of important systems, state variables which are determined by systems operation (e.g., Reactor Coolant System pressure), accident initiator type and timing of key events (e.g., power recovery).

The first step in the source term assessment effort is to identify the sequence characteristics which are most important to definition of the source term. These characteristics are identifiable from the Plant Damage State (PDS) characteristics and from the Containment Event Tree CET sequence characteristics since one of the primary objectives in the PDS grouping and CET evaluation has been to define those events and conditions most important to source term assessment. This selected set of sequence characteristics important to source term assessment are used as grouping criteria to define the release categories and the associated source term magnitude, composition and timing.

Eight criteria were selected for use in defining the NAPS source term categories. A description of these criteria and the bases for their selection are discussed in detail below. Generally, these criteria were selected because they have a controlling influence on determining key accident progression characteristics; the time, mode and location of containment failure and the radionuclide source term to the environment.

The Source Term Category Grouping diagram is a tool used to perform the classification of the CET end points by combining the various grouping parameters into unique plant damage states. As for the Plant Damage State diagram, arranging the STC Logic diagram in such a way that the most important parameters are considered before parameters of lesser importance, and eliminating decision points by allowing only one decision branch results in the collapse of the number of source term categories to a reasonable number while still preserving the most important differences among the various sequences. The reasons for suppressing branching on a decision branch are somewhat judgmental and involve the following considerations: (1) Is this branch necessary to distinguish an important difference among the sequences? (2) Is the frequency of sequences following this pathway likely to be sufficiently large to warrant additional plant damage states?, and (3) Can a conservative choice be made which allows for branch suppression which is not likely to significantly impact the overall results.

The Source Term Logic Grouping Diagram for NAPS is shown in Figure 4.7.2-1. The basis for the frequencies shown is described in the next Section. The associated rules for assigning the Level 1 sequences to Plant Damage States are given in Table 4.7.2-1. These rules are shown in NUCAP+ (Fulford, 1991) format and the diagram is drawn with the assistance of NUCAP+. The basis for the assignment rules for each criterion is given in detail below. The endpoints of the logic diagram represent individual source term categories and the pathway through the diagram (i.e., the set of decision paths taken at each decision branch define the attributes for each source term category). Twenty four (24) individual source term categories were defined for NAPS.

The rationale for selection of each of the STC grouping criteria (or parameters) and the possible attributes is discussed below. In scanning the rules specified in Table 4.7.2-1, the following brief list of NUCAP+ rule syntax and semantics may be useful:

"==" denotes an equality test. If the item in front is equal to the item after, then the result is TRUE otherwise it is FALSE.

"!=" denotes an inequality test and returns the opposite results from the equality test.

":" The first item in the test for STC classifications is denoted as "T:ccccc" where 'ccccc' is a top event or criterion in either the PDS diagram (T = P), the CET (T = C), or the STC diagram (T = S). The attribute of the sequence for this criterion is then tested against the attribute given as the second item in the test.

"*" denotes a Boolean AND conjunction of the tests before and after it. Both tests have to return TRUE for the ANDed tests to be TRUE.

"IF .." A statement line beginning with 'IF' is evaluated to see if the tests on the line '..' are TRUE. A IF statement line may have more than one AND in it. Subsequent statement rule lines without an intervening THEN line are de facto OR statements.

"THEN aaa" When a complete statement (line) is evaluated as TRUE then the first subsequent THEN gives the assigned attribute 'aaa' for this RULE. A DEFAULT statement is always TRUE and the attribute is given on the same line. After an attribute is assigned no further processing of the rule is performed, so that the ordering of rule statements is important.

The Containment accident sequence characteristics selected for use in definition of the North Anna source term release categories are:

Containment Bypass (Event V/SGTR)
Debris Cooled In-Vessel
Alpha Mode Containment Failure
Status of Containment Isolation
Time of Containment Failure (relative to core melt)
Time Period of Recirculation Spray Operation
Mode of Containment Failure
Auxiliary Building/Safeguards Building/Secondary System
Fission Product Attenuation Effectiveness

The reasons for selection of these parameters for use in defining the different release categories are discussed below.

Containment Building Bypass

Containment Building natural and engineered mitigation features are ineffective in reducing fission product releases if the accident causes opening of a path directly from the Reactor Coolant System to a point outside of the Containment boundary which bypasses the main Containment.

The two ways that this can occur are if a Steam Generator tube rupture occurs or an interfacing system LOCA (event V) occurs. These are both defined by the plant damage state attribute "Containment Bypass". Additionally, the Containment event tree assesses the probability for an induced SGTR as a result of high RCS temperature and pressures. These two SGTR variants are treated as the same in that the two frequencies are added together for the source term frequency.

For interfacing system LOCAs the failure occurs into the Safeguards Building because of failure of the check valves between the RCS and the LPI (Low Pressure Injection), and subsequent failure of the LPI piping outside of the Containment. Because of the size of the bypass, subsequent events in Containment are not of much interest, and the only remaining factor of interest to source term assessment is whether or not the release point is above or below water in the Safeguards Building at the time fission product releases are occurring.

If the Containment bypass is an SGTR, then a similar question arises - is the tube break submerged? MAAP-calculated results indicate that the broken Steam Generator boils essentially dry by the time release occurs. Consequently, no further source term category characteristics are required to define this class of sequences and SGTR core melt sequences are assigned to a single release category.

The Event V and SGTR classes are explicitly treated in the release category logic because they represent the two major ways that fission products can be directly released outside of Containment relatively early in time without any significant Containment mitigation of the source term.

Containment Building bypass is the first heading in the source term grouping logic diagram. This question is asked directly for all CET sequences.

Debris Cooled In-Vessel

This characteristic is important (for non-bypass sequences) since there is a significant probability of arresting the core-melt process in-vessel, thus preventing vessel failure, ex-vessel release (core-concrete interaction) processes and Containment failure.

If the debris is cooled in-vessel, then the next and only major question would be if the Containment is isolated. The Containment integrity is not challenged for accident sequences terminated in-vessel, so only isolation failures would allow significant release to the atmosphere.

This characteristic is only considered if Containment is not bypassed. Interfacing system LOCAs (Event V) and SGTRs generally preclude long-term cooling as RCS and RWST inventory is lost from Containment. The induced SGTR case, by definition, has progressed significantly past start of core melt (in order to have failed the tubes) and thus there is little chance of recovery (none was modeled).

Alpha Mode Containment Failure

This attribute is considered only for non-bypass sequences where the core melt process has not been arrested in-vessel.

The alpha mode failure is important because it allows the direct release of fission products to the atmosphere at the time of vessel failure. This is because of the assumption that the same dynamic forces that cause failure of the top closure head of the vessel simultaneously cause a large-area failure in the Containment. Alpha mode failures are included as a source term characteristic and a CET heading because the uncertainties regarding this phenomena are large and, at the upper end of the uncertainty range, the contribution of alpha mode failures to the probability of early Containment failure may not be negligible.

This attribute is important only for non-bypass CET sequences where the core melt process has not been terminated in-vessel.

Containment Building Isolation Status

This attribute is considered important because any fission products in the Containment atmosphere are released to the environment early (i.e., near the time of core melt) and continuously, if the Containment is not isolated.

In this case, the effective available time for fission product deposition and possible spray washout in Containment is reduced. The size of the most likely isolation failure path (~ 4 in^2) is large enough so that even if a later larger area Containment failure were to occur it should not significantly increase radionuclide release magnitudes.

For sequences assigned the attribute of Not Isolated, the remaining important question is whether or not the sprays are operating. It is assumed that the leak path is either directly to atmosphere, or, if to the Auxiliary Building, to a location where further attenuation is not effective. This assumption is partly based on conservatism and partially on a review of the available release pathways.

Time of Containment Building Failure

This release category attribute is considered important because it affects the time available for fission product release mitigation by natural removal processes and spray washout. It applies to all CET sequences that do not involve Containment bypass, loss of isolation, alpha mode failures, or core melt arrest in-vessel.

The times selected as significant are Early, Late, and Late Late (very late). Containment failures are treated in the CETs for the relevant sequences. Early Containment failure is at, or near, the time of reactor vessel failure. Late Containment failures occur hours after vessel failure. Late Late is a time longer than Late, say at least 24 or more hours after vessel failure, and represents an ultimate failure mode of Containment.

The possibility of no Containment failure exists and is assigned to its own unique source term category.

The attribute, time of Containment Building failure, is assigned directly from the characteristics of the particular CET branch for the headings of early, late and long term Containment failures.

Time Recirculation Sprays Operate

This attribute is considered significant because it determines whether or not fission product washout occurs in the Containment.

This attribute also affects the energy level (temperature) of the release. It implicitly affects the amount of ex-vessel fission product and aerosol release by cooling the debris on the floor below the point where a core-concrete interaction occurs and/or by covering the debris with a layer of water.

The possible time periods over which Recirculation Spray operation is considered are:

- Continuous - the sprays are operational from the time core melting starts and fission products are released from the RCS to the end of the accident

- Early Only - the sprays are operational from the time of initial core damage up to a few hours after vessel failure

- Late Only - the sprays come on several hours after vessel failure but prior to a late Containment failure in time to avoid a late failure if Containment heat removal is present, or, if an early failure has occurred still in sufficient time to reduce fission product release from core/concrete interactions

- Never - the Recirculation Sprays never operate (the Quench Sprays may have operated but they are not considered effective for fission product removal because of timing considerations)

The bases for selecting the particular time period to assign are based entirely on the branch attributes that the CET sequence has for the early and later Recirculation Spray questions in the CET. Thus for release category characteristics, the definition of "EARLY" and "LATE" are exactly those used in the CET.

The question is not considered for alpha mode failure sequences, since sprays are assumed to fail if an alpha mode Containment failure occurs. For Containment bypass sequences, the sprays do not attenuate the important in-vessel releases. It is irrelevant for all sequences in which Containment failure does not occur as no significant release would occur.

The sprays are considered to operate continuously for all sequences that have the core melt arrested in-vessel. Since the melt was arrested, in-vessel injection and recirculation must have occurred. Long-term operation of the Low Pressure Recirculation System requires that Containment heat removal (CHR) be available. Operation of CHR implies that the Recirculation Sprays are operational.

The sprays are also considered to operate continuously for all sequences with a Late Late/basemat melt-through failure mode. The sprays and Containment heat removal must have been operating for these cases to preclude earlier over-pressure failures. (The melt-

through occurs because, even though the debris is overlain by water, the debris may not be cooled (a phenomenological possibility in the CET).

Mode of Containment Building Failure

This attribute is important because it governs the rate at which fission products are released to the atmosphere. It also affects the magnitude of the release by governing the time available for effective fission product attenuation in Containment.

The two attributes considered significant are leak-type or larger (Rupture or Catastrophic Rupture). These are evaluated using the branch attributes for the CET headings Mode of Early Containment Failure and Mode of Late Containment Failure so the definitions of Leak, Rupture and Catastrophic Rupture are those employed in the CET.

This attribute is only considered for those sequences evaluated to have an early or late Containment failure and for which the Recirculation Sprays operate early only or late only.

This attribute is clearly not a discriminant for sequences with no Containment failure, and alpha mode sequences (for which it is classically assumed that the Containment failure is very large). This attribute is not relevant, (or at least not significant) for Containment bypass sequences, as most of the fission products escape through the bypass. It is not considered relevant for sequences with the core melt arrested in-vessel as Containment failure is highly unlikely. Containment failure is also considered not relevant for other sequences that have an isolation failure for the reasons that the fission products will leave by the isolation path defect, that Containment structural failure is less likely because of the pressure relief path, and the overall frequency of such sequences is very low.

This attribute is not considered an important attribute for sequences with late Containment failure and the Recirculation Sprays running continuously as this class of sequences would be expected to have small source terms regardless of Containment failure size. Similarly for sequences with continuous sprays and early Containment failure only leak type failures would be expected.

Auxiliary/Safeguards/Secondary(Side) Fission Product Attenuation

This attribute is considered important in cases where a high level of fission product attenuation can occur in structures outside of the primary Containment.

The only sequences that this attribute is evaluated as an important discriminant are the V-sequence(s) in which there is a high probability that low pressure piping break location may be flooded over by the time fission products are being vented through the break.

This attribute is not used as a discriminant for non-isolated sequences because of their low frequency of occurrence and since the flow pathway for one of the two dominant isolation failures would be expected to be directly to the environment.

Fission product attenuation effectiveness is not considered applicable to cases with large Containment structural failures because mechanisms do not exist to remove significant amounts of fission products at the high flow rates that would initially exist (if the failure is into an external building, the building will probably fail in any event) and since the likelihood of Containment shell failure into the auxiliary (or Safeguards Building) is of relatively low probability. The attribute is not relevant to sequences with no Containment failure.

This attribute is not considered as a discriminant for all other Containment failure sequences.

The assignment of this attribute to V sequences is derived directly from the heading of the same name in the V sequence CET (Figure 4.5.1-4).

4.7.2 STC Quantified Logic Diagram and STC Characteristics

The quantified Source Term Category Logic diagram for NAPS is shown in Figure 4.7.2-1. The endpoints of the logic diagram represent the individual source term categories and the pathway through the diagram (i.e., the set of decision paths taken at each decision branch define the attributes for each source term category). Twenty four individual STCs are defined. It is structurally the same as the Surry Source Term Logic Diagram a fact which is utilized in specifying source term magnitudes in Section 4.7.3 below.

All Level 1 sequences are represented in this diagram. The frequencies shown at intermediate branch nodes on this diagram are the sum of the frequency of the branches stemming from this node. They are developed by simply combining branches starting from the "end" or rightmost end of the diagram and working back to the starting point recording the intermediate sums.

The STC diagram was quantified as follows:

- The STC assignment rules shown in Table 4.7.2-1 were used to assign the CET end points (sequences) to the STCs of this

diagram. The STC assignments shown on the CETs referred to in Section 4.5 above are the ones assigned in this step.

- The frequency for each containment sequence was calculated as the product of the plant damage state frequency for that tree and the end point (dependent) probability developed for that sequence as described in Section 4.5 above.

- The frequencies of all the assigned sequences for each of the categories were summed. These are shown in the rightmost column of the STC diagram for each category.

4.7.3 Source Term Characteristics

The Surry and North Anna plants are physically similar and it is reasonable to assume that the severe accident fission product release and containment mitigation factors would also be similar. Some of the physical characteristics are compared in Table 4.3.1-1. As reported in Section 4.4, the containment design, strength and failure modes of North Anna and Surry are sufficiently similar that the Surry containment values are suitable for use in this North Anna study. It has been assumed therefore that, for the purposes of this PRA, the plants are sufficiently similar such that the release fractions for similar source term categories are also sufficiently similar to allow the use of the Surry release fractions (Virginia Power, 1991) to characterize the North Anna release fractions. (The same source term grouping criteria are used for North Anna as for Surry and so there is a one-to-one correspondence between the North Anna and the Surry source term category reference numbers.) The basis for the Surry release calculations is presented below and is taken directly from the Surry IPE report.

For the Surry IPE, MAAP calculations were performed to assess the source terms for release categories 2,5,7,8,11,13,15,18,21,23, and 24. The release fractions for the other categories were then characterized by similarity to one of the calculated source terms. For the North Anna IPE, release category 18 was redefined because there was a significant difference between the North Anna and Surry MAAP calculations which showed that Containment isolation sequences at NAPS are dominated by the 'core vulnerable' sequences. The reason for this difference is that at Surry, the LHSI pumps will fail prior to Containment failure because the pump bearing design temperature is lower than the Containment failure temperature. At NAPS, the LHSI pumps will continue to operate until Containment failure occurs at which point the LPI system is assigned a failure probability of .02. Therefore, core damage and vessel rupture will occur in the same time frame as Containment failure. NAPS source terms for release category 18 are therefore higher than those for Surry.

The representative sequences used for the MAAP release category calculations were selected on the basis of having the required characteristics as set forth in the release category definitions (Section 4.7.2), having a significant frequency of occurrence (at the plant damage state level) and having a significant Containment event tree probability. An effort was made to introduce some variety into the sequence selection (that is, not using the same Level 1 sequences for all calculations). Calculations were performed for 10 of the 24 release categories. The sequences modeled for these runs are listed in Table 4.7.3-1. The accident process analyses for release categories 15, 23 and 24 have been previously described in Section 4.6.3. The calculated times of occurrence of events important to radionuclide release are listed in Table 4.7.3-2 (for Surry). Timing for radionuclide (CSI) release is presented in Table 4.7.3-8. Here, the start time for the release and the release duration is given for in-vessel and ex-vessel releases. Finally, Table 4.7.3-9 presents the heat of release for the different release categories.

The calculated radionuclide release fractions for the analyzed release categories are shown in Table 4.7.3-3. This table is labeled as the "Composite" release fractions. It is a copy of the corresponding Surry IPE table. The release fractions in this table are listed by MAAP "species" which are described below. The bases for characterizing the unanalyzed categories are primarily the Containment failure mode and time together with a consideration of the time periods of spray operation. This is summarized in Table 4.7.3-4.

In MAAP, once fission products leave the core in-vessel or core debris ex-vessel, the chemical state is "frozen" and defined by the twelve species listed in Table 4.7.3-5. The chemical state is important in determining the transition between vapor and aerosol forms which affects the deposition and retention of fission products.

Each fission product specie can exist, in MAAP, in up to four states in each region of the Containment and of the Primary System. These states are "vapor", "aerosol", "deposited", and "contained in the core or in corium". These states, and the species of Table 4.7.3-5, are used here to characterize the calculated source term characteristics.

The highest radionuclide release fractions are found to occur for the case of the SGTR sequence with the open pathway to atmosphere (STC 24). This is because the broken Steam Generator has boiled dry previous to the onset of radionuclide release from the core, so that little mitigation occurs in the secondary system.

Large release fractions are also calculated for the interfacing LOCA (V) sequence when the release point is not submerged (STC 23).

Releases begin earlier (at 6 hours vs. 20 hours) than for the SGTR case, but proceed at a lower rate because of the protracted period of core melting.

The next highest release is for the case of an early large Containment rupture with no spray operation (STC 8). Here there is a period (2 hrs) between the onset of core damage and Containment failure during which some deposition occurs. A somewhat lower release occurs when the early Containment failure is smaller (a "leak"), because of increased retention times (STC 7).

The other case calculations have release fractions that can be characterized as small, since they are generally lower by an order of magnitude or more, (except for the noble gases) from the source terms for STC 23, 24 and 7.

The other release fractions are generally what would be expected. The tellurium releases are smaller than found by analysis using the Source Term Code Package (cf. the comparisons noted below) because of the in-vessel Te release model option selected in the MAAP code. When this option is toggled (i.e., large in-vessel Te releases are predicted), the Te release increased significantly (e.g., from 0.0026 to 0.55 for the SGTR case). On the other hand the MAAP strontium and cesium releases are higher than for the STC Package. There is also variability in the other non-volatiles reflecting uncertainties and variations in process modeling and rates.

The noble gas fraction released to the environment for STC 2 is low because, for the sequence modeled, there was no continuing outflow from the Containment even though a "leak" size failure occurred. The action of the Quench and Recirculation Sprays keeps the pressure at or below atmospheric, and with the core debris covered and/or frozen, there is no volumetric gas addition to the Containment after vessel rupture. A similar circumstance occurs for the non-isolated in-vessel arrest case (STC 21).

Release fractions calculated with MAAP for the V sequence and for an SBO with an early, large Containment failure are compared with similar reported cases in Table 4.7.3-6. The comparison data were calculated with the Source Term Code Package (STCP), or its components (Silberburg et al., 1986). Agreement is quite good, except for tellurium (presumably due to the modeling option selected as noted previously), and strontium and barium, where MAAP predicts somewhat higher values for the V sequence.

Considering the frequency of occurrence of the sequences that contribute to each category (see Section 4.7.4), it is seen that SGTR (STC 24) is clearly the most significant release category both because of the high frequency of occurrence and the high release fractions. For the situation of internal initiators, the only other categories of significance are the two V categories (STC 22 and STC 23) because of the combination of high release and

frequency of expected occurrence. All other categories are of much lower importance because of the combination of smaller release and/or much lower frequency of occurrence.

SGTR sequence releases, for the in-vessel phase, are compared with STCP predictions (Denning, 1990) in Table 4.7.3-7(a). The reported MAAP Te release is with the in-vessel release option on. Again the agreement is quite good, with the MAAP results on the high side for the less volatile elements. Table 4.7.3-7(b) compares certain radionuclide groups by location at vessel failure. The agreement is reasonably good considering modeling differences, as shown by the timing of key events listed in Table 4.7.3-7(c).

NUREG-1150 results for Surry "Late Failure" and "Containment Bypass" are compared in Figures 4.7.3-1 and 4.7.3-2 (taken from NUREG-1150). Superimposed on these plots are comparable MAAP results for STC 15 and STC 23, respectively. These are shown as cross-hatched circles. The bands for the release fraction result from the various sequences that were calculated for each class. The MAAP values are generally above the means calculated for the V sequence but within the 95th percentiles, except for Te, and are overall interpreted as being in good agreement. For the Late Failures the MAAP iodine release is below the NUREG-1150 median but above the 5th percentile. The cesium release is higher than the STCP/SURSOR values as is the barium release. The Te release is above the mean. The others are too small to report. Considering that the release fractions are very small in the case of late failure, the agreement is acceptable.

The bands shown in Figures 4.7.3-1 and 4.7.3-2 can be considered as a minimum estimate of the uncertainty in source term calculations. They are mostly two orders of magnitude or more in extent.

Two sensitivity analyses were performed in regard to the release fractions. One has previously been mentioned - where, for the sequence used for SGTR, the turning-on of the MAAP in-vessel Te release option raised the Te release fraction from .0026 to .55. Another calculation was made to show the benefit of sprays (with Containment heat removal) in the case of a non-isolated Containment. The basis analysis for STC 21 presumes that the sprays operate, consistent with the fact that low pressure core cooling has been established in time to arrest core melt in-vessel. A more improbable circumstance is that the sprays would not operate. These two situations are compared in the table below.

Release Fractions

Specie	With Sprays	Without Sprays
NOBLES	6.8E-4	5.2E-1
CSI	7.6E-5	5.3E-2
CSOH	7.6E-5	5.3E-2
SRO	2.7E-7	2.9E-2
M002	2.9E-7	1.6E-3
LA203	1.4E-11	1.8E-3

The mitigation benefit of the sprays is apparent. There are two effects that contribute to this. The heat removal inherent in spray operation keeps the Containment pressure subatmospheric and little outflow occurs and this is the most important factor here. There is also the "washout" effect of the sprays themselves that causes retention of radionuclides in the Containment.

4.7.4 STC Frequencies and Dominant Sequences

The STC frequencies are shown on Figure 4.7.4-1. Table 4.7.4-1 presents these release categories ranked by frequency. Table 4.7.4-2 shows the contribution of the most significant, at approximately the 1E-8 level, PDS contributors to each release category. It should be noted that the STC release point estimate frequencies shown on the Figures and Tables in this section are the plant damage state point estimate frequencies multiplied by the conditional probability that a sequence in a particular plant damage state will have a Containment response outcome (as modeled by the Containment event tree) that results in its assignment to a particular source term release category (summed over all plant damage states).

Figure 4.7.4-2 shows the percentage of Bypass STCs. SGTRs from both the initiating event and induced from high RCS pressures and temperatures, are 11% of the total frequency. Event V sequences represent 2%. Figure 4.7.4-3 shows that 12% of the non-bypass sequences are arrested in-vessel. Figure 4.7.4-4 shows a breakdown of the time of Containment failure for all sequences with the containment intact at vessel failure. No Containment failure occurs in 84% of these cases. No Containment failure occurs in 89% of all of the non-bypass sequences (this includes the cooled in-vessel STCs, the Alpha mode STC and the Not Isolated/Not Intact STCs.) Early Containment failures (e.g., loss of isolation, alpha mode failures and early over-pressure failures) are predicted for only 11% of the total plant damage state sequence frequency. As shown in Figure 4.7.4-5 sprays are not operating in 56% of the sequences and operate throughout the accident or early only in 43% of the sequences. For the sequences with the Containment intact at

vessel failure, as shown on Figure 4.7.4-6, 54% of the subsequent failures are leak type, 32% are large ruptures, and 14% are predicted as potential basemat melt-throughs.

As listed in Table 4.7.3-3, the most significant release fractions (for example CsI releases greater than 10% of the core inventory) are estimated to occur for the unmitigated bypass sequences (STCs 23 and 24),and the early large ruptures (STCs 4,6,8 and 19.) These STCs with large release fractions represent 12% of the core damage frequency, almost entirely due to the relatively high frequency of occurrence of the SGTR STC. The next largest release group (say between 1% and 10% CsI release) includes the early failure leak types without spray operation (STC 7) and the Event V sequences with the release point under water (STC 22), and represents about 2% of the total core damage frequency. The remaining 86% of the accident frequency results in low or negligible releases.

The noteworthy observations that can be made are as follows:

- Given the same accident initiators, early Containment failures are somewhat more likely at North Anna than at Surry because of, among other factors, the relatively larger NAPS core and RCS in the same volume and strength containment as Surry.

- The "core vulnerable" class of sequences comprise those accidents where the core is being maintained in an undamaged state with the SI systems in recirculation mode but without an operable means of containment heat removal. The water being pumped is in excess of 300°F, which is the stated qualification temperature of the pumps. If the pumps were to fail at a high temperature (necessarily just before containment failed), then all these sequences would go to core damage. If this type of failure occurred at a probability of 1, the core damage frequency would increase by 1E-5 and it is likely that Containment failure would occur shortly after core melt/vessel failure in all these sequences. This makes the high temperature capability of the low pressure SI systems in recirculation mode significant.

- SGTR sequences are very important high release accidents, just as at Surry. This is driven by the frequency of random tube failure initiating the accident and includes a high fraction of cases where the steam generator secondary side is failed open to the atmosphere.

4.8 SENSITIVITY CALCULATIONS

4.8.1 Introduction

An important element of the Level 2 Containment analyses is addressing the question: "To which aspects of the Containment modeling are the overall results most sensitive?" The structured sensitivity analysis presented here aids in the identification of possible weaknesses in the analysis or areas which may require further effort or further support.

A sensitivity analysis can be represented by the equation.

$$\Delta P_{rc} = f\ (\Delta P_{event})$$

where ΔP_{rc} is the change in an important output (for example the change in the conditional probability of a source term release category)

ΔP_{event} is the change in value of an input to the model (for example, a change in an event split fraction probability)

and f is the functional relationship between the two defined by the overall backend Containment event analysis.

Important sensitivities can be considered as those where a "reasonable" change in a basic CET (or DET) event probability results in a significant change in the overall results. For example, a change may produce a significant increase (decrease) in the probability of a high source term release category and a corresponding decrease (increase) in a low source term release category. A "reasonable" change in a basic event probability refers to a change that is within the assessed uncertainty range for the event probability.

For phenomenological events the range of reasonable values to use in a sensitivity analyses is not always evident. These event probabilities can be interpreted as being degrees-of-belief in the outcome of an uncertain event where only one outcome is physically possible but we are not completely certain which is the correct one. Two approaches can be taken for these type events. The first approach acknowledges that either event may be possible and that our probability estimates merely state our belief as to which is most likely to be the correct outcome. For this approach we would set the value of one event branch equal to 1 (and the other branches equal 0) and assess the impact on release category probabilities. This type of approach addresses the question of what the impact on the final results are if this event branch is the correct one for the phenomenological process. We then systematically assign a value of 1 to the other event branch

probabilities and repeat the analyses. A second approach to sensitivity analyses for phenomenological events (which is related to assessment of uncertainties) is to investigate the impact of variations in the degree-of-belief probability estimates on the overall results for each of the phenomenological events (i.e., change the assessed probabilities from the baseline values but not necessarily to (1,0), (0,1) combinations discussed above. This approach is analogous to assuming what other experts might select.

The decomposition event trees were reviewed and the phenomenological events which were judged to either have large uncertainties or were expected to have a substantial influence on important outcomes were identified. Sensitivity calculations were performed for most of these phenomenological events. The parameters which were varied in the Level 2 sensitivity study are listed in Table 4.8-1 along with the parameter variations investigated in each case. For most parameters the sensitivity calculation involved changing one branch probability to one (with all other branch probabilities set to zero) and requantifying the CETs. This was then repeated for each branch. These types of calculations are identified as (1,0) in Table 4.8-1. For several parameters it was more appropriate to use branch probabilities in the sensitivity analysis which were increased/decreased by a multiplicative factor (generally 10) from the base case value.

The sensitivity studies were performed using the PDS frequencies without internal flooding. A complete list of the parameter variations and the results obtained for each case are contained in the Tables of Appendix F.6.

The principal observations made for the sensitivity studies are presented below.

4.8.2 Results and Conclusions

Induced Hot Leg Failure Sensitivities

Sensitivity calculations were performed to assess the impact of assumptions regarding the probability of induced RCS failure on important level 2 results. For this sensitivity calculation the probability for induced RCS failure for "High" and "Hi Hi" pressure sequences was set equal to zero for case A1 and to one for case A2. For both cases the probability of induced SGTR was set to zero. For Case A1 (no induced RCS failures), there was a marginal decrease in the frequency of debris cooled in-vessel sequences (10.4% to 9.2% of the total core damage sequence frequency). For case A2 (hot leg failure for all "High" and "Hi Hi" pressure sequences) the increase in the frequency of debris cooled in-vessel sequences was smaller from 10.4% to 10.9%.

The significant outcome with respect to Containment failure for this sensitivity calculation was a doubling in the frequency of early Containment failures if no induced RCS failures was assumed (Case A1) from 1.1% to 2.3% of the total CDF. Similarly, if an induced RCS failure was assumed at all times for RCS pressures above 2000 psig (Case A2), the frequency for early Containment failures would decrease from 1.1% to 0.7% of the total CDF. The frequency of all other Containment failure modes/times were relatively unchanged for this calculation.

Alpha Mode Containment Failure Sensitivities

The alpha mode Containment failure probability was increased (Case B1) and decreased (Case B2) by a factor of 10 in this calculation. As expected, increasing the alpha mode Containment failure probability by a factor of ten from the base case results in the alpha mode failures becoming the dominant mode of early Containment failure. With the alpha mode probability decreased by a factor of 10 alpha mode failures are insignificant contributors to early Containment failure.

In-Vessel Debris Cooling Sensitivities

Several events on the Debris Cooled In-vessel DET were included in this sensitivity analysis.

The first sensitivity study was to vary the split fraction for in-vessel cooling when the Low Pressure Injection System is operating for large LOCAs without functional accumulators. The probability for successful in-vessel cooling was changed to zero (Case C1) and to one (Case C2). Since the base case probability for successful in-vessel cooling under these conditions was 0.95 the Case C2 results are very similar to the base case. However, where the in-vessel cooling probability was set equal to zero (Case C1) there was a 4% decrease (10.4% to 6.4%) in the frequency of core damage sequences terminated prior to reactor vessel failure.

The second study varied the in-vessel cooling split fraction for sequences where the Low Pressure Injection System is available but the system pressure is initially too high to allow injection and where system depressurization occurs after core damage has commenced. The probability of successful in-vessel cooling was set equal to zero for Case D1 and was set equal to one for Case D2. As for the Case C1/C2 calculations discussed above the base case value for successful in-vessel cooling is large (.9). Hence the Case D2 results are similar to the base case. Case D1 indicates that the overall frequency of core damage sequences successfully mitigated in-vessel would decrease by only 1.2% (from 10.4% to 9.2%) if in-vessel cooling is not possible under this specific set of conditions.

The third study varied the in-vessel cooling probability for sequences where in-vessel injection is recovered (e.g. power recovery from a station blackout) after core damage has initiated. For Case E1 the probability of successful in-vessel cooling following power recovery (after core damage has been initiated) was set to zero and for Case E2 this probability was set to one. For case E1 the overall frequency of core damage sequences mitigated in-vessel decreased 5.1% from the base case and for Case E2 the overall frequency was increased by 2.2%.

Mode of Early Containment Failure Sensitivities

Several sensitivity calculations were performed for the events in the Mode of Early Containment Failure DET, including the amount of hydrogen produced in-vessel, the fraction of mass involved in DCH, the extent of hydrogen burn at reactor vessel failure, and the Containment failure pressure. Each of these variations is discussed below.

The sensitivity calculation on the amount of hydrogen produced in-vessel considered increasing each branch probability from its base value to unity (Cases G1 and G2). The impact on the important level 2 results of changing the fraction of zirconium oxidized in-vessel was negligible for either extreme. The PDS frequencies were changed only slightly. For example, the frequency of early Containment failure varied only (0.1%) from the base case.

The importance of the DCH phenomenon to the time of Containment failure was investigated by increasing (Case H1) and decreasing (Case H2) the probability of the largest DCH events by a factor of ten. As expected, there was a shift toward early Containment failure when the probability of a large DCH was increased by a factor of ten. The early Containment failure source term category frequency increased from 1.1% to 5.1%.

The sensitivity to Uncontrolled Hydrogen Burn (UCHB) probability results were similar to those for the DCH sensitivity. There is an increase in the frequency of early Containment failure and a decrease in the frequency of late Containment failure. This is as expected since both phenomenon contribute to Containment pressurization at vessel failure. As with the DCH sensitivity calculation, the change represented about a 4.4% increase in the frequency of early Containment failures (i.e., 1.1% base case to 5.5% with UCHB probability = 1).

A Containment failure pressure sensitivity calculation was run assuming that the (actual) Containment failure pressure was the pressure (93 psig) at the fifth percentile on the Containment fragility curve (Case J1). The frequency of early Containment failure increased by 2.6%. However, even with the Containment failure pressure set to the 5th percentile value on the fragility

curve, early over-pressure failure still only accounts for 3.7% of the overall core damage sequence frequency. In Case J2, the Containment failure pressure was assumed to be at the 50th percentile on the fragility curve (12 psig) with the Containment failing at or above this pressure and staying intact below it. Since the Containment pressures calculated for the majority of the CET end points fall below the 128 psig limit, this case resulted in a decrease in the frequency of early Containment failures (from a base case value of 1.1% to 0.4%).

Sensitivity of Early Recirculation Spray Failure Probability

A sensitivity study (Case K1) was performed to assess the impact of changes in the probability of early failure of the Recirculation Sprays due to severe accident physical phenomena and environmental conditions. The relative frequencies of different PDS groups representing the different times of Containment failure changed only slightly when sprays are assumed to not fail due to Containment conditions. Late CF decreased slightly and the no Containment failure sequence increased slightly when sprays are assumed to not fail due to Containment conditions. The frequencies of early Containment failure and very late Containment failure sequences remained unchanged.

Sensitivity to Debris Cooled Ex-Vessel

The Debris Cooled Ex-Vessel DET contains three branches which are quantified using split fractions: debris dispersal, depth of the debris pool, and debris cooled ex-vessel. The branch probabilities for each of these events were varied in the sensitivity analyses (to the extreme values of zero - one).

All three sensitivity calculations showed similar results. None of these changes in the split fractions produced a significant change in the frequency of the source term categories representing the different times of Containment failure. The one exception is in the Containment base mat melt-through failure mode. When the debris depth is assumed to be deep or when no ex-vessel cooling is assumed, the frequency of melt-through will increase by as much as 3.1% of the core damage frequency.

Sensitivity of Late Recirculation Spray Failure Probability

These sensitivity calculations and results were similar to those described under "Sensitivity of Early Recirculation Spray Failure Probability" above.

Frequency of Event-V

The frequency of the Event-V (PDS 24) plant damage state was changed to equal the NUREG/CR-4550 5th and 95th percentiles (Cases P1 and P2). The impact of these changes in the Event V sequence frequency was evaluated by assessing the increase/decrease in the frequency of Event V sequences relative to other sequences with the potential for early and large source terms (e.g., SGTRs, loss of isolation, early Containment failure). For the case (P2) where the Event V frequency was increased to the NUREG/CR-4550 95% percentile frequency (5.1E-6 per year), the relative contribution of Event V sequences increased by a factor of greater than three. With the V sequence frequency reduced to the NUREG/CR-4550 5th percentile value (3.9E-11 per year), these sequences become insignificant contributors.

Loss of Containment Building Isolation

The loss of Containment Building isolation (PDS 1 and 2) sensitivity was analyzed by increasing the loss of isolation frequency by a factor of ten (Case Q1) and then decreased by a factor of ten (Case Q2). The results of this analysis were compared to the dominant PDSs from the base case, the frequency for early Containment failure sequences and for Containment bypass sequences. None of the important source term category fractions showed any impact from increasing or decreasing the loss of isolation frequency within the range described. It is noted that the source term categories (17, 18 and 21) for loss of isolation sequences changed substantially in magnitude. However, the frequency associated with failure to isolate Containment is still a very small fraction of the overall core damage frequency (i.e., less than .5%).

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TABLE 4.1.1-1
NORTH ANNA DESIGN INFORMATION

Basic Information About the Plant

Type of Reactor	Pressurized Water Reactor	
Manufacturer	Westinghouse	
Date of Commercial Operation	1978 - Unit 1 1980 - Unit 2	
Reactor Core		
Nominal Power	2893 MWt	9874E+6 Btu/hr
Number of fuel assemblies	157	
Number of fuel rods	41,448	
Core weight		
Uranium dioxide	82,160 kg	181,128 lb
Zircaloy	17,108 kg	37,718 lb
Miscellaneous	7,120 kg	15,700 lb
Average Fuel Enrichment	.037 atom fraction	
Reactor Vessel		
Inside diameter	3.98 m	157 in
Overall internal height	13.0 m	42.5 ft
Thickness at beltline (excl. clad)	0.20 m	7.9 in
Lower Head thickness	0.130 m	5.125 in
Water capacity with core and internals in place	96 m^3	3390 ft^3
Reactor Coolant System		
Volume (nominal, including (PZR)	282 m^3	9957 ft^3
Water in system (nominal)	192,000 kg	423,200 lb
Operating temperature (vessel average)	306°C	583°F
Operating pressure (nominal)	15.5 MPa	2250 psia
Number of Reactor Coolant Pumps	3	
Number of Steam Generators	3	
Type of Steam Generators	U-tube (W Model 51°F)	
Containment		
Inside diameter	38.4 m	126 ft
Maximum inside height	58.1 m	190.6 ft
Free volume	51,680 m^3	1,825,000 ft^3
Design pressure	413.7 kPa	45 psig
Operating pressure	72 kPa	10.4 psia
Operating temperature	43.3°C	110°F

TABLE 4.1.1-1 (Continued)
NORTH ANNA DESIGN INFORMATION

Construction	Reinforced concrete	
Wall thickness	1.4 m	4.5 ft
Dome thickness	0.76 m	2.5 ft
Basemat thickness	3.0 m	10.0 ft
Floor thickness above liner (outside cavity only)	0.61 m	2.0 ft
Containment Liner	Welded steel liner	
Liner thickness, walls	0.95 cm	0.375 in
Liner thickness, dome	1.27 cm	0.500 in
Liner thickness, floor outside cavity	0.64 cm	0.250 in
Liner thickness, cavity floor	1.90 cm	0.750 in
Reactor Cavity		
Annular Cavity	3.4 m radius	11 ft radius
In-Core Instrument Room	3.0 x 7.3 m	10 x 24 ft
Floor area (cavity & ICIR)	57.6 m^2	620 ft^2
Water capacity (cavity & ICIR) (to bottom of RV)	250 m^3	8828 ft^3
Refueling Water Storage Tank		
Volume	1844 m^3	487,000 gal
RWST Temperature	7.2°C	45.0°F
Casing Cooling Tank		
Volume	466 m^3	123,000 gal
Temperature	7.2°C	45°F
Inside Recirculation Spray Pumps		
Number	2	
Design flow (each)	.21 m^3/sec	3300 gpm
Design head	0.81 MPa	269 ft
Outside Recirculation Spray Pumps		
Number	2	
Design flow (each)	.23 m^3/sec	3700 gpm
Design head	0.86 MPa	286.7 ft
Recirculation Spray Heat Exchangers		
Number	4	
Design capacity (each)	1.67 MW	56.8E6 Btu/hr

TABLE 4.1.1-1 (Continued)
NORTH ANNA DESIGN INFORMATION

Accumulators		
Number	3	
Pressure	4.6 MPa	665 psia
Water capacity (total)	86 m^3	3039 ft^3
Containment Structure		
Reinforced Concrete		
Subatmospheric		
Concrete type - Basaltic Aggregate		
Free H_2O	3.86 w/o	
Bound H_2O	2.0 w/o	
Pressurizer Safety valves		
Number	3	
Capacity(each)	48.0 kg/s	380,000 lbm/hr
Setpoint	17.2 MPa	2500 psia
Pressurizer PORVs		
Number	2	
Capacity(each)	26.5 kg/s	210,000 lbm/hr
Setpoint	16.1 MPa	2335 psia
RCS flowrate (normal full power)		
3 pumps	6.1 m^3/sec	96,400 gpm
RCS Enthalpy (normal full power) hf =	1.377E6 J/kg	592.2 Btu/lbm
hg =	2.740E6 J/kg	1177.8 Btu/lbm
Vessel Internal Structural Masses		
Core barrel	34721 kg	76545 lbm
Core baffle	11968 kg	26385 lbm
Thermal shield	29573 kg	65196 lbm
Core former plates	9476 kg	20891 lbm
Lower core plate	2182 kg	4810 lbm
Lower support plate	11863 kg	26153 lbm
Diffuser plate	1208 kg	2663 lbm
Lower instrument/supports/ tie plates/base plate	6552 kg	14444 lbm
Upper support plate	10152 kg	22381 lbm

TABLE 4.1.1-1 (Continued)
NORTH ANNA DESIGN INFORMATION

Other upper internals (upper core plate/control rod guide shafts and housings/support columns)	20648 kg	45520 lbm
Vessel lower head mass	24650 kg	54343 lbm
Control Rods Material Mass		
Number	960	
Silver Mass	2202 kg	4854 lbm
Indium Mass	413 kg	910 lbm
Cadmium Mass	138 kg	304 lbm
Quench Spray Pumps		
Number	2	
Design flow (each)	.1262 m^3/sec	2000 gpm
Shutoff head	.74 MPa	240 ft
Spray Initiation setpoint	.19 MPa	27.75 psia
Fan Coolers - Not considered in Containment evaluation		
Charging Pumps [High Pressure Safety Injection (HPSI) pumps]		
Number	3 (only 2 can operate simultaneously)	
Design flow	1.89E-2 m^3/sec	300 gpm
Design head	17.8 MPa	5800 ft
Safety Injection initiation	12.2 MPa	1770 psia
Shutoff head	18.4 MPa	2673 psia
Low Pressure Injection Pumps		
Number	2	
Design flow	.278 m^3/sec	4400 gpm
Design head	.55 MPa	180 ft
Safety Injection initiation	12.2 MPa	1770 psia
Shutoff head	1.2 MPa	175 psia
Residual Heat Removal Pumps		
Number	2	
Design flow	.252 m^3/sec	4000 gpm
Design pressure	4.24 MPa	600 psig
Design temperature	204°C	400°F
RH Heat Exchangers		
Number	2	
Design capacity	8.94E6 J/sec	30.5E6 Btu/hr

TABLE 4.1.1-1 (Continued)
NORTH ANNA DESIGN INFORMATION

Safeguards Building		
Free volume	1140 m^3	40,251 ft^3

Auxiliary Building
Not considered in Containment evaluation

Source for Table 4.1.1-1 Information

1. North Anna MAAP Parameter File Analysis File 326MAF.N.1
2. North Anna UFSAR
3. (Breeding, 1990)
4. (Ritzman, 1985)
5. (Cole, 1984)
6. Virginia Power Internal Documents
   a. NCRODP-52, NAPS
   b. NCRODP-40, NAPS
   c. NCRODP-53, NAPS

TABLE 4.1.1-2
NORTH ANNA PASSIVE HEAT SINKS

Heat Sink No.	Heat Sink Description	Area FT^2	Heat Sink Thickness	Description Material	Mass lbm
1	Interior Walls, Floors	8148.0	0.500 ft	Concrete	(a)
2	Interior Walls, Floors	60458.0	1.000 ft	Concrete	
3	Interior Walls, Floors	53752.0	1.500 ft	Concrete	
4	Interior Walls, Floor	11254.0	2.000 ft	Concrete	
5	Interior Walls, Floor	9130.0	2.250 ft	Concrete	
6	Interior Walls, Floors	3530.0	2.000 ft	Concrete	
7	Containment Wall Below Grade	20108.0	0.375 in.	C. Steel	
8	Containment Wall Above Grade	24576.0	0.375 in.	C. Steel	
9	Containment Dome	24656.0	0.500 in.	C. Steel	
10	Containment Mat & Sub-floor	11757.0	2.200 ft	Concrete	
			0.250 in.	C. Steel	
			10.000 ft	Concrete	
11	Stainless Steel	16968.0	0.306 in.	St. Steel	
12	Carbon Steel	8690.0	0.152 in.	C. Steel	
13	Cable Tray & Conduit	26769.0	0.066 in.	C. Steel	
14	Grating	26573.0	0.094 in.	C. Steel	
15	Ductwork	27167.0	0.018 in.	C. Steel	
16	Carbon Steel	35520.0	0.250 in.	C. Steel	
17	Carbon Steel	4019.0	0.529 in.	C. Steel	
18	Carbon Steel	3803.0	0.984 in.	C. Steel	
19	Carbon Steel	8928.0	1.535 in.	C. Steel	
20	Carbon Steel, Polar Crane	193.0	2.532 in.	C. Steel	

(a) Mass can be calculated by taking the product of area X thickness X density. The density for concrete = 145 lb/ft^3, carbon steel and stainless steel = 490 lb/ft^3.

TABLE 4.2.1-1
MODELS AND FEATURES ADDED FOR PWR MAAP 3.B REVISION 17

Feature	Minor Rev.	Description	Impact
Coding/ Output Improvement	16.01/ 16.02	Improve FORTRAN coding structure and readability of code output	No impact on results
Integration Logic	16.02	1) Limit Prim. Sys. press. change rate to 2% when non-equilibrium model is used prior to RV failure 2) Properly update the time step based on the limiting time step 3) Limit the time step relaxation to no more than twice the previous value	Improved code performance
Numerical Improvement	16.03	Thermal-hydraulics related with core heatup and melt progression have been reviewed, enhanced and some discontinuity points are identified and removed	Reproducibility of revision 16.03 results from single, double and PC runs has been greatly improved
Break Flow	16.03	Improved break flow void fraction calculation and flowrate for better stability when injection occurs	More stable break flow
Quench Tank Location	16.04	Allow users to locate the quench tank in either lower or annular compartment	Increase user flexibility

TABLE 4.2.1-1 (Continued)
MODELS AND FEATURES ADDED FOR PWR MAAP 3.B REVISION 17

Feature	Minor Rev.	Description	Impact
Generalized ESF	16.04	The generalized ESF models independent pump systems; Each system can have its own water source and discharge location, thus allowing the users to model the exact ESF lineups at their plants	Users can model accidents more realistically
Generalized Fan Cooler	16.04	1) Allow the users to select the suction/discharge locations of the existing fan cooler 2) Add a second fan cooler (chiller)	Increase user flexibility
Half-Loop	16.04	Add several new features to allow the user simulating severe accidents during half-loop operation (operation after normal shutdown with decreased primary system inventory	Increase user flexibility
Turbine Driven AFW	16.04	Allow auxiliary feedwater to be operated by turbine driven pump	Increase user flexibility
SG PORV Control	16.04	Allow broken/unbroken steam generator PORVs to be activated independently	Increase user flexibility

TABLE 4.2.1-1 (Continued)
MODELS AND FEATURES ADDED FOR PWR MAAP 3.B REVISION 17

Feature	Minor Rev.	Description	Impact
Spray Model	16.04	Update the gas temperature and steam partial pressure during droplet fall	Reduce the messages from SPRAY/FLOW, improve code performance
Direct Containment Heating (DCH)	16.05	Add three new features, 1) Allow entrainment to occur to upper and lower compartments simultaneously 2) Include Cr and Fe reactions. 3) Allow Zr, Cr, and Fe to react with the oxygen in the entry volume	Expect more H2 and energy release during DCH
Jet Burn	16.05	Do not allow jet burning whenever the entry region is steam inerted or lacks of enough oxygen	May reduce the amount of jet burning
Corium-Water Interaction (EXVIN)	16.05	Allow time delay for corium-water interaction in cavity after the dropping corium contacts with the floor	Expect more steam generation and lower corium temperature in the cavity at time of vessel failure
H2 from Corium-Water Interaction (PLH2)	16.05	Track the corium droplet temperature during the Zr-water reaction by considering reaction heat and convective heat	Expect more hydrogen generation and higher water temperature

TABLE 4.2.1-1 (Continued)
MODELS AND FEATURES ADDED FOR PWR MAAP 3.B REVISION 17

Feature	Minor Rev.	Description	Impact
Parameter File Input	16.04	1) Improve parameter change format 2) Make code maintenance easier, 3) Make INPUTS generic	Increase user flexibility
User Events and Actions	16.04	1) Allow user-defined automatic actions 2) Allow user-defined time delays 3) Enhance user-defined event I/O	Users can model operator actions more realistically
To Mass Balance/ Correction	16.05	1) Clarify the Te mass balance in tabular output 2) Allow Te to release in the NUREG-0772 model after 90% of Zr oxidized	No impact on Te release if FTEREL=1 is used or less 90% Zr oxidized; Otherwise, the timing of Te release may be different

TABLE 4.3.1-1
COMPARISON OF CERTAIN NORTH ANNA AND SURRY FACTORS

ITEM	NAPS	SURRY	(NAPS-SURRY) /SURRY, %
Power, MWth	2893	2441	18
Containment Volume, ft3	1.825E6	1.8E6	0
RCS Volume, ft3	9957	9508	5
Zircaloy, lbs	41289	36300	14
(RATIOS)			
Containment Volume/Power	630.8	737.4	-14
Containment Volume/Zircaloy	44.2	49.6	-11
Containment Volume/RCS Volume	183.3	189.3	-3
RWST Water Volume, gallons	466200*	387100	
Casing Cooling Tank Water, gallons	123000**	-	
RWST Level @ Recirc Switchover, %	26-29	18-19	
RWST Water Left @ Switchover, gal	151030	86200	
Net RWST Injection, gallons	315000	301000	5
Spray Injection after switch, gal	274000	86200	218
Quench Spray Pump, gpm	2000	3200	
IRS,gpm	3300	3500	
ORS,gpm	3700	3500	

NOTES: * Minimum
** Injected into ORS suction

TABLE 4.3.2-1
PLANT DAMAGE STATE BINNING LOGIC RULES

```
   LOGIC RULES for  REV2NAPS.PDD
              Rule for   CONBYPASS
IF A:VX == FAILURE;
THEN EVENT V;
IF A:T7 == FAILURE   * A:P  == FAILURE;
IF A:T7 == FAILURE   * A:SGI== FAILURE;
IF A:T7 == FAILURE   * A:O  == FAILURE;
IF A:T7 == FAILURE   * A:D1 == FAILURE  * A:L == FAILURE;
THEN SGTR;
DEFAULT NO BYPASS;
              Rule for   CONISOLAT
IF A:H1 == FAILURE;
IF A:H2 == FAILURE;
THEN ISOLATED;
IF A:IS == FAILURE;
IF A:D3 == SUCCESS * A:Rs == FAILURE;
IF A:D3 == SUCCESS * A:Ch == FAILURE;
IF A:H2 == SUCCESS * A:Rs == FAILURE;
IF A:H2 == SUCCESS * A:Ch == FAILURE;
IF A:H1 == SUCCESS * A:Rs == FAILURE;
IF A:H1 == SUCCESS * A:Ch == FAILURE;
THEN NOT ISOLATED;
IF A:IS != FAILURE;
THEN ISOLATED;
              Rule for   TRANLOCA
IF A:A    == FAILURE;
IF A:RX   == FAILURE;
THEN LARGE LOCA;
IF A:S1   == FAILURE;
IF A:S2   == FAILURE;
IF A:T7 == FAILURE;
IF A:Q == FAILURE;
IF A:P == SUCCESS;
IF A:S1c == FAILURE;
IF A:T4     == FAILURE * A:O == FAILURE;
IF A:T6     == FAILURE * A:O == FAILURE;
IF A:T8     == FAILURE * A:O == FAILURE;
IF A:T1Tr   == FAILURE * A:O == FAILURE;
IF A:T2Tr   == FAILURE * A:O == FAILURE;
IF A:T2ATr == FAILURE * A:O == FAILURE;
IF A:T3Tr   == FAILURE * A:O == FAILURE;
IF A:T9ATr == FAILURE * A:O == FAILURE;
IF A:T9BTr == FAILURE * A:O == FAILURE;
THEN SMALL/MED LOCA;
DEFAULT TRANSIENT;
```

TABLE 4.3.2-1 (Continued)
PLANT DAMAGE STATE BINNING LOGIC RULES

```
            Rule for   SBO
IF A:T1A   == FAILURE;
IF A:T8       == FAILURE * A:RC1 != SUCCESS;
IF A:T6       == FAILURE * A:RC1 != SUCCESS;
IF A:T1Tr     == FAILURE * A:RC1 != SUCCESS;
IF A:T2Tr     == FAILURE * A:RC1 != SUCCESS;
IF A:T2ATr    == FAILURE * A:RC1 != SUCCESS;
IF A:T3Tr     == FAILURE * A:RC1 != SUCCESS;
IF A:T9ATr    == FAILURE * A:RC1 != SUCCESS;
IF A:T9BTr    == FAILURE * A:RC1 != SUCCESS;
THEN YES;
DEFAULT NO;
            Rule for   POWRECOV
IF A:B     == SUCCESS;
IF A:B1    == SUCCESS;
IF A:RC2   == SUCCESS;
THEN PRIOR RV FAIL;
IF A:B2    == SUCCESS;
IF A:RC3   == SUCCESS;
THEN PRIOR CONT FAIL;
DEFAULT NO POWER REC;
            Rule for   RECSPRAYS
IF A:Rs     == SUCCESS;
IF A:SPRAY == SUCCESS;
THEN YES;
DEFAULT NO;
            Rule for   CNHEATREM
IF A:Ch    == SUCCESS;
THEN YES;
DEFAULT NO;
            Rule for   INVESSINJ
IF A:H1 == FAILURE;
IF A:H2 == FAILURE;
THEN FAILED;
IF A:A == FAILURE * A:H1 == SUCCESS * A:Dh == SUCCESS;
IF A:A == FAILURE * A:D3 == SUCCESS * A:Rs == SUCCESS;
IF A:RX== FAILURE * A:D3 == SUCCESS * A:Rs == SUCCESS;
IF A:RX== FAILURE * A:Qs == SUCCESS * A:Rs == SUCCESS;
IF A:S1 == FAILURE * A:H1 == SUCCESS;
IF A:S1 == FAILURE * A:H2 == SUCCESS;
IF A:S2 == FAILURE * A:H2 == SUCCESS;
THEN ON;
IF A:A == FAILURE * A:H1 == SUCCESS * A:Dh == FAILURE;
IF P:SBO==NO *P:RCSPRESS!=LO LO * A:H1==SUCCESS;
IF A:S2 == FAILURE * A:H1 == SUCCESS;
THEN LPI DEADHEAD;
IF P:POWRECOV==PRIOR RV FAIL * P:RCSPRESS !=LO LO*A:H1==SUCCESS;
THEN RECOVERED;
DEFAULT FAILED;
```

TABLE 4.3.4-1
PLANT DAMAGE STATES RANKED BY FREQUENCY

Rank	PDS No.	Frequency	Percent of 6.80E-5
1	PDS 21	1.926E-5	28.33
2	PDS 4	1.268E-5	18.66
3	PDS 20	8.219E-6	12.09
4	PDS 25	7.011E-6	10.31
5	PDS 14	2.924E-6	4.30
6	PDS 12	2.916E-6	4.29
7	PDS 23	2.647E-6	3.89
8	PDS 5	2.596E-6	3.82
9	PDS 3	2.066E-6	3.04
10	PDS 24	1.601E-6	2.35
11	PDS 13	1.424E-6	2.09
12	PDS 8	1.274E-6	1.87
13	PDS 7	1.146E-6	1.69
14	PDS 18	1.055E-6	1.55
15	PDS 15	2.478E-7	0.36
16	PDS 11	2.349E-7	0.35
17	PDS 9	2.306E-7	0.34
18	PDS 16	1.223E-7	0.18
19	PDS 1	1.183E-7	0.17
20	PDS 6	6.821E-8	0.10
21	PDS 22	5.442E-8	0.08
22	PDS 17	4.046E-8	0.06
23	PDS 2	3.143E-8	0.05
24	PDS 19	4.311E-9	0.01
25	PDS 10	1.624E-9	0.00

TABLE 4.3.4-2
SIGNIFICANT LEVEL 1 CONTRIBUTORS TO THE PLANT DAMAGE STATES

Plant Damage Category: 1	**1.183E-7**	
	Amount	**Percent**
S2P02	3.217E-8	27.19
S1P08	3.189E-8	26.96
ISP01	2.849E-8	24.08
AP04	1.600E-8	13.52
T7P09	4.129E-9	3.49
S2P30	2.521E-9	2.13
S2P15	1.178E-9	1.00

Plant Damage Category: 2	**3.143E-8**	
	Amount	**Percent**
ISP02	9.501E-9	30.23
S2P03	7.565E-9	24.07
S1P09	7.504E-9	23.87
AP05	4.280E-9	13.62
T7P10	1.036E-9	3.30
S2P31	5.992E-10	1.91
S2P16	3.499E-10	1.11

Plant Damage Category: 3	**2.066E-6**	
	Amount	**Percent**
T1AP46	1.407E-6	68.10
T1AP02	6.510E-7	31.51

Plant Damage Category: 4	**1.268E-5**	
	Amount	**Percent**
T8P22	3.169E-6	24.99
T1AP51	2.990E-6	23.58
T8P02	2.517E-6	19.85
T9ATrP08	1.525E-6	12.03
T1AP07	1.385E-6	10.92
T9ATrP02	8.325E-7	6.57
T3TrP22	1.206E-7	0.95
T3TrP02	5.829E-8	0.46
T2ATrP22	4.745E-8	0.37
T2ATrP02	2.192E-8	0.17

Plant Damage Category: 5	**2.596E-6**	
	Amount	**Percent**
T3TrP03	1.572E-6	60.54
T2ATrP03	6.403E-7	24.66
T3TrP23	1.843E-7	7.10
T2ATrP23	7.506E-8	2.89
T2TrP03	5.821E-8	2.24
T9BTrP03	1.886E-8	0.73
T9ATrP03	1.869E-8	0.72
T8P03	1.011E-8	0.39

TABLE 4.3.4-2 (Continued)
SIGNIFICANT LEVEL 1 CONTRIBUTORS TO THE PLANT DAMAGE STATES

Plant Damage Category: 6	6.821E-8	
	Amount	Percent
T9ATrP11	1.889E-8	27.69
T9ATrP05	1.167E-8	17.11
T8P25	1.131E-8	16.59
T1AP55	8.079E-9	11.84
T8P05	6.352E-9	9.31
T3TrP05	3.843E-9	5.63
T2ATrP05	1.563E-9	2.29
T1AP11	1.426E-9	2.09
T8P24	1.288E-9	1.89
T1AP54	1.112E-9	1.63
T8P04	9.192E-10	1.35

Plant Damage Category: 7	1.146E-6	
	Amount	Percent
T8P06	6.057E-7	52.85
T3TrP06	2.829E-7	24.69
T2ATrP06	1.146E-7	10.00
T9ATrP06	1.066E-7	9.30
T8P26	2.016E-8	1.76

Plant Damage Category: 8	1.274E-6	
	Amount	Percent
T1P15	5.161E-7	40.51
T9BP02	2.369E-7	18.60
T9AP02	1.719E-7	13.50
T2P14	1.301E-7	10.21
T5BP02	9.370E-8	7.35
T5AP02	9.360E-8	7.35
T3P15	1.924E-8	1.51
T2AP15	1.228E-8	0.96

Plant Damage Category: 9	2.306E-7	
	Amount	Percent
THP30	2.058E-7	89.23
T9BP03	1.150E-8	4.99
T1P16	4.392E-9	1.90
T5AP03	3.560E-9	1.54
T5BP03	3.506E-9	1.52

Plant Damage Category: 10	1.624E-9	
	Amount	Percent
T9BP04	5.439E-10	33.49
T1P17	3.496E-10	21.53
T5AP04	2.885E-10	17.77
T5BP04	2.885E-10	17.77
T2P16	1.535E-10	9.45

TABLE 4.3.4-2 (Continued)
SIGNIFICANT LEVEL 1 CONTRIBUTORS TO THE PLANT DAMAGE STATES

Plant Damage Category: 11	2.349E-7	
	Amount	Percent
T1P19	1.910E-7	81.31
T9BP06	1.779E-8	7.57
T5AP06	9.631E-9	4.10
T5BP06	9.326E-9	3.97
T9AP06	6.246E-9	2.66

Plant Damage Category: 12	2.916E-6	
	Amount	Percent
AP15	2.120E-6	72.70
AP02	5.170E-7	17.73
RXP01	2.663E-7	9.13

Plant Damage Category: 13	1.424E-6	
	Amount	Percent
AP03	8.258E-7	57.99
AP11	5.883E-7	41.32

Plant Damage Category: 14	2.924E-6	
	Amount	Percent
T3TrP11	1.670E-6	57.12
T2ATrP11	6.777E-7	23.18
T1AP58	4.171E-7	14.27
T8P11	9.565E-8	3.27
T2TrP11	6.060E-8	2.07

Plant Damage Category: 15	2.478E-7	
	Amount	Percent
T3TrP08	6.671E-8	26.92
T8P08	5.848E-8	23.60
T1AP22	4.887E-8	19.72
T1AP14	3.179E-8	12.83
T2ATrP08	2.708E-8	10.93
T3TrP12	6.163E-9	2.49
T8P12	3.206E-9	1.29
T2ATrP12	2.511E-9	1.01

Plant Damage Category: 16	1.223E-7	
	Amount	Percent
T3TrP09	4.702E-8	38.45
T8P09	3.899E-8	31.88
T2ATrP09	1.916E-8	15.66
T1AP15	1.143E-8	9.35
T3TrP13	2.363E-9	1.93
T2TrP09	1.741E-9	1.42

TABLE 4.3.4-2 (Continued)
SIGNIFICANT LEVEL 1 CONTRIBUTORS TO THE PLANT DAMAGE STATES

Plant Damage Category: 17	4.046E-8	
	Amount	**Percent**
T3TrP10	1.271E-8	31.41
T8P10	1.045E-8	25.83
T2ATrP10	5.150E-9	12.73
T3TrP15	5.086E-9	12.57
T1AP16	3.217E-9	7.95
T2ATrP15	2.072E-9	5.12
T3TrP14	6.079E-10	1.50
T2TrP10	4.675E-10	1.16

Plant Damage Category: 18	1.055E-6	
	Amount	**Percent**
T1AP67	8.863E-7	84.01
T1AP26	1.038E-7	9.84
T9ATrP22	2.005E-8	1.90
T3TrP16	1.444E-8	1.37

Plant Damage Category: 19	4.311E-9	
	Amount	**Percent**
T3TrP20	1.758E-9	40.79
T8P20	8.720E-10	20.23
T2ATrP20	8.176E-10	18.97
T1AP72	7.582E-10	17.59
T2TrP20	7.433E-11	1.72

Plant Damage Category: 20	8.219E-6	
	Amount	**Percent**
T1TrP17	4.003E-6	48.70
T1P10	2.705E-6	32.92
T2P09	7.225E-7	8.79
T9ATrP17	3.066E-7	3.73
T9BP13	1.297E-7	1.58
T9AP13	1.022E-7	1.24
T1P36	7.750E-8	0.94
T1P46	5.423E-8	0.66
T3P10	4.873E-8	0.59
T2AP10	4.550E-8	0.55
T9BTrP17	1.762E-8	0.21

Plant Damage Category: 21	1.926E-5	
	Amount	**Percent**
S2P35	5.154E-6	26.76
S1P38	4.039E-6	20.97
S1P10	2.451E-6	12.73
S2P04	2.451E-6	12.73
S2P43	1.189E-6	6.18

TABLE 4.3.4-2 (Continued)
SIGNIFICANT LEVEL 1 CONTRIBUTORS TO THE PLANT DAMAGE STATES

T1TrP14	1.014E-6	5.26
T1P07	5.658E-7	2.94
S2P39	5.204E-7	2.70
T9ATrP14	3.882E-7	2.02
S2P47	3.271E-7	1.70
THP46	2.142E-7	1.11
T9BP10	1.791E-7	0.93
T1P06	1.690E-7	0.88
T9AP10	1.311E-7	0.68
S2P32	9.120E-8	0.47
S2P17	9.024E-8	0.47
T1P21	8.168E-8	0.42
S2P26	5.452E-8	0.28
T9BTrP14	2.529E-8	0.13
S1P46	2.493E-8	0.13
T2P06	2.434E-8	0.13
T1P11	2.214E-8	0.11
S1P42	1.364E-8	0.07
S2P23	1.198E-8	0.06
Plant Damage Category: 22	**5.442E-8**	
	Amount	**Percent**
T9ATrP19	1.688E-8	31.02
T1TrP19	1.203E-8	22.11
S1P39	6.189E-9	11.37
S2P36	5.348E-9	9.83
T1TrP15	4.283E-9	7.87
S1P11	3.206E-9	5.89
S2P05	2.442E-9	4.49
T1P12	1.482E-9	2.72
S2P44	1.264E-9	2.32
T2P11	8.978E-10	1.65
Plant Damage Category: 23	**2.647E-6**	
	Amount	**Percent**
T1TrP21	2.224E-6	84.01
T1P14	2.072E-7	7.83
S2P06	5.128E-8	1.94
S1P12	5.091E-8	1.92
S2P38	3.257E-8	1.23
S1P41	1.526E-8	0.58
T1TrP16	1.103E-8	0.42
Plant Damage Category: 24	**1.601E-6**	
	Amount	**Percent**
VXP07	1.524E-6	95.18
VXP03	7.682E-8	4.80

TABLE 4.3.4-2 (Continued)
SIGNIFICANT LEVEL 1 CONTRIBUTORS TO THE PLANT DAMAGE STATES

Plant Damage Category: 25	7.011E-6 Amount	Percent
T7P04	2.984E-6	42.56
T7P03	1.983E-6	28.29
T7P06	1.104E-6	15.75
T7P26	3.853E-7	5.50
T7P23	1.799E-7	2.57
T7P07	1.096E-7	1.56
T7P25	8.444E-8	1.20
T7P27	7.198E-8	1.03
T7P14	3.147E-8	0.45
T7P22	3.104E-8	0.44
T7P15	2.590E-8	0.37
T7P24	1.861E-8	0.27

TABLE 4.6.1-1
MAAP ACCIDENT PROCESS ANALYSIS BASE DESCRIPTIONS

Case	Description [1]
1	6" Cold Leg LOCA NO SI, NO AFW, 2 CS, 2 IRS, 2 ORS (estimate RCS pressure at RV failure)[2]
2	4" Cold Leg LOCA NO SI, NO AFW, 2 CS, 2 IRS, 2 ORS (estimate RCS pressure at RV failure)
3	2" Cold Leg LOCA NO SI, NO AFW, 2 CS, 2 IRS, 2 ORS (estimate RCS pressure at RV failure)
4	2" Cold Leg LOCA NO SI, AFW ON, 2 CS, 2 IRS, 2 ORS (estimate RCS pressure at RV failure)
5	29" Cold Leg LOCA 1 Charging Pump, 1 LPI, 2 MD MFW, 2 CS, 2 IRS, 2 ORS No accumulators (see extent of core damage)
6	VOID
7	LT SBO w/o seal LOCA (20 gpm per pump leakage) (determine time between core damage and Containment high pressure for estimating power recovery probability)
8	LT SBO with seal LOCA (200 gpm per pump leakage) (determine time between core damage and Containment high pressure for estimating power recovery probability)
9	ST SBO with seal LOCA (200 gpm per pump leakage) (determine time between core damage and Containment high pressure for estimating power recovery probability)
10	ST SBO w/o seal LOCA (20 gpm per pump leakage) (determine time between core damage and Containment high pressure for estimating power recovery probability)
11	ST SBO Large DCH 50% debris participation (DCH Parametric Calculation)

TABLE 4.6.1-1 (Continued)
MAAP ACCIDENT PROCESS ANALYSIS BASE DESCRIPTIONS

Case	Description [1]
12	ST SBO Large DCH 50% debris participation Entrain debris into upper Containment (DCH Parametric Calculation)
13	ST SBO Blockage model turned on (In-vessel Hydrogen Generation Parametric Calculation)
14	Large Break LOCA (29 in.dia.) NO SI, no sprays, 3 accumulators (Test Containment pressure at RV failure)
15	Intermediate Break LOCA (6 in.dia.) NO SI, no sprays, 3 accumulators (Test Containment pressure at RV failure)
16	Small Break LOCA (2 in.dia.) NO SI, no sprays, 3 accumulators (Test Containment pressure at RV failure)
17	ST SBO w/o seal LOCA (20 gpm per pump leakage) A large break hot leg LOCA occurs at core slump. (Determine pressure in Containment resulting from overtemperature induced hot leg failure)
18	Large Break LOCA NO SI, 2 CS, 2 IRS, 2 ORS available, 3 accumulators (Test Containment pressure at RV failure)
19	Small Break LOCA (2 in) NO SI, 2 CS, 2 IRS, 2 ORS available, 3 accumulators (Test Containment pressure at RV failure)
20	Large Break LOCA NO SI, no sprays, 3 accumulators Blockage model turned on (In-vessel Hydrogen Generation Parametric Calculation)
21	Small Break LOCA (2 in) NO SI, no sprays, 3 accumulators Blockage model turned on (In-vessel Hydrogen Generation Parametric Calculation)

TABLE 4.6.1-1 (Continued)
MAAP ACCIDENT PROCESS ANALYSIS BASE DESCRIPTIONS

Case	Description [1]
22	ST SBO Large DCH 100% debris participation Entrain debris into upper Containment (DCH Parametric Calculation)
23	ST SBO Large DCH 50% debris participation Entrain debris into upper Containment Force H2 Burn at vessel failure (DCH Parametric Calculation)
24	ST transient Turn off all in-vessel ST and AFW All sprays systems available Large DCH 100% debris participation Entrain debris into upper Containment (DCH Parametric Calculation)

Notes:

(1) Explanation of Certain Abbreviations

SI - Safety Injection, i.e. both High Head Charging Systems and Low Head Injection Systems

AFW - Auxiliary Feedwater System

CS - Containment Injection Spray Train

IRS - Inside Recirculation Spray Train

ORS - Outside Recirculation Spray Train

LT SBO - Long Term Station Blackout - AFW available for several hours

ST SBO - Short Term SBO - no AFW

RCS, LOCA, RV, DCH have their usual meaning

(2) Comments in parentheses are the nominal objectives for the various cases.

TABLE 4.6.1-2
TIMING OF KEY EVENTS FOR MAAP ACCIDENT PROGRESSION CASES

Case	Core Uncovery Time (sec)	Fuel Temp >2200°F Time (sec)	Support Plate Failure Time (sec)	Vessel Failure Time (sec)	Cont Press >108 psia Time (sec)	Cont Press >143 psia Time (sec)	Zr Reacted In Vessel (%)
1	368.3	2600	5395.7	---	---	---	28.9
2	717.7	6150	10014	---	---	---	36.5
3	2503	3500	22192	29392	---	---	58
4	2511	3530	---	---	---	---	38.9
5	9.2	100	---	---	---	---	1.3
6	Void						
7	75428	8.2E4	87657	88157	1.84E5	2.3E5	59.6
8	15972	2E4	26017	26517	---	---	48.7
9	7914	1E4	14921	15421	9.2E4	1.3E5	62.7
10	8864	1.13E4	14772	15272	7.6E4	1.2E5	54.8
11	7914	9.75E3	14907	15407	---	---	62.6
12	7914	990	14907	15407	---	---	58.4
13	7914	1E4	14436	14936	9E4	---	31.7
14	7.6	1150	3891	4391	---	---	31.4
15	1766	2600	5381	5881	---	---	28.3
16	2502	3500	24896	25396	---	---	55.3
17	8946	1.13E4	14852	15352	---	---	56.2
18	7.6	1000	3561	4061	---	---	27.4
19	2503	3500	22192	22692	---	---	44.3
20	7.6	1125	3659	4159	---	---	20.9
21	2503	3500	15974	16474	---	---	32.4
22	7914	9.6E3	14907	15407	---	---	57.9
23	7914	9.7E3	14907	15407	---	---	57.9
24	5995	7.75E3	12759	13259	---	---	51.6

TABLE 4.6.1-3
CALCULATED QUANTITIES FOR MAAP ACCIDENT PROGRESSION CASES

Case	H2 Produced In-Vessel (kg)	H2 Produced Ex-Vessel (kg)	RCS Pressure @ RV Fail (MPa)	Cont Pressure Before RV Fail (MPa)	Peak Cont. Press After RV Fail (MPa)	H2 Burned Cont Press Rise @ RV (MPa)	Outside Cavity (kg)	H2 Burned @ RV Fail (kg)
1	210	0.0	---	---	---	---	0	---
2	265	0.0	---	---	---	---	0	0
3	420	1.3	0.4	.079	.079	0.0	158	---
4	280	0.0	---	---	---	---	67	---
5	9.5	0.0	---	---	---	---	0	---
6	Void							
7	430	0.0	16.94	.148	.309	.161	0	---
8	350	420	4	.15	.3	.15	0	0
9	455	0.0	17.2	.195	.2875	.0925	0	---
10	400	0.19	16.7	.2	.298	.098	0	0
11	420	92	17.2	.195	.47	.275	214	---
12	420	92	17.2	.195	.505	.310	266	---
13	230	120	16.2	.172	.370	.198	0	---
14	225	370	.31	.21	.27	.06	0	---
15	205	290	.83	.24	.252	.012	0	---
16	400	0.28	5.28	.283	.418	.138	0	---
17	400	0.28	15.7	.2	.32	.12	0	0
18	194	19	.0796	.0784	.0791	.0007	0	---
19	330	0.6	7.0	.078	.125	.047	68	---
20	152	340	.21	.21	.27	.06	0	---
21	235	0.36	7.0	.23	.37	.14	---	---
22	420	180	.17	.18	.6	.42	365	---
23	420	100	16.5	.18	.6	.42	487	487
24	375	260	15.2	.08	.58	.5	618	618

TABLE 4.6.3-1
LONG TERM STATION BLACKOUT (LTSBO) MAAP EVENT SUMMARY

EVENT SUMMARY L2C28 RUN 09

0.0	4	MAIN COOLANT PUMPS OFF
0.0	13	REACTOR SCRAM
0.0	46	LETDOWN FLOW OFF
0.0	103	CONTMT SPRAYS OFF
0.0	118	VP-TURBINE-DRIV EN AUXILIARY FEEDWATER TO BSTGEN
0.0	127	VP-TURBINE-DRIV EN AUXILIARY FEEDWATER TO USTGEN
0.0	156	MSIV CLOSED
0.0	205	POWER NOT AVAILABLE
37909.7	40	PZR SOLID
38009.7	40	PZR HAS STEAM
38749.7	40	PZR SOLID
39269.7	40	PZR HAS STEAM
39669.7	40	PZR SOLID
46729.7	118	VP-TURBINE-DRIV EN AUXILIARY FEEDWATER TO BSTGEN
46729.7	127	VP-TURBINE-DRIV EN AUXILIARY FEEDWATER TO USTGEN
46729.7	191	CST WATER DEPLETED
71629.1	151	BROKEN S/G DRY
71629.1	161	UNBKN S/G DRY
73153.5	92	Q/T RUPTURE DISK FAILED
74245.4	81	WATER ON LOWER CMPT FLOOR
74272.6	113	VP-SUFF NPSH FOR IRS
74272.6	116	VP-SUFF NPSH FOR ORS
75367.8	113	VP-INSUFF NPSH FOR IRS
75367.8	116	VP-INSUFF NPSH FOR ORS
75901.4	116	VP-SUFF NPSH FOR ORS
75921.4	113	VP-SUFF NPSH FOR IRS
76081.4	113	VP-INSUFF NPSH FOR IRS
76121.4	116	VP-INSUFF NPSH FOR ORS
76181.4	40	PZR HAS STEAM
76215.7	215	MCP SWITCH OFF OR HI-VIBR TRIP
76233.7	116	VP-SUFF NPSH FOR ORS
76251.8	40	PZR SOLID
76251.8	113	VP-SUFF NPSH FOR IRS
76549.9	40	PZR HAS STEAM
76569.9	40	PZR SOLID
76570.8	40	PZR HAS STEAM
76573.7	40	PZR SOLID
76574.5	111	VP-AUTO ACTUATION SIGNAL FOR IRS RECEIVED
76574.5	114	VP-AUTO ACTUATION SIGNAL FOR ORS RECEIVED
76586.5	25	PS NONEQ THERMO
76661.2	40	PZR HAS STEAM
77473.4	14	FP MODELS ON
77473.4	49	CORE HAS UNCOV

TABLE 4.6.3-1 (Continued)
LONG TERM STATION BLACKOUT (LTSBO) MAAP EVENT SUMMARY

87704.8	57	WATER IN CAVITY
89768.6	2	SUPPORT PLATE FAILED
89836.7	28	DWNCMR NOT BLCKD FOR GAS XPORT
90268.6	3	RV FAILED
90268.6	59	WATER FLOODING IN CAVITY TO B
90268.7	61	CORIUM IN CAVITY
90270.2	57	CAVITY DRY
90271.4	59	WATER NOT FLOODING IN CAVITY TO B
90271.4	58	CORIUM FLOODING IN CAVITY TO B
90271.4	59	WATER FLOODING IN CAVITY TO B
90271.4	82	CORIUM IN LOWER CMPT
90271.4	57	WATER IN CAVITY
90272.9	27	UNBKN LOOPS NOT BLOCKED AT PUMP BOWLS
90275.7	61	NO CORIUM IN CAVITY
90276.7	58	CORIUM NOT FLOODING IN CAVITY TO B
90277.2	57	CAVITY DRY
90279.9	57	WATER IN CAVITY
90284.2	59	WATER NOT FLOODING IN CAVITY TO B
90284.3	59	WATER FLOODING IN CAVITY TO B
90284.6	59	WATER NOT FLOODING IN CAVITY TO B
90284.7	59	WATER FLOODING IN CAVITY TO B
90285.0	59	WATER NOT FLOODING IN CAVITY TO B
90285.0	59	WATER FLOODING IN CAVITY TO B
90285.3	59	WATER NOT FLOODING IN CAVITY TO B
90285.3	59	WATER FLOODING IN CAVITY TO B
90285.6	59	WATER NOT FLOODING IN CAVITY TO B
90285.7	59	WATER FLOODING IN CAVITY TO B
90286.1	59	WATER NOT FLOODING IN CAVITY TO B
90286.1	59	WATER FLOODING IN CAVITY TO B
90286.7	59	WATER NOT FLOODING IN CAVITY TO B
90286.8	59	WATER FLOODING IN CAVITY TO B
90286.9	59	WATER NOT FLOODING IN CAVITY TO B
90302.1	188	ACCUMULATOR WATER DEPLETED
90362.4	113	VP-INSUFF NPSH FOR IRS
90362.4	116	VP-INSUFF NPSH FOR ORS
90971.7	61	CORIUM IN CAVITY
103977.6	65	CAV CPLD MODEL USED
164679.3	116	VP-SUFF NPSH FOR ORS
164699.3	116	VP-INSUFF NPSH FOR ORS
164719.2	116	VP-SUFF NPSH FOR ORS
164739.2	116	VP-INSUFF NPSH FOR ORS
164758.7	116	VP-SUFF NPSH FOR ORS
164778.2	116	VP-INSUFF NPSH FOR ORS
164797.7	116	VP-SUFF NPSH FOR ORS
164817.2	116	VP-INSUFF NPSH FOR ORS
169581.5	116	VP-SUFF NPSH FOR ORS
169601.5	116	VP-INSUFF NPSH FOR ORS

TABLE 4.6.3-1 (Continued)
LONG TERM STATION BLACKOUT (LTSBO) MAAP EVENT SUMMARY

169621.5	116	VP-SUFF NPSH FOR ORS
169821.5	113	VP-SUFF NPSH FOR IRS
170181.5	81	LOWER CMPT FLOOR DRY
216101.5	65	CAV UNCPLD MODEL USED
216121.5	65	CAV CPLD MODEL USED
216141.5	65	CAV UNCPLD MODEL USED
216301.5	65	CAV CPLD MODEL USED
216581.5	65	CAV UNCPLD MODEL USED
238301.5	104	CONTMT FAILED
238402.1	113	VP-INSUFF NPSH FOR IRS
238402.1	116	VP-INSUFF NPSH FOR ORS
271856.8	57	CAVITY DRY

TABLE 4.6.3-2
V SEQUENCE EVENT SUMMARY

EVENT SUMMARY L2C33 RUN 06

0.0	209	PS BREAK(S) FAILED
0.0	238	V SEQUENCE
106.0	13	REACTOR SCRAM
106.0	117	VP-MOTOR-DRIVEN AUXILIARY FEEDWATER TO BSTGEN ON
106.0	126	VP-MOTOR-DRIVEN AUXILIARY FEEDWATER TO USTGEN ON
106.0	156	MSIV CLOSED
116.1	6	LPI ON
116.1	11	CHARGING PUMPS ON
176.5	15	UNBKN LOOP HOMOGENEOUS
13894.7	124	VP-LOW RWST -- CONTAINMENT SPRAYS TURNED OFF
14619.1	110	VP-INSUFF NPSH FOR CS
14619.1	182	A SPRAY PUMPS INSUFF NPSH
14619.1	183	CH PUMPS INSUFF NPSH
14619.1	184	LPI PUMPS INSUFF NPSH
14619.1	185	HPI PUMPS INSUFF NPSH
14619.1	187	RWST WATER DEPLETED
14619.1	189	B SPRAY PUMPS INSUFF NPSH
15570.1	4	MAIN COOLANT PUMPS OFF
15570.1	215	MCP SWITCH OFF OR HI-VIBR TRIP
15584.0	15	UNBKN LOOP PHASES SEPARATED
16208.2	25	PS NONEQ THERMO
17934.2	14	FP MODELS ON
17934.2	49	CORE HAS UNCOV
51308.7	188	ACCUMULATOR WATER DEPLETED
55184.9	2	SUPPORT PLATE FAILED
55684.9	3	RV FAILED
55684.9	59	WATER FLOODING IN CAVITY TO B
55684.9	61	CORIUM IN CAVITY
55686.2	27	UNBKN LOOPS NOT BLOCKED AT PUMP BOWLS
55690.1	59	WATER NOT FLOODING IN CAVITY TO B
55690.1	58	CORIUM FLOODING IN CAVITY TO B
55690.1	59	WATER FLOODING IN CAVITY TO B
55690.1	58	CORIUM NOT FLOODING IN CAVITY TO B
55690.1	59	WATER NOT FLOODING IN CAVITY TO B
55690.1	82	CORIUM IN LOWER CMPT
55690.1	58	CORIUM FLOODING IN CAVITY TO B
55690.1	59	WATER FLOODING IN CAVITY TO B
55690.1	57	WATER IN CAVITY
55690.1	28	DWNCMR NOT BLCKD FOR GAS XPORT
55690.1	81	WATER ON LOWER CMPT FLOOR
55694.6	57	CAVITY DRY
55694.6	61	NO CORIUM IN CAVITY
55695.3	58	CORIUM NOT FLOODING IN CAVITY TO B

TABLE 4.6.3-2 (Continued)
V SEQUENCE EVENT SUMMARY

55695.3	58	CORIUM FLOODING IN CAVITY TO B
55695.4	58	CORIUM NOT FLOODING IN CAVITY TO B
55702.7	59	WATER NOT FLOODING IN CAVITY TO B
55839.9	116	VP-SUFF NPSH FOR ORS
55851.0	113	VP-SUFF NPSH FOR IRS
55965.2	81	LOWER CMPT FLOOR DRY
56006.0	81	WATER ON LOWER CMPT FLOOR
56246.0	81	LOWER CMPT FLOOR DRY
57883.5	61	CORIUM IN CAVITY

TABLE 4.6.3-3
STGR SUMMARY OF EVENTS

EVENT SUMMARY L2C40 RUN 02

0.0	209	PS BREAK(S) FAILED
188.3	13	REACTOR SCRAM
188.3	117	VP-MOTOR-DRIVEN AUXILIARY FEEDWATER TO BSTGEN ON
188.3	126	VP-MOTOR-DRIVEN AUXILIARY FEEDWATER TO USTGEN ON
188.3	156	MSIV CLOSED
199.0	6	LPI ON
199.0	11	CHARGING8 PUMPS ON
1800.0	266	VP-BROKEN S/G PORV MANUALLY OPENED
1800.0	267	VP-UNBROKEN S/G PORV MANUALLY OPENED
2400.0	267	VP-UNBROKEN S/G PORV AUTO
23811.3	124	VP-LOW RWST LEVEL-- CONTAINMENT SPRAYS TURNED OF
24179.3	110	VP-INSUFF NPSH FOR CS
24179.3	182	A SPRAY PUMPS INSUFF NPSH
24179.3	183	CH PUMPS INSUFF NPSH
24179.3	184	LPI PUMPS INSUFF NPSH
24179.3	185	HPI PUMPS INSUFF NPSH
24179.3	187	RWST WATER DEPLETED
24179.3	189	B SPRAY PUMPS INSUFF NPSH
27052.8	188	ACCUMULATOR WATER DEPLETED
27076.6	15	UNBKN LOOP HOMOGENEOUS
27078.0	4	MAIN COOLANT PUMPS OFF
27078.0	215	MCP SWITCH OFF OR HI-VIBR TRIP
27154.6	15	UNBKN LOOP PHASES SEPARATED
46216.3	25	PS NONEQ THERMO
65931.0	151	BROKEN S/G DRY
67122.6	14	FP MODELS ON
67122.6	49	CORE HAS UNCOV
78822.9	2	SUPPORT PLATE FAILED
79322.9	3	RV FAILED
79322.9	59	WATER FLOODING IN CAVITY TO B
79322.9	61	CORIUM IN CAVITY
79327.3	27	UNBKN LOOPS NOT BLOCKED AT PUMP BOWLS
79329.5	59	WATER NOT FLOODING IN CAVITY TO B
79329.5	58	CORIUM FLOODING IN CAVITY TO B
79329.5	59	WATER FLOODING IN CAVITY TO B
79329.5	58	CORIUM NOT FLOODING IN CAVITY TO B
79329.5	59	WATER NOT FLOODING IN CAVITY TO B
79329.5	82	CORIUM IN LOWER CMPT
79329.6	59	WATER FLOODING IN CAVITY TO B
79329.6	58	CORIUM FLOODING IN CAVITY TO B
79329.6	57	WATER IN CAVITY
79330.7	81	WATER ON LOWER CMPT FLOOR

TABLE 4.6.3-3 (Continued)
STGR SUMMARY OF EVENTS

79331.2	28	DWNCMR NOT BLCKD FOR GAS XPORT
79333.5	58	CORIUM NOT FLOODING IN CAVITY TO B
79333.7	58	CORIUM FLOODING IN CAVITY TO B
79334.1	58	CORIUM NOT FLOODING IN CAVITY TO B
79335.7	57	CAVITY DRY
79341.0	59	WATER NOT FLOODING IN CAVITY TO B
79635.2	116	VP-SUFF NPSH FOR ORS
79654.2	113	VP-SUFF NPSH FOR IRS
80376.9	81	LOWER CMPT FLOOR DRY

TABLE 4.7.2-1
SOURCE TERM CATEGORY ASSIGNMENT RULES

```
RULES
RULE:CONBYPASS        9
IF P:CONBYPASS == NO BYPASS * P:CONISOLAT== NOT ISOLATED;
IF P:CONBYPASS == NO BYPASS * C:RCSFAIL != SGTR;
THEN NO BYPASS;
IF P:CONBYPASS == EVENT V;
THEN EVENT V;
IF P:CONBYPASS == SGTR;
THEN SGTR;
IF C:RCSFAIL == SGTR;
THEN SGTR;
RULE:INVCOOL          4
IF C:INVCOOL != COOLED;
THEN NOT COOLED;
IF C:INVCOOL == COOLED;
THEN COOLED;
RULE:ALPHA            5
IF P:CONBYPASS == NO BYPASS * P:CONISOLAT== NOT ISOLATED;
IF C:ALPHA != ALPHA CF;
THEN NO ALPHA CF;
IF C:ALPHA == ALPHA CF;
THEN ALPHA CF;
RULE:CONISOLAT        4
IF P:CONISOLAT == ISOLATED;
THEN ISOLATED;
IF P:CONISOLAT == NOT ISOLATED;
THEN NOT ISOLATED;
RULE:TIME-CF          8
IF C:CF-EARLY==NO EARLY CF*C:CF-LATE==NO LATE
CF*C:CF-LONG!=MELTTHRU;
THEN NO CF;
IF C:CF-EARLY != NO EARLY CF;
THEN EARLY;
IF C:CF-EARLY == NO EARLY CF * C:CF-LATE != NO LATE CF;
THEN LATE;
IF C:CF-EARLY==NO EARLY CF*C:CF-LATE==NO LATE
CF*C:CF-LONG==MELTTHRU;
THEN LATE LATE;
RULE:TIME-RS         12
IF P:CONISOLAT == NOT ISOLATED * P:RECSPRAYS == YES;
THEN CONTINUOUS;
IF P:CONISOLAT == NOT ISOLATED * P:RECSPRAYS == NO;
THEN NEVER;
IF C:RS-EARLY == NO FAILURE * C:RS-LATE == NO FAILURE;
THEN CONTINUOUS;
IF C:RS-EARLY != NO FAILURE * C:RS-LATE != NO FAILURE;
THEN NEVER;
IF C:RS-EARLY == NO FAILURE * C:RS-LATE != NO FAILURE;
```

TABLE 4.7.2-1 (Continued)
SOURCE TERM CATEGORY ASSIGNMENT RULES

```
THEN EARLY ONLY;
IF C:RS-EARLY != NO FAILURE * C:RS-LATE == NO FAILURE;
THEN LATE ONLY;
RULE:MODECF         12
IF C:CF-EARLY == LEAK;
THEN LEAK;
IF C:CF-EARLY == RUPTURE;
THEN RUP/CAT RUP;
IF C:CF-EARLY == CAT RUPTURE;
THEN RUP/CAT RUP;
IF C:CF-LATE == LEAK;
THEN LEAK;
IF C:CF-LATE == RUPTURE;
THEN RUP/CAT RUP;
IF C:CF-LATE == CAT RUPTURE;
THEN RUP/CAT RUP;
RULE:AUXSGSEC        4
IF C:AUXSGSEC == YES;
THEN YES;
IF C:AUXSGSEC == NO;
THEN NO;
```

TABLE 4.7.3-1
REPRESENTATIVE SEQUENCES FOR RELEASE FRACTION ANALYSIS

STC	MAAP Case Number	Description
2	31	2.5" dia SBLOCA, no injection, no AFW, Recirculation Sprays operate, Containment failure at vessel rupture due to DCH & H_2 burn, 0.1 ft^2 failure area
5	36	Long term SBO, 1 turbine-driven AFW, 0.1 ft^2 Containment failure at vessel rupture, sprays recovered 6 hours later
7	37	Short term SBO, no AFW, induced seal LOCA, 0.1ft^2 Containment failure at vessel rupture, no sprays
8	29	Short term SBO, no AFW, induced seal LOCA, 7 ft^2 Containment failure at vessel rupture, no sprays
11	38	2" dia. SBLOCA, no injection, sprays on initially but tripped off 1 hour after vessel failure, large late Containment failure of 7ft^2
13	35	Long term SBO, 1 turbine-drive AFW, induced seal LOCA, sprays recovered 18 hours after core damage begins
15	28	Long term SBO, 1 turbine-driven AFW, no sprays or injection, 7ft^2 Containment failure late
21	26	Large (29") Cold Leg LOCA, low pressure injection recovered to arrest core melt in-vessel, 2" diameter isolation failure, sprays operate
23	33	V sequence, break is 2.57" dia equivalent in LHSI line due to venture, no injection spray, break point never submerged.
24	40	SGTR (2 tubes), broken SG isolated, broken SG RV open after 40 minutes, tellurium not released in-vessel

TABLE 4.7.3-2
TIME OF OCCURRENCE FOR IMPORTANT EVENTS, RELEASE CATEGORY CASES

Time, in Seconds

Release Category	Coolant Pumps off	Core Uncovery	Vessel Failure	Recirc Sprays on	Recirc Sprays off	Contain-ment Failure	Cavity Flooded	Hydrogen Burns
2	0	1706	19916	1560	-	19923	2953	19916
5	0	15972	26517	48454	-	26825	53595	-
7	0	7914	15407	-	-	15711	-	15410
8	0	7914	15407	-	-	15734	-	15410
11	1458	2503	22692	112	26300	367849	4011	8357
13	0	15972	26517	316738	-	313126	319288	-
15	0	77473	90629	-	-	238301	-	-
21	3	9	-	-	-	30000	-	-
23	15570	17934	55685	-	-	-	-	-
24	27078	67123	79323	-	-	-	-	-

TABLE 4.7.3-3
COMPOSITE SOURCE TERM CATEGORY RELEASE FRACTIONS

Source Term Category	Basis*	NOBLE	CSI	SRO	MOO2	CSOH	BAO	LA2O3	CEO2	SB	TE2
		MAAP SPECIE RELEASE FRACTION									
1	R						none				
2	M	0.072	<E-5	<E-5	<E-5	<E-5	<E-5	<E-5	<E-5	<E-5	<E-5
3	R						(See STC 5)				
4	R						(See STC 8)				
5	M	0.61	7.7E-5	6.5E-4	2.6E-3	6.9E-3	5.3E-4	2.6E-4	3.3E-4	7.5E-3	1.3E-3
6	R						(See STC 8)				
7	M	0.90	0.074	1.5E-2	2.5E-2	9.7E-2	8.7E-3	8.1E-6	9.7E-5	0.13	1.4E-2
8	M	0.94	0.11	0.023	0.016	0.15	9.0E-3	1.7E-5	3.4E-4	0.24	1.7E-2
9	R						(See STC 11)				
10	R						(See STC 5)				
11	M	0.82	2.3E-6	3.2E-4	3.9E-4	1.4E-5	1.3E-5	1.8E-11	1.4E-11	1.2E-4	1.6E-5
12	R						(See STC 5)				
13	M	0.99	4.6E-3	1.6E-8	4.6E-8	3.2E-3	2.6E-6	4.5E-10	7.3E-9	5.6E-4	1.7E-5
14	R						(See STC 15)				
15	M	0.90	1.1E-4	3.1E-4	4.1E-4	3.4E-4	9.2E-5	5.2E-8	5.5E-8	3.7E-3	-
16	R						(See STC 11)				
17	R						(See STC 2)				
18	R						(See STC 8)				
19	R						(See STC 8)				
20	R						(NONE)				
21	M	6.8E-4	7.6E-5	2.7E-7	2.9E-7	7.6E-5	4.2E-8	1.4E-11	1.4E-11	1.4E-6	-
22	R						(See STC 23, DF 5.6)**				
23	M	0.94	0.29	0.23	0.28	0.31	0.15	3.6E-4	3.7E-2	0.50	1.6E-5
24	M	0.996	0.52	0.034	0.14	0.54	0.021	5.5E-5	5.2E-3	0.68	2.6E-3

Notes:

* Basis: M = MAAP Results; R = Recommended Alternate

** Use STC 23 Noble release; reduce other species by DF of 5.6 as per NUREG/CR-4551

TABLE 4.7.3-4
RECOMMENDATIONS FOR RELEASE FRACTIONS FOR UNANALYZED SOURCE TERM CATEGORIES

Category	Containment Failure	Sprays	Recommended Release Categ. Alternate	Recommendation Reason
1, 20	None		Use Zero Release	
3	Early Leak	Early	5	Ignore Early Sprays
4	Early Large	Early	8	Ignore Early Sprays
6	Early Large	Late	8	Ignore Early Spray
9	Late	Continuous	11	Assume spray same as early only
10	Late Leak	Early	5	Sprays minimize failure time
12	Late Leak	Late	5	Sprays minimize failure time
14	Late Leak	None	15	Late leak same as rupture
16	Base Mat		11	Late failure with some Sprays
17	None	Continuous	2	No isolation same as leak
18	Early Large		8	Nearest match
19	Alpha		8	Nearest match
22	V(submerged)		23÷5.6DF*	NUREG/CR-4551

Note:
* As recommended in NUREG/CR-4551

TABLE 4.7.3-5
MAAP FISSION PRODUCT SPECIES [a]

Specie Number	Specie I.D.	Composition
1	NOBLES	Noble Gases and Radioactively Inert Aerosols [b]
2	CSI	CsI + RbI
3	TEO2	TeO_2
4	SRO	SrO
5	MOO2	MoO_2
6	CSOH	CsOH + RbOH
7	BAO	BaO
8	LA2O3	$La_2O_3 + Pr_2O_3 + Nd_2O_3 + Sm_2O_3 + Y_2O_3$
9	CEO3	CeO_2
10	SB	Sb
11	TE2	Te_2
12	UO2	$UO_2 + NpO_2 + PuO_2$

Notes: see following pages

Notes

(a) Explanation of Species

Specie (1): The Specie (1) vapors represent the noble gases. the Specie (1) aerosols are used to represent all non-radioactive aerosols (except for water droplets which are tracked separately in the thermal-hydraulic routines). The aerosol and deposited masses represent the core structural materials along with any concrete aerosols generated ex-vessel. The Specie (1) solid aerosols are assumed to have negligible vapor pressure at the temperatures of interest except in the core or core debris. The vapor pressure assumption used in the core and core debris are discussed in the write-ups for subroutines FPRATP and METOXA.

Specie (2): This specie represents the compounds CsI and RbI. All of the iodine is assumed to combine with the alkali fission products since the molar ratio is about 10 to 1 in favor of cesium and rubidium. Due to the dominance of cesium, CsI properties are chosen.

Specie (3): This specie represents tellurium that is oxidized to TeO_2. Tellurium released in-core is assumed to form TeO_2 directly. Tellurium released ex-vessel is assumed to be elemental; it is allowed to oxidize to TeO_2 in the cavity if steam or oxygen are present (see subroutine METOXA).

Specie (4): Strontium is primarily released in elemental form ex-vessel and is assumed to oxidize to SrO in Containment. In-vessel release is also assumed to lead to SrO formation.

Specie (5): This specie is MoO_2. This chemical state is assumed since molybdenum is thought to be mainly released during concrete attack.

Specie (6): This specie includes CsOH and RbOH. It represents any cesium and rubidium that is left over after combination with iodine.

Specie (7): This specie is BaO. Barium behaves similarly to strontium due to its chemical periodicity.

Specie (8): This specie represents the lanthanides. All oxides in the lanthanide series are grouped together due

to similar chemical behavior. These are rather nonvolatile, but in-vessel release is allowed. They are believed primarily to be released ex-vessel as monoxides, which are further oxidized in Containment.

Specie (9): Cerium behavior is similar to lanthanide behavior but stoichiometry and vapor pressure differ enough to warrant a separate group.

Specie (10): Antimony is released in-vessel and ex-vessel in elemental form.

Specie (11): Tellurium released ex-vessel which doesn't oxidize in the cavity remains "frozen" as Te_2.

Specie (12): Uranium and the transuranics are grouped separately from the other fission products such as cesium because of their different radiological characteristics. These are only released ex-vessel, and are assumed to oxidize (or reduce) to the dioxide form in Containment.

(b)

A similar scheme is used to track structural materials in the core and core debris. As described above, all such materials are tracked as Specie (1) "fission product" aerosols after they are released. The structural materials which are accounted for in the PWR code are:

a. Group (1): cadmium
b. Group (2): indium
c. Group (3): silver
d. Group (4): tin
e. Group (5): manganese

TABLE 4.7.3-6
COMPARISON OF REACTOR SAFETY STUDY AND RISK ASSESSMENT RESULTS FOR BLACKOUT WITH EARLY CONTAINMENT FAILURE AND INTERFACING SYSTEM LOCA SEQUENCES

Release Fractions [1]

Sequence	Noble Gases	Iodine (CSI)	Cesium (CSOH)	Tellurium (TE2)	Strontium (SRO)	Lanthanium (LA2O3)	Barium (BAO)	Cerium (CEO2)
Wash-1400 PWR2	0.9	0.7	0.5	0.3	0.06	4E-3		
NUREG-0956 SBO Early Overpres-surization	1.0	0.2	0.2	0.1	0.02	2E-4		
This study STC #8 SBO with Early OP	0.94	0.1	0.15	0.017	0.02	2E-5		
NUREG- Interfacing System LOCA (V) w/o water	1.0	0.3	0.3	0.06	0.005	3E-4	4E-3	4E-4
This study STC #23 (V w/o submergence)	0.94	0.3	0.3	1.6E-5	0.23	4E-4	2E-1	4E-2

Note: (1) Headings in parentheses are MAAP Species used for release fraction. Main headings are surrogates as defined in Table 4.16 of NUREG-0956

TABLE 4.7.3-7
SEQUENCE RELEASE COMPARISON

(a) Core Inventory Fraction Released to Environment

Group/Specie	Surry HINY-NXY During In-Vessel Release	MAAP SGTR (RV Failure)
I	0.502	0.449
CS	0.490	0.432
TE	0.159	0.55 (1)
SR	2.6E-4	2.0E-2
LA	4.1E-8	5E-5
BA	4.8E-3	1.8E-2
CE	0	5.5E-3
RU	4.7E-7	
MO		0.138
SB		0.477

(b) Fission Product Location at Vessel Failure (%)

Group	Location	HINY-NXY	MAAP-SGTR
CSI	Fuel	7.5	1
	RCS	37	32
	SG Sec	5.5	22
CsOH	Fuel	7.2	1
	RCS	38	32
	SG Sec	5.8	24
Te	Fuel	39	12
	RCS	41	17
	SG Sec	3.9	16

(c) Timing of Key Events

	HINY-NXY	MAAP-SGTR
Reactor Pumps off,s	35640	27080
Core Uncovery,s	47780	67122
Core Slump,s	52450	-
Support Plate Failure,s	-	78830
RV Failure,s	56020	79330

Note: (1) With MAAP option Te released in-vessel enabled.

TABLE 4.7.3-8
TIMING* FOR CSI RELEASE

STC	MAAP Case No.	Release In Vessel Start	Release In Vessel Duration	Release Ex Vessel Start	Release Ex Vessel Duration
2	31	0.3E4	1.0E5	0.3E4	1.0E5
5	36	2.1E4	3.5E4	2.1E4	3.5E4
7	37	1.1E4	1.0E5	2.0E4	1.0E5
8	29	1.4E4	1.0E5	2.0E4	1.0E5
11	38	0.3E5	3.5E5	0.3E5	3.5E5
13	35	0.3E5	3.0E5	0.3E5	3.0E5
15	28	1.2E5	2.5E5	1.2E5	2.5E5
21	26	0.3E4	4.0E4	0.3E4	4.0E4
23	33	2.0E4	6.5E4	2.0E4	6.5E4
24	40	7.3E4	1.0E5	7.7E4	1.0E5

* All units are in seconds

TABLE 4.7.3-9
RADIONUCLIDE HEAT OF RELEASE

STC	MAAP Case No.	Fraction of Decay Heat Released in Fission Products
2	31	0.322
5	36	0.305
7	37	0.339
8	29	0.321
11	38	0.359
13	35	0.324
15	28	0.326
21	26	0.063
23	33	0.338
24	40	0.237

TABLE 4.7.4-1
FREQUENCY-RANKED SOURCE TERM CATEGORIES

Rank		Frequency	Percent of 6.80E-5
1	STC 1	4.320E-5	63.55
2	STC 24	7.376E-6	10.85
3	STC 20	7.063E-6	10.39
4	STC 14	2.588E-6	3.81
5	STC 9	2.463E-6	3.62
6	STC 15	1.613E-6	2.37
7	STC 22	1.361E-6	2.00
8	STC 16	7.418E-7	1.09
9	STC 2	5.535E-7	0.81
10	STC 23	2.401E-7	0.35
11	STC 7	1.603E-7	0.24
12	STC 10	1.501E-7	0.22
13	STC 19	1.445E-7	0.21
14	STC 11	1.177E-7	0.17
15	STC 17	8.876E-8	0.13
16	STC 8	3.720E-8	0.05
17	STC 18	3.143E-8	0.05
18	STC 21	2.959E-8	0.04
19	STC 3	3.929E-9	0.01
20	STC 12	3.562E-9	0.01
21	STC 5	2.622E-9	0.00
22	STC 13	2.160E-9	0.00
23	STC 6	1.150E-9	0.00
24	STC 4	9.081E-10	0.00

TABLE 4.7.4-2
INITIATING EVENT CONTRIBUTION TO THE SOURCE TERM CATEGORIES

Source Term Category: 1	4.320E-5	
	Amount	Percent
S2	9.683E-6	22.415
S1	6.391E-6	14.795
T1A	5.694E-6	13.181
T8	5.274E-6	12.209
T1Tr	4.913E-6	11.373
T1	3.770E-6	8.728
T9ATr	2.847E-6	6.590
A	1.451E-6	3.359
T2	7.777E-7	1.800
T3Tr	7.439E-7	1.722
TH	4.007E-7	0.927
T9B	3.902E-7	0.903
T2ATr	2.990E-7	0.692
T9A	2.861E-7	0.662
T3	5.511E-8	0.128
T2A	4.900E-8	0.113
T9BTr	4.395E-8	0.102
T5A	3.705E-8	0.086
T5B	3.572E-8	0.083
T2Tr	2.495E-8	0.058
RX	1.259E-8	0.029
T4	1.038E-8	0.024

Source Term Category: 2	5.535E-7	
	Amount	Percent
T1A	1.380E-7	24.933
T8	9.708E-8	17.540
S2	7.857E-8	14.195
S1	5.205E-8	9.403
T9ATr	4.674E-8	8.445
T1Tr	4.058E-8	7.331
T3Tr	3.781E-8	6.832
T1	3.017E-8	5.452
T2ATr	1.534E-8	2.772
T2	6.250E-9	1.129

Source Term Category: 3	3.929E-9	
	Amount	Percent
T3Tr	2.335E-9	59.422
T2ATr	9.512E-10	24.211
T2ATr	9.512E-10	24.211
T8	1.975E-10	5.026
T9ATr	1.008E-10	2.565
T2Tr	8.648E-11	2.201
T1Tr	7.547E-11	1.921

TABLE 4.7.4-2 (Continued)
INITIATING EVENT CONTRIBUTION TO THE SOURCE TERM CATEGORIES

T1A	5.978E-11	1.522
S1	4.305E-11	1.096
S2	4.221E-11	1.074
Source Term Category: 4	**9.081E-10**	
	Amount	**Percent**
T3Tr	5.396E-10	59.422
T2ATr	2.199E-10	24.211
T8	4.565E-11	5.027
T9ATr	2.329E-11	2.565
T2Tr	1.999E-11	2.201
T1Tr	1.744E-11	1.921
T1A	1.382E-11	1.522
S1	9.950E-12	1.096
S2	9.756E-12	1.074
Source Term Category: 5	**2.622E-9**	
	Amount	**Percent**
T1A	1.125E-9	42.896
T8	7.879E-10	30.048
T9ATr	3.372E-10	12.861
T3Tr	2.550E-10	9.724
T2ATr	1.034E-10	3.943
Source Term Category: 6	**1.150E-9**	
	Amount	**Percent**
T1A	4.933E-10	42.899
T8	3.455E-10	30.045
T9ATr	1.479E-10	12.860
T3Tr	1.118E-10	9.724
T2ATr	4.535E-11	3.943
Source Term Category: 7	**1.603E-7**	
	Amount	**Percent**
T1Tr	1.154E-7	71.975
T1	1.367E-8	8.526
T8	9.321E-9	5.815
S2	6.145E-9	3.833
T3Tr	5.019E-9	3.131
S1	3.968E-9	2.475
T9ATr	2.409E-9	1.503
T2ATr	2.039E-9	1.272
Source Term Category: 8	**3.720E-8**	
	Amount	**Percent**
T1Tr	2.687E-8	72.235
T1	3.160E-9	8.495

TABLE 4.7.4-2 (Continued)
INITIATING EVENT CONTRIBUTION TO THE SOURCE TERM CATEGORIES

T8	2.177E-9	5.853
S2	1.364E-9	3.666
T3Tr	1.176E-9	3.161
S1	8.800E-10	2.366
T9ATr	5.588E-10	1.502
T2ATr	4.777E-10	1.284
Source Term Category: 9	**2.463E-6**	
	Amount	**Percent**
T3Tr	1.553E-6	63.035
T2ATr	6.325E-7	25.680
T8	8.252E-8	3.350
T2Tr	5.749E-8	2.334
T9ATr	4.599E-8	1.867
T1A	4.162E-8	1.690
T9BTr	1.712E-8	0.695
T1Tr	1.411E-8	0.573
Source Term Category: 10	**1.501E-7**	
	Amount	**Percent**
T3Tr	9.780E-8	65.154
T2ATr	3.984E-8	26.544
T2Tr	3.622E-9	2.413
T8	2.890E-9	1.925
T9ATr	1.940E-9	1.293
Source Term Category: 11	**1.177E-7**	
	Amount	**Percent**
T3Tr	7.666E-8	65.128
T2ATr	3.123E-8	26.534
T2Tr	2.839E-9	2.412
T8	2.282E-9	1.939
T9ATr	1.528E-9	1.298
Source Term Category: 12	**3.562E-9**	
	Amount	**Percent**
T3Tr	2.409E-9	67.636
T2ATr	9.815E-10	27.555
T2Tr	8.924E-11	2.505
Source Term Category: 13	**2.160E-9**	
	Amount	**Percent**
T3Tr	1.461E-9	67.636
T2ATr	5.952E-10	27.555
T2Tr	5.411E-11	2.505

TABLE 4.7.4-2 (Continued)
INITIATING EVENT CONTRIBUTION TO THE SOURCE TERM CATEGORIES

Source Term Category: 14	2.588E-6	
	Amount	**Percent**
T1Tr	1.319E-6	50.968
T8	3.891E-7	15.035
T1	2.574E-7	9.946
T3Tr	1.863E-7	7.198
S2	1.212E-7	4.684
T9ATr	9.106E-8	3.518
S1	7.903E-8	3.054
T2ATr	7.550E-8	2.917
T9B	1.540E-8	0.595
T1A	1.334E-8	0.515

Source Term Category: 15	1.613E-6	
	Amount	**Percent**
T1Tr	8.221E-7	50.966
T8	2.425E-7	15.035
T1	1.604E-7	9.945
T3Tr	1.161E-7	7.200
S2	7.555E-8	4.684
T9ATr	5.675E-8	3.518
S1	4.925E-8	3.054
T2ATr	4.707E-8	2.918

Source Term Category: 16	7.418E-7	
	Amount	**Percent**
T8	2.412E-7	32.516
T1A	2.160E-7	29.120
T9ATr	1.016E-7	13.693
A	9.262E-8	12.486
S2	2.232E-8	3.009
S1	1.473E-8	1.986
T1Tr	1.138E-8	1.534
T1	1.101E-8	1.484
TH	9.426E-9	1.271
T3Tr	8.942E-9	1.206

Source Term Category: 17	8.876E-8	
	Amount	**Percent**
S2	2.708E-8	30.511
S1	2.394E-8	26.968
IS	2.137E-8	24.081
A	1.211E-8	13.643
T7	3.098E-9	3.491

TABLE 4.7.4-2 (Continued)
INITIATING EVENT CONTRIBUTION TO THE SOURCE TERM CATEGORIES

Source Term Category: 18	3.143E-8	
	Amount	**Percent**
IS	9.502E-9	30.231
S2	8.571E-9	27.271
S1	7.504E-9	23.874
A	4.417E-9	14.054
T7	1.037E-9	3.298

Source Term Category: 19	1.445E-7	
	Amount	**Percent**
T8	3.801E-8	26.305
T1A	3.092E-8	21.400
T9ATr	1.561E-8	10.800
T3Tr	1.380E-8	9.553
A	1.244E-8	8.612
S2	8.002E-9	5.538
T1Tr	5.824E-9	4.031
T2ATr	5.598E-9	3.874
S1	5.283E-9	3.656
T1	4.684E-9	3.241

Source Term Category: 20	7.063E-6	
	Amount	**Percent**
A	2.516E-6	35.626
T1A	1.714E-6	24.267
T3Tr	1.169E-6	16.552
T2ATr	4.744E-7	6.717
T1	3.345E-7	4.736
RX	2.544E-7	3.602
T9B	1.535E-7	2.173
T9A	1.114E-7	1.578
T2	8.425E-8	1.193
T8	6.694E-8	0.948
T5B	6.071E-8	0.860
T5A	6.066E-8	0.859
T2Tr	4.243E-8	0.601
T3	1.247E-8	0.177

Source Term Category: 21	2.959E-8	
	Amount	**Percent**
S2	9.028E-9	30.511
S1	7.980E-9	26.968
IS	7.126E-9	24.081
A	4.037E-9	13.643
T7	1.033E-9	3.491

TABLE 4.7.4-2 (Continued)
INITIATING EVENT CONTRIBUTION TO THE SOURCE TERM CATEGORIES

Source Term Category: 22	1.361E-6	
	Amount	Percent
VX	1.361E-6	00.000

Source Term Category: 23	2.401E-7	
	Amount	Percent
VX	2.401E-7	00.000

Source Term Category: 24	7.376E-6	
	Amount	Percent
T7	7.011E-6	95.046
T1A	1.163E-7	1.577
T8	1.143E-7	1.549
T9ATr	4.528E-8	0.614
T3Tr	4.003E-8	0.543
T2ATr	1.623E-8	0.220
T1	1.282E-8	0.174

TABLE 4.8-1
LEVEL 2 SENSITIVITY ANALYSIS

Case(s)	Parameter Varied	Variation(s)	Result Investigated
A1,A2	Induced Hot Leg Failure Probability	(1,0)	Containment Failure Times, Debris Cooled In-vessel
B1,B2	Alpha Mode Containment Failure Probability	(x10,x.1)	Alpha - mode Containment Failure
C1,C2	Probability of In-vessel Debris Cooling for LBLOCA without Accumulators	(1,0)	Debris Cooled In-vessel
D1,D2	Probability of In-vessel Debris Cooling for Late Depressurization Sequences	(1,0)	Debris Cooled In-vessel
E1,E2	Probability of In-vessel Debris Cooling for Power Recovery after Core Damage	(1,0)	Debris Cooled In-vessel
F1,F2	Probability of In-vessel Debris Cooling (C, D, and E above combined)	(1,0)	Debris Cooled In-vessel
G1,G2	Amount of H_2 Produced In-vessel (Probability > 40% Core Zr oxidized)	(1,0)	Early Containment Failure
H1,H2	Probability of Large DCH Event	(x10,x.1)	Early Containment Failure

TABLE 4.8-1 (Continued)
LEVEL 2 SENSITIVITY ANALYSIS

Case(s)	Parameter Varied	Variation(s)	Result Investigated
I1,I2	Probability of Unconditional Hydrogen Burn (UCHB)	(1,0)	Early Containment Failure
J1,J2	Containment Failure Pressure	(to 5th & 50th percentile)	Containment Failure Time and Mode
K1	Probability of Early Recirculation (due to Energetic Events or environmental conditions) Spray Failure	(1,0)	Containment Failure Time, Time Recirc Sprays Operate
L1,L2	Probability of Debris Dispersal from Cavity	(1,0)	Containment Failure Time
M1,M2	Depth of Debris Pools in Cavity/Lower Containment (Probability of Deep/Shallow/very Shallow Pools)	(1,0)	Containment Failure Time
N1,N2	Probability of Debris Cooling Ex-vessel	(1,0)	Containment Failure Time
O1	Probability of Late Recirculation Spray Failure	(1,0)	Containment Failure Time, Time Recirc Sprays Operate

TABLE 4.8-1 (Continued)
LEVEL 2 SENSITIVITY ANALYSIS

Case(s)	Parameter Varied	Variation(s)	Result Investigated
P1,P2	Frequency of Event V (Plant Damage State)	(x10,x.1)	Event V Source Term Category (PDS)
Q1,Q2	Frequency of Loss of Containment Isolation (Plant Damage States)	(x10,.1)	Frequencies of Loss of Isolation PDS

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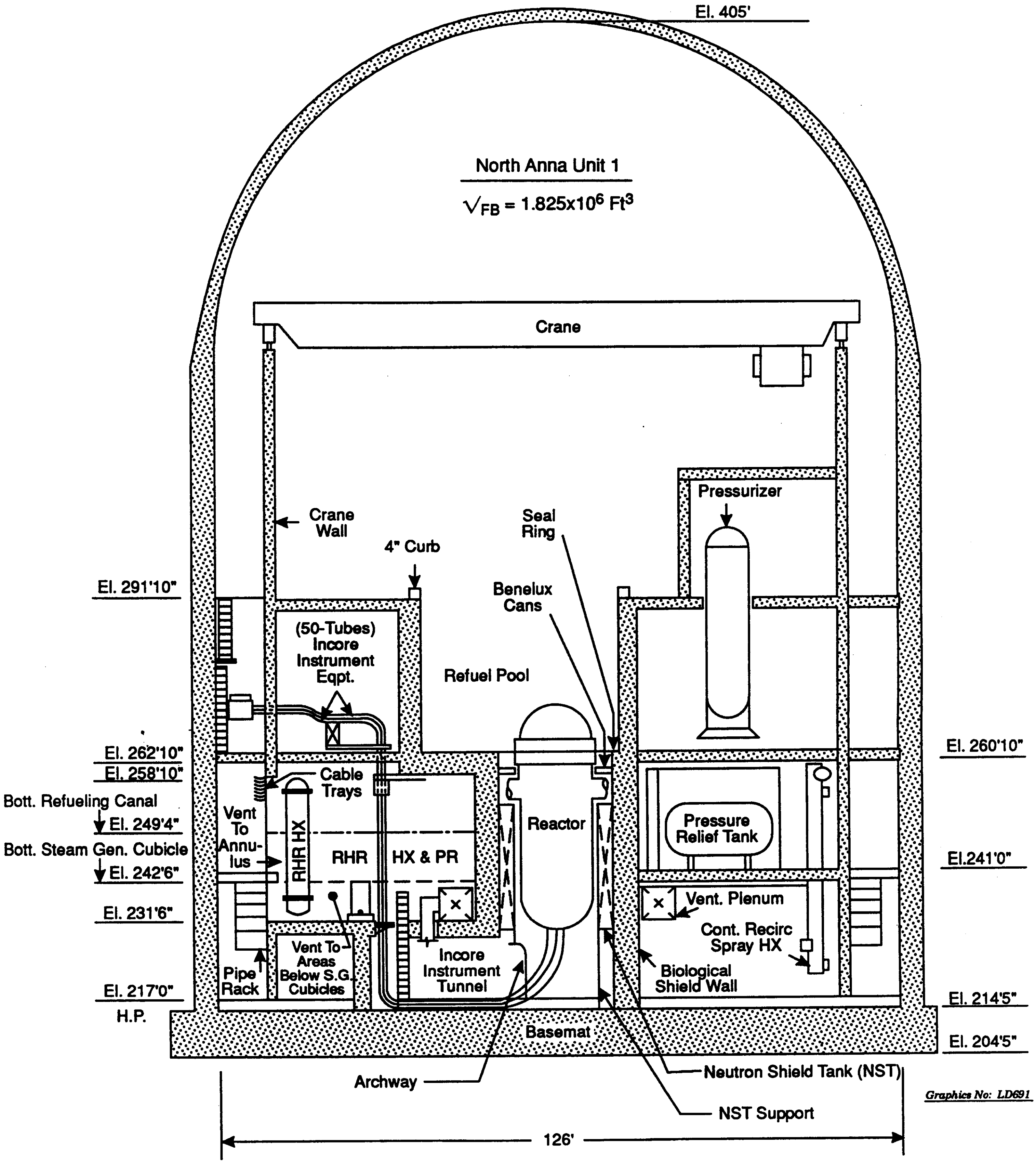
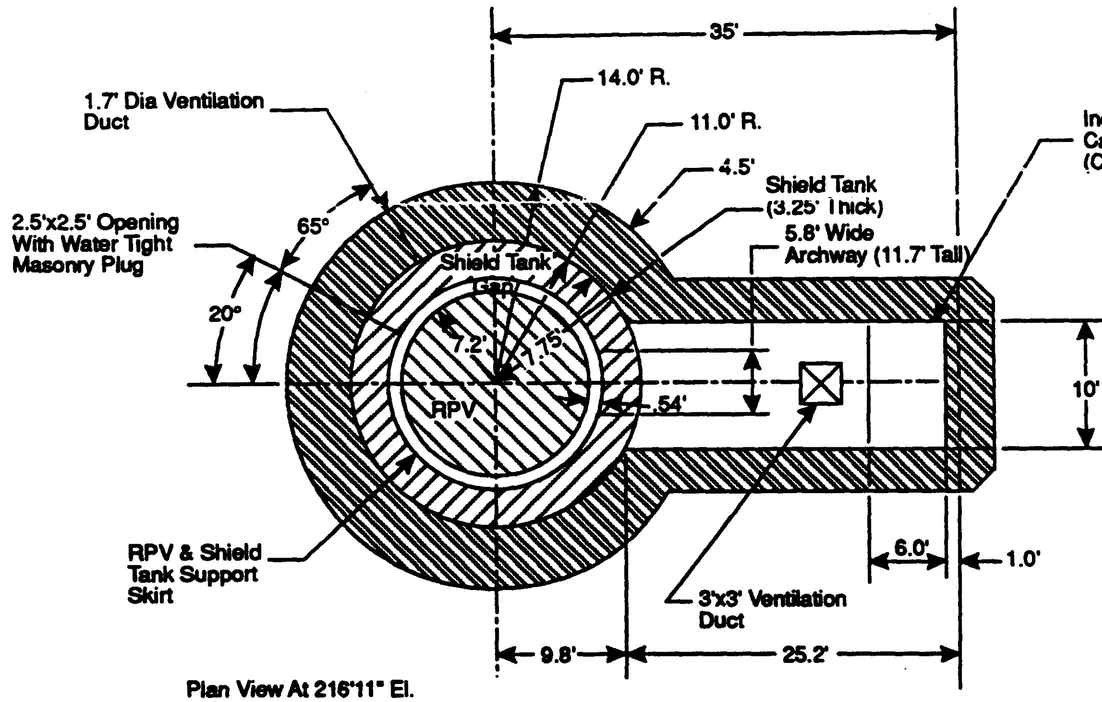
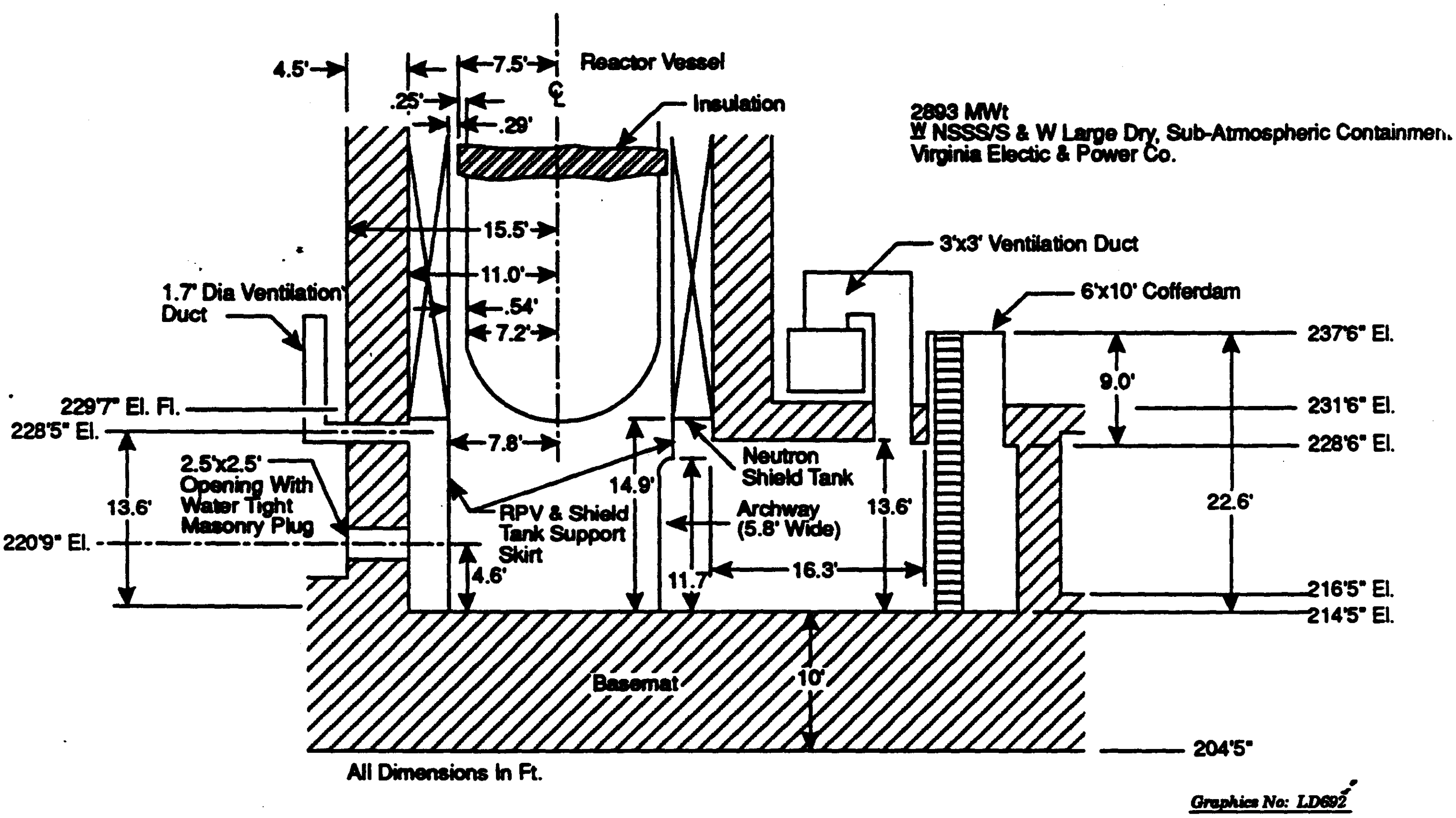


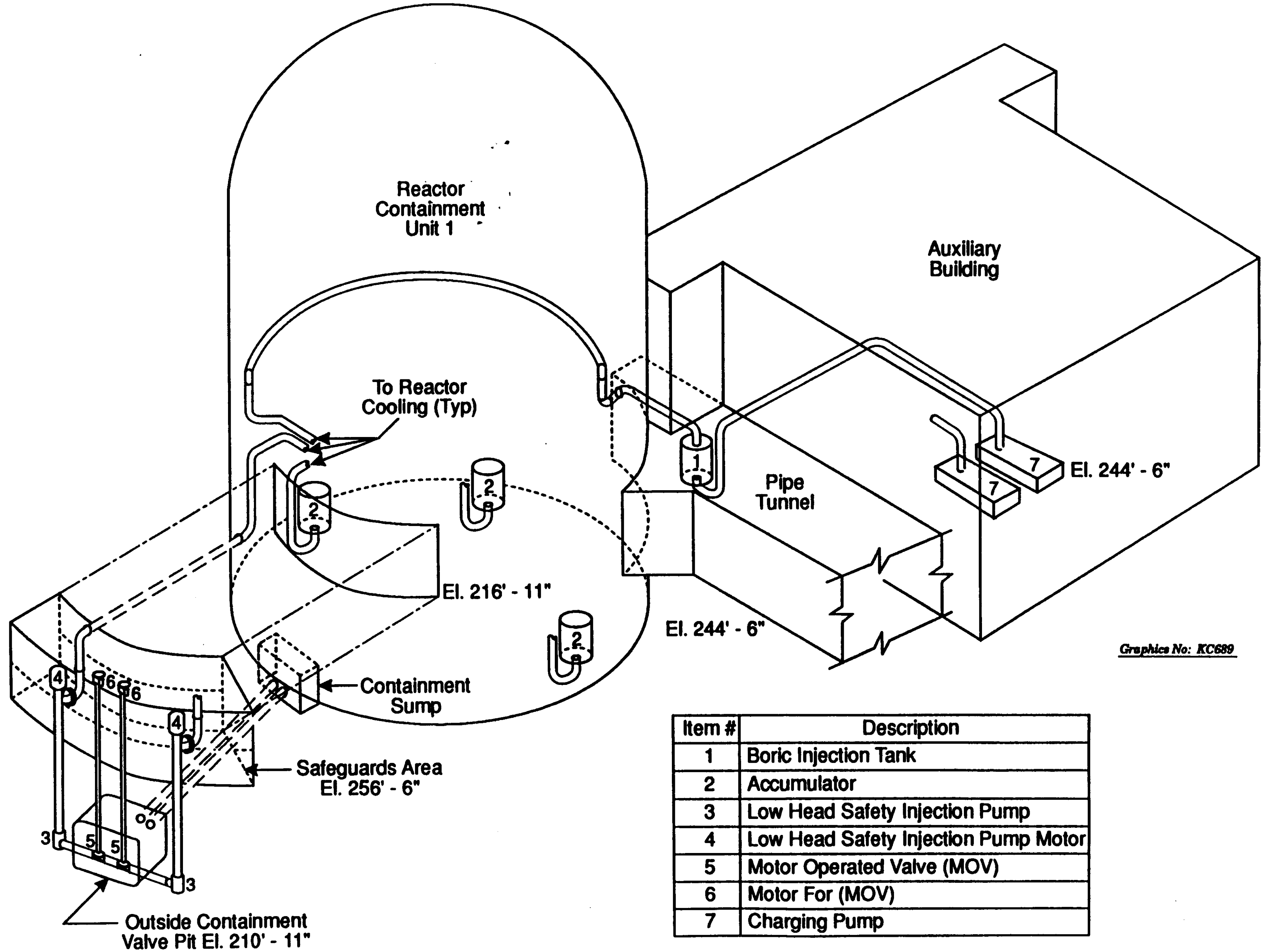
FIGURE 4.1.1-1
NORTH ANNA UNIT 1 CONTAINMENT VERTICAL CUTAWAY VIEW



35'
14.0' R.
1.7' Dia Ventilation Duct
11.0' R.
Incore Instrument Cable Opening (Cofferdam)
4.5'
Shield Tank (3.25' Thick)
2.5'x2.5' Opening With Water Tight Masonry Plug
65°
5.8' Wide Archway (11.7' Tall)
Shield Tank Gap
20°
7.2'
7.75'
10'
RPV
.54'
RPV & Shield Tank Support Skirt
6.0'
1.0'
3'x3' Ventilation Duct
9.8'
25.2'
Plan View At 216'11" El.

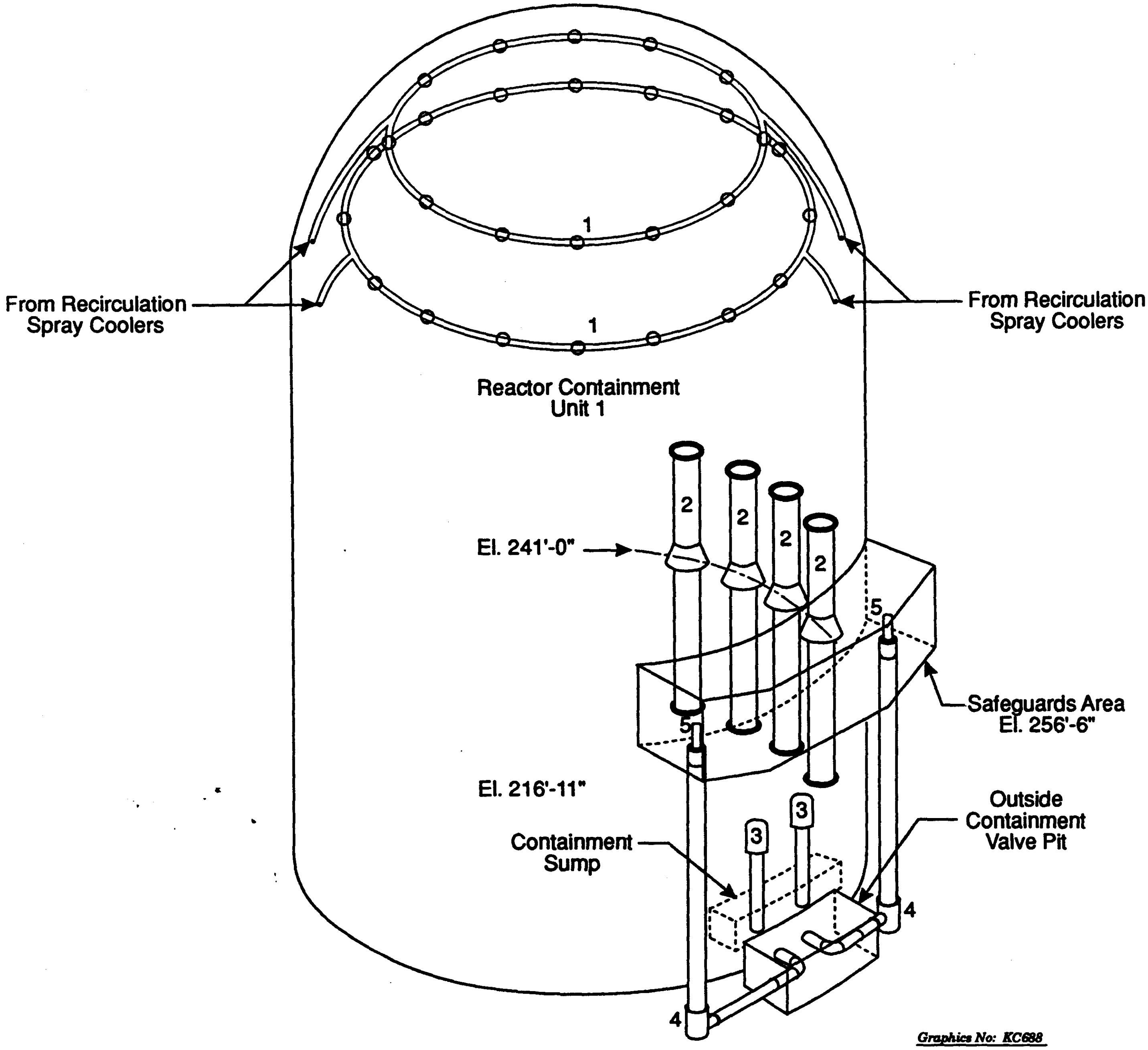


4.5'
7.5'
Reactor Vessel
.25'
℄
Insulation
.29'
2893 MWt
W NSSS/S & W Large Dry, Sub-Atmospheric Containment
Virginia Electic & Power Co.
15.5'
3'x3' Ventilation Duct
11.0'
1.7' Dia Ventilation Duct
6'x10' Cofferdam
.54'
7.2'
237'6" El.
9.0'
229'7" El. Fl.
231'6" El.
228'5" El.
7.8'
228'6" El.
2.5'x2.5' Opening With Water Tight Masonry Plug
Neutron Shield Tank
14.9'
13.6'
Archway (5.8' Wide)
13.6'
22.6'
RPV & Shield Tank Support Skirt
220'9" El.
4.6'
11.7'
16.3'
216'5" El.
214'5" El.
Basemat
10'
204'5"
All Dimensions In Ft.
Graphics No: LD692



Item #	Description
1	Boric Injection Tank
2	Accumulator
3	Low Head Safety Injection Pump
4	Low Head Safety Injection Pump Motor
5	Motor Operated Valve (MOV)
6	Motor For (MOV)
7	Charging Pump

FIGURE 4.1.3-1
DRAWING OF THE SI SYSTEM COMPONENT LOCATIONS



Item #	Description
1	Spray Headers
2	Recirculation Spray Cooler
3	Inside Recirculation Spray Pump
4	Outside Recirculation Spray Pump
5	Outside Recirculation Spray Pump Motor

FIGURE 4.1.3-2
DRAWING OF THE RECIRCULATION SPRAY SYSTEM COMPONENT LOCATIONS

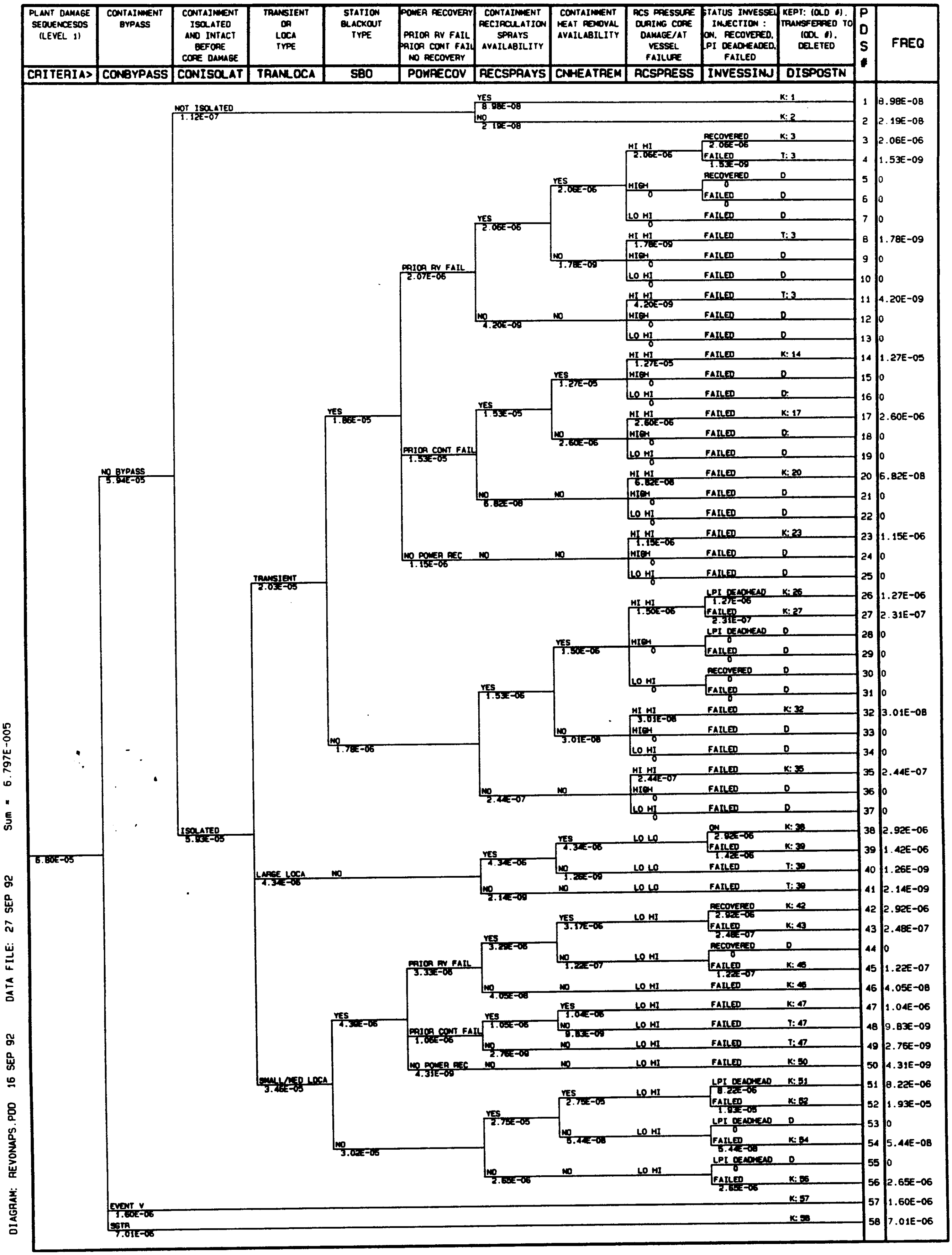


Figure 4.3.2-1 Initial Plant Damage State Logic Diagram

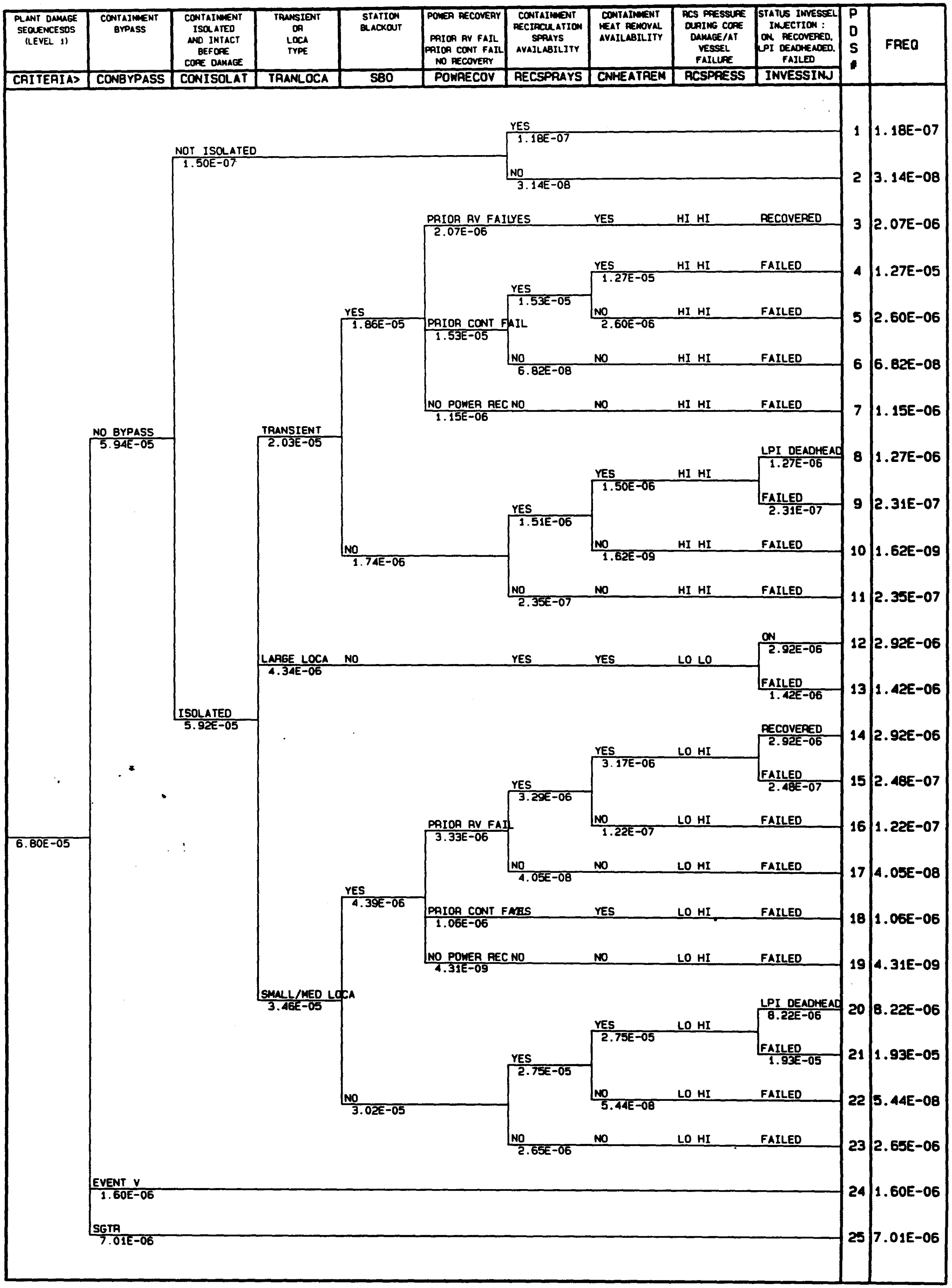


DIAGRAM: REV2NAPS.PDD 16 SEP 92 DATA FILE: 27 SEP 92 Sum = 6.797E-005

Figure 4.3.3-1 NAPS Plant Damage State Logic Diagram

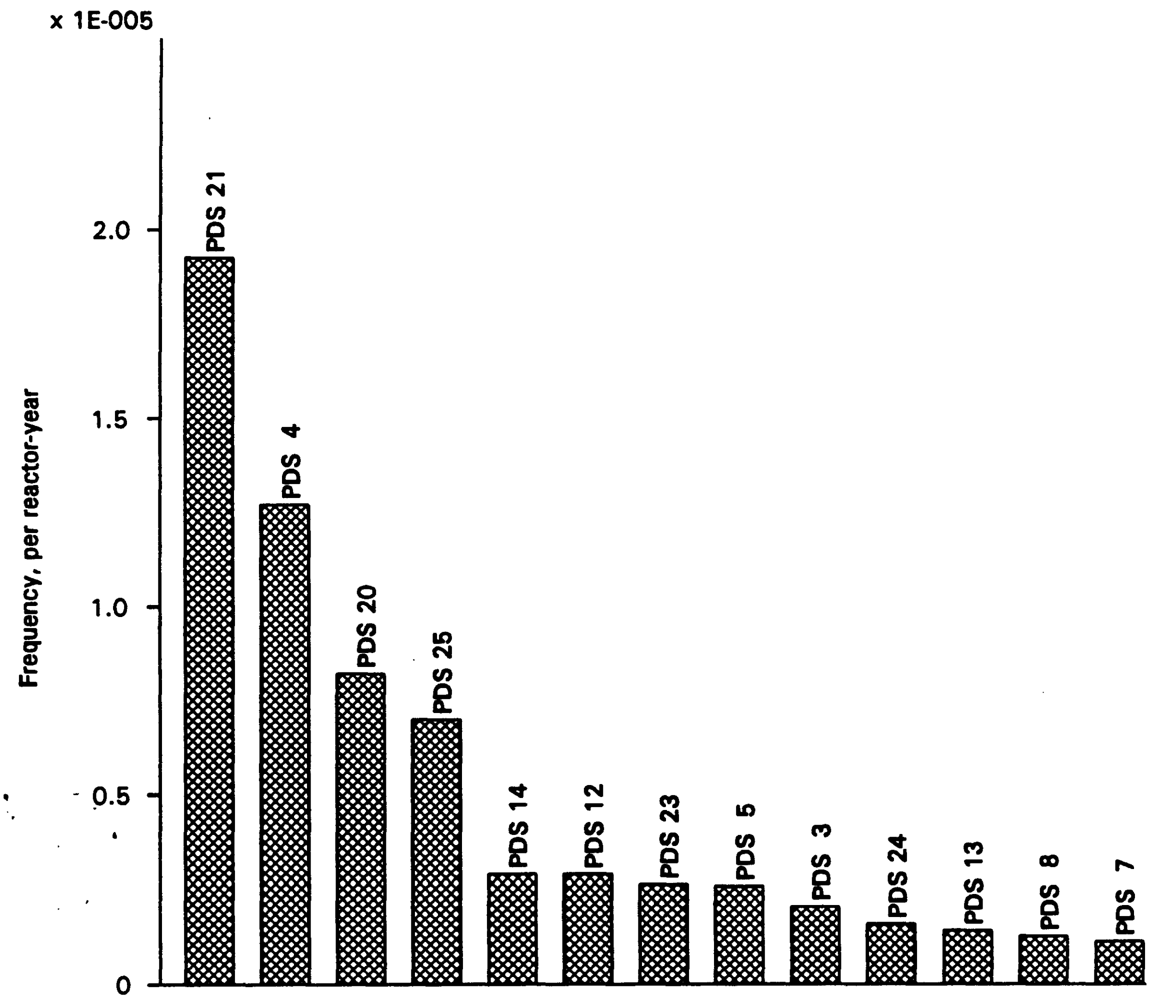


FIGURE 4.3.4-1 Plant Damage States ranked by Frequency

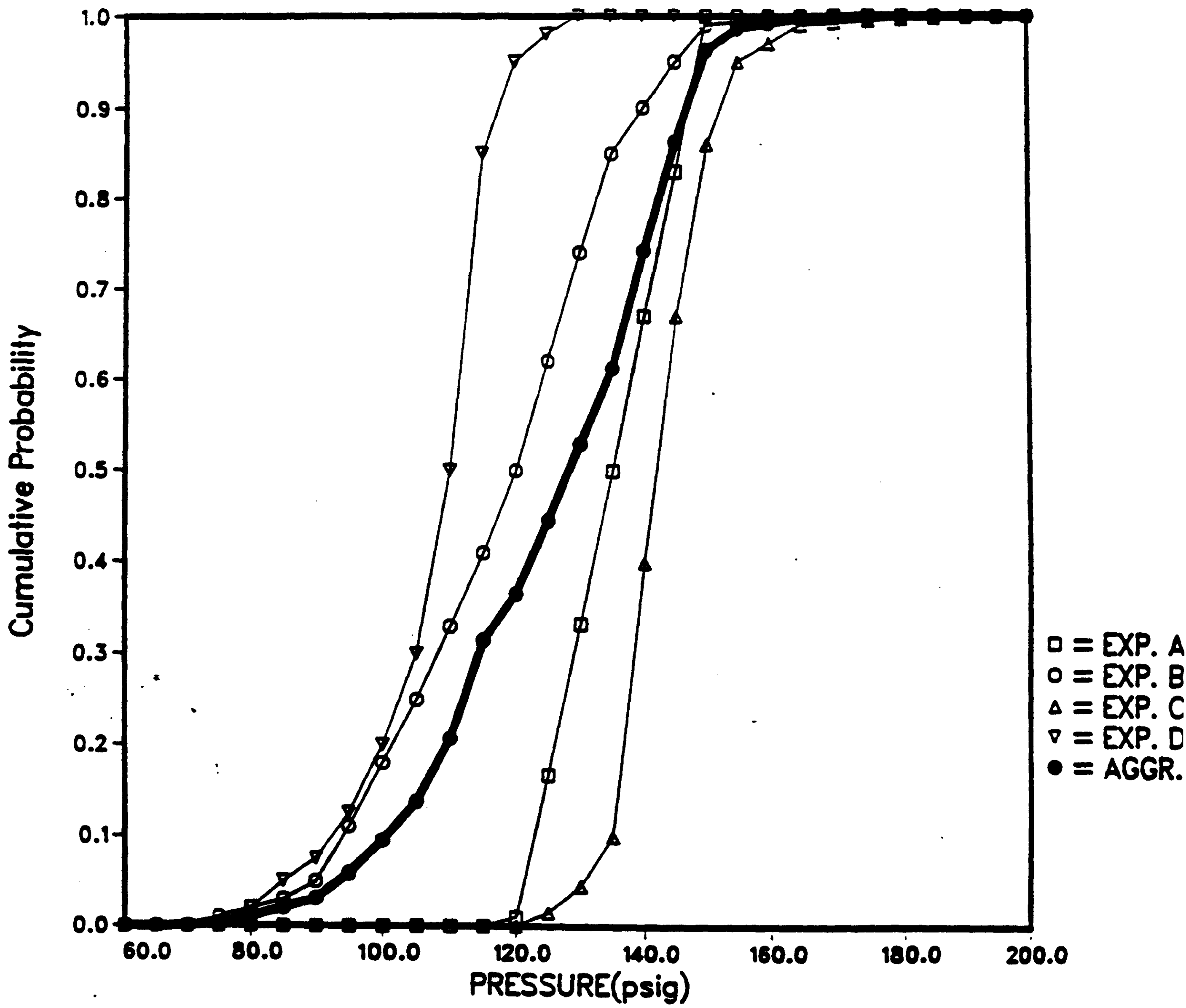


FIGURE 4.4.2-1 RESULTS OF EXPERT ELICITATION FOR STATIC FAILURE PRESSURE OF THE SURRY CONTAINMENT (FROM NUREG-1150)

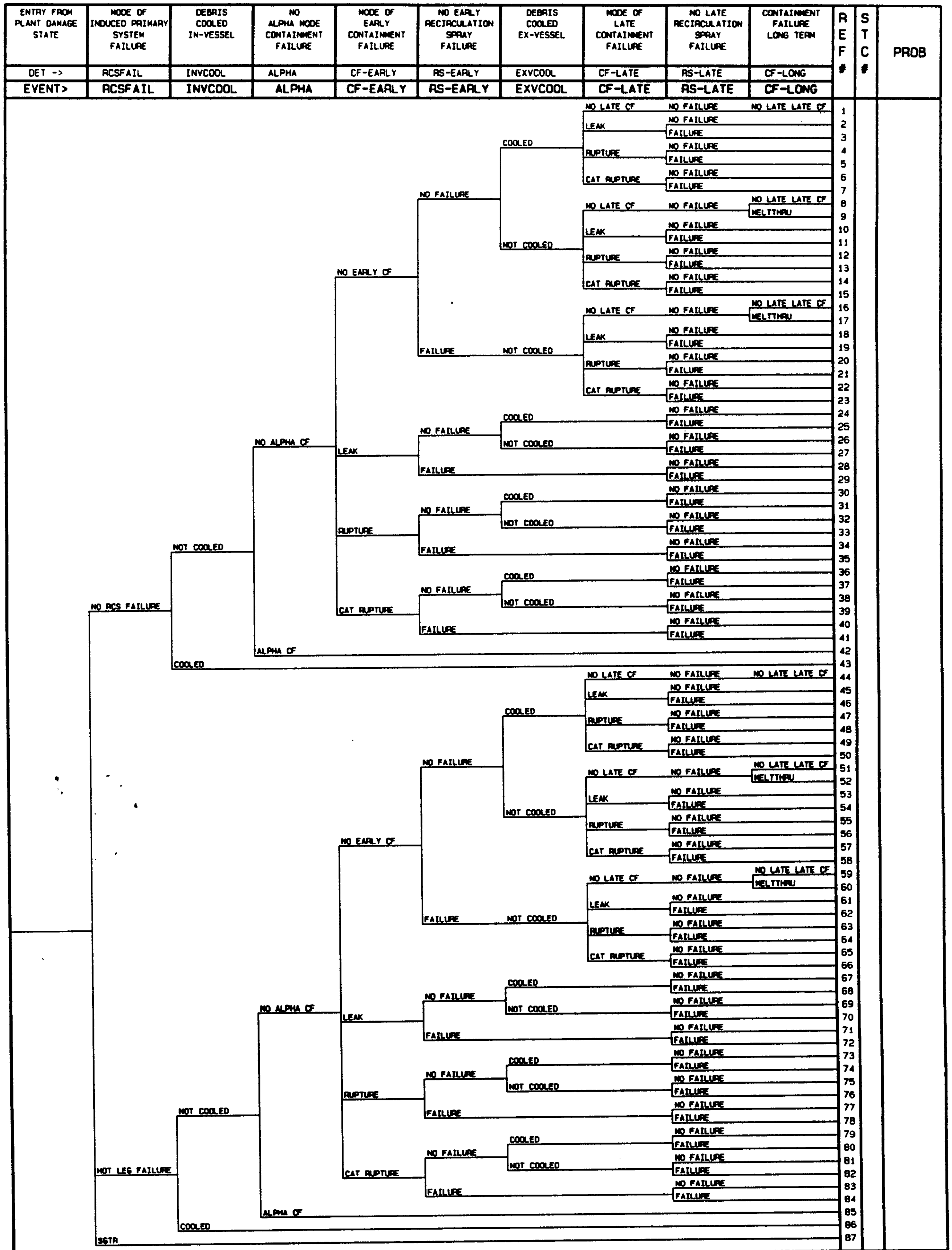


DIAGRAM: NAPSGEN .CET 19 FEB 92

FIGURE 4.5.1-1
LEVEL 2 GENERAL CONTAINMENT EVENT TREE

VIRGINIA ELECTRIC POWER COMPANY
NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
LEVEL 2
GENERAL CONTAINMENT EVENT TREE
REV. 0

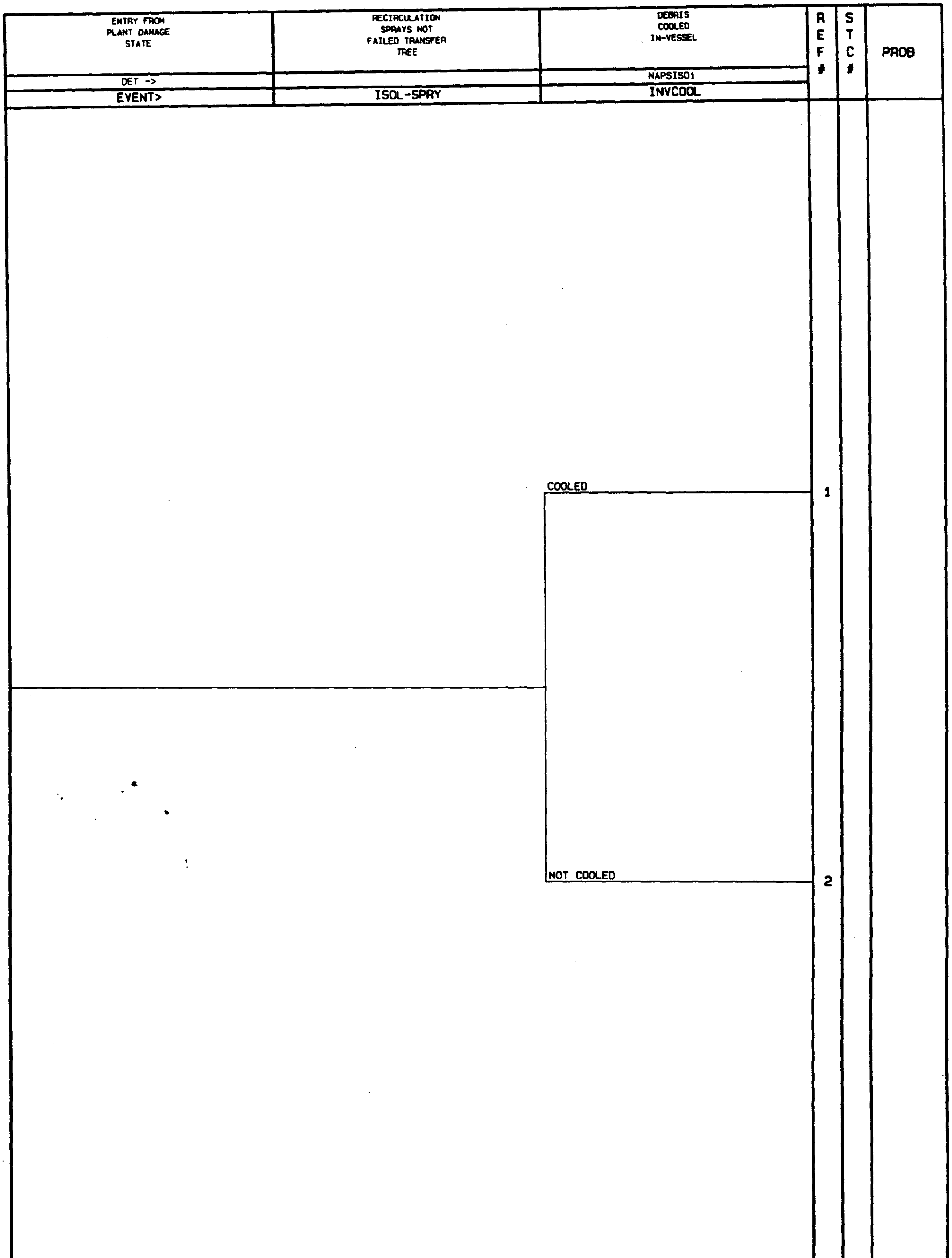


DIAGRAM: NAPSISO1.CET 7 JUL 92

FIGURE 4.5.1-2
LEVEL 2 CONTAINMENT EVENT TREE FOR LOSS OF ISOLATION WITH OPERABLE RECIRCULATION SPRAYS

VIRGINIA ELECTRIC POWER COMPANY NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
LEVEL 2 CONTAINMENT EVENT TREE FOR LOSS OF ISOLATION WITH OPERABLE RECIRCULATION SPRAYS

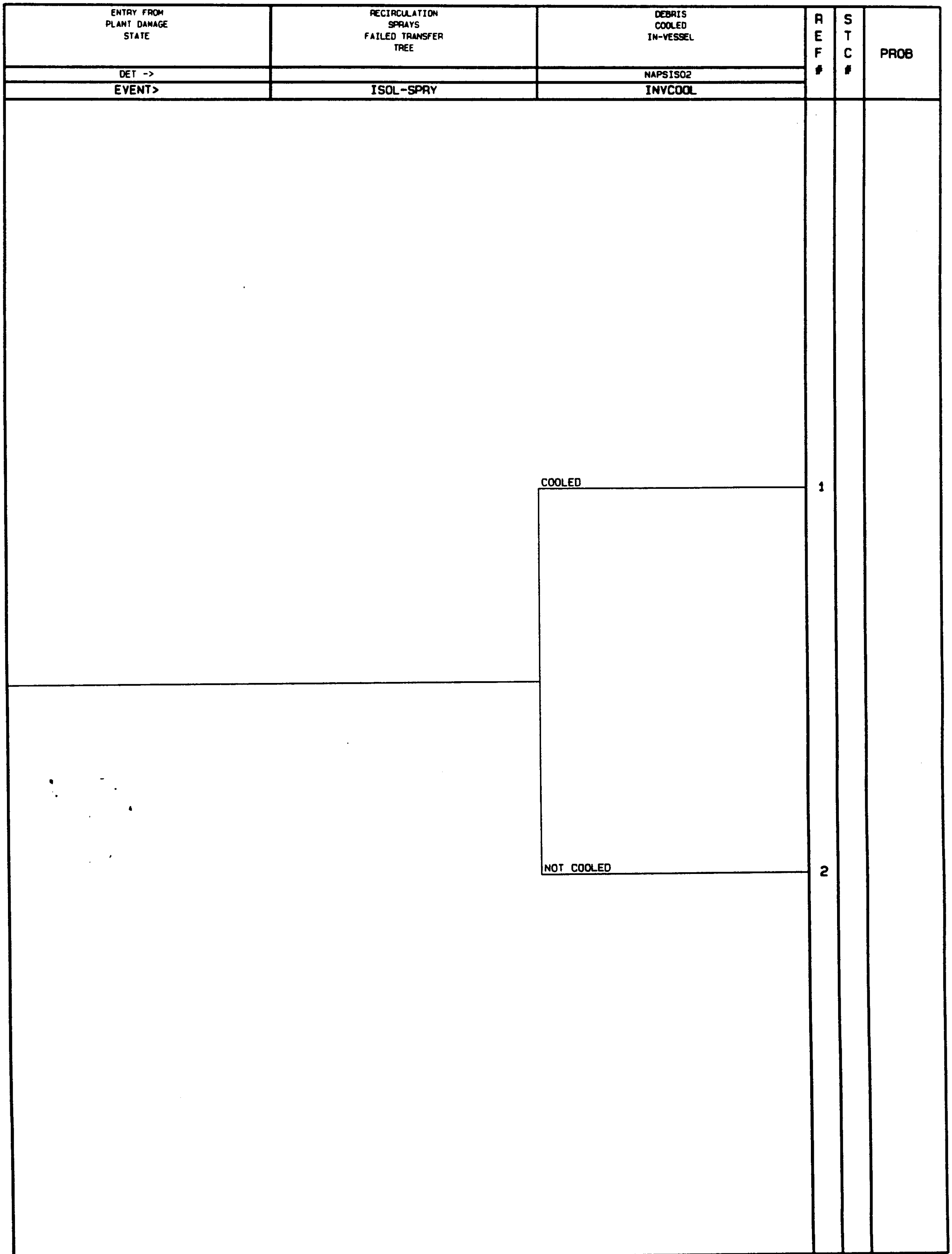


DIAGRAM: NAPSIS02.CET 7 JUL 92

FIGURE 4.5.1-3
LEVEL 2 CONTAINMENT EVENT TREE FOR LOSS OF ISOLATION WITH FAILED RECIRCULATION SPRAYS

VIRGINIA ELECTRIC POWER COMPANY
NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
LEVEL 2
CONTAINMENT EVENT TREE FOR
LOSS OF ISOLATION WITH FAILED RECIRCULATION SPRAYS

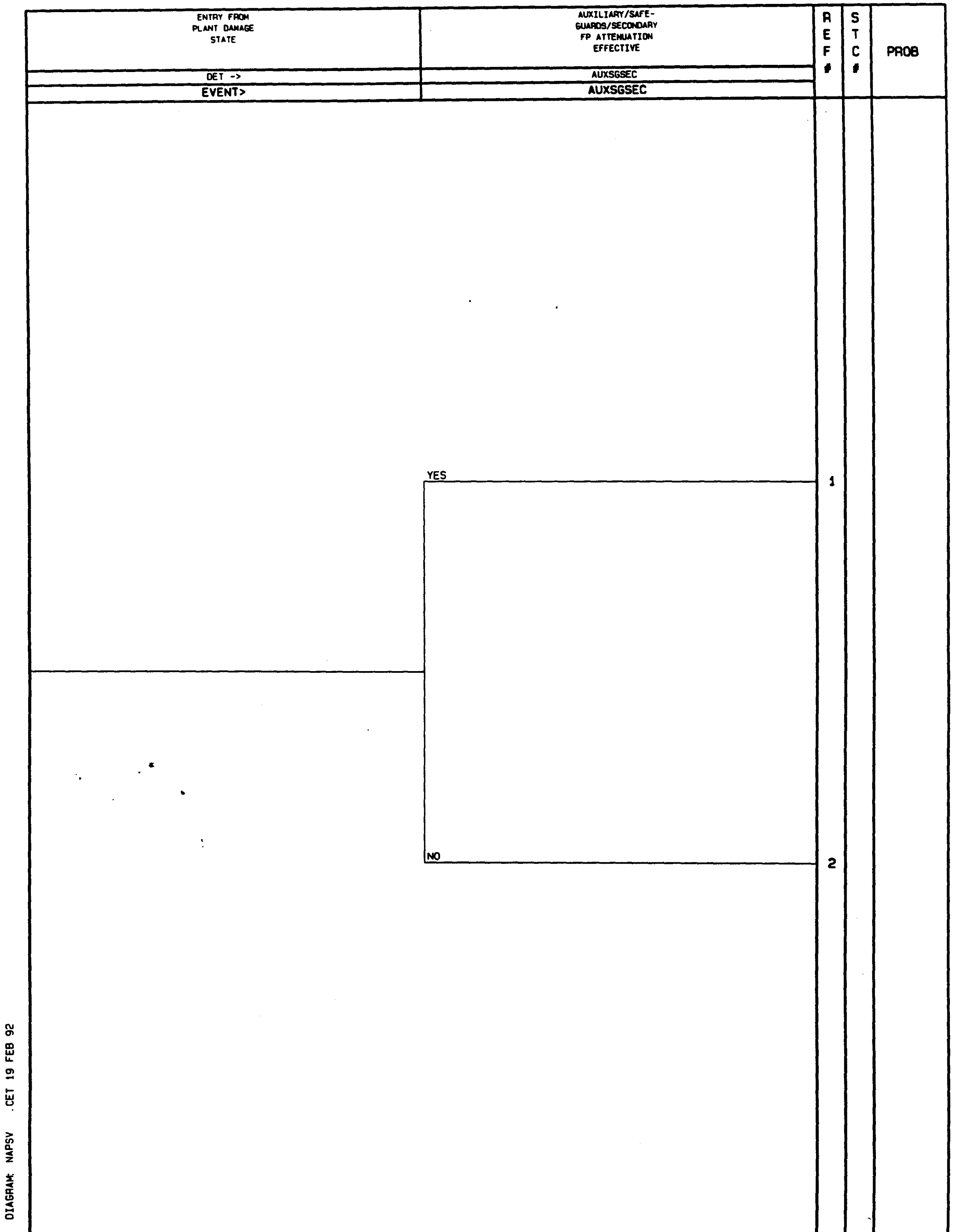


FIGURE 4.5.1-4
LEVEL 2 "CONTAINMENT" EVENT TREE FOR CONTAINMENT BYPASS EVENT V SEQUENCES

VIRGINIA ELECTRIC POWER COMPANY NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
LEVEL 2 "CONTAINMENT" EVENT TREE FOR CONTAINMENT BYPASS EVENT V SEQUENCES

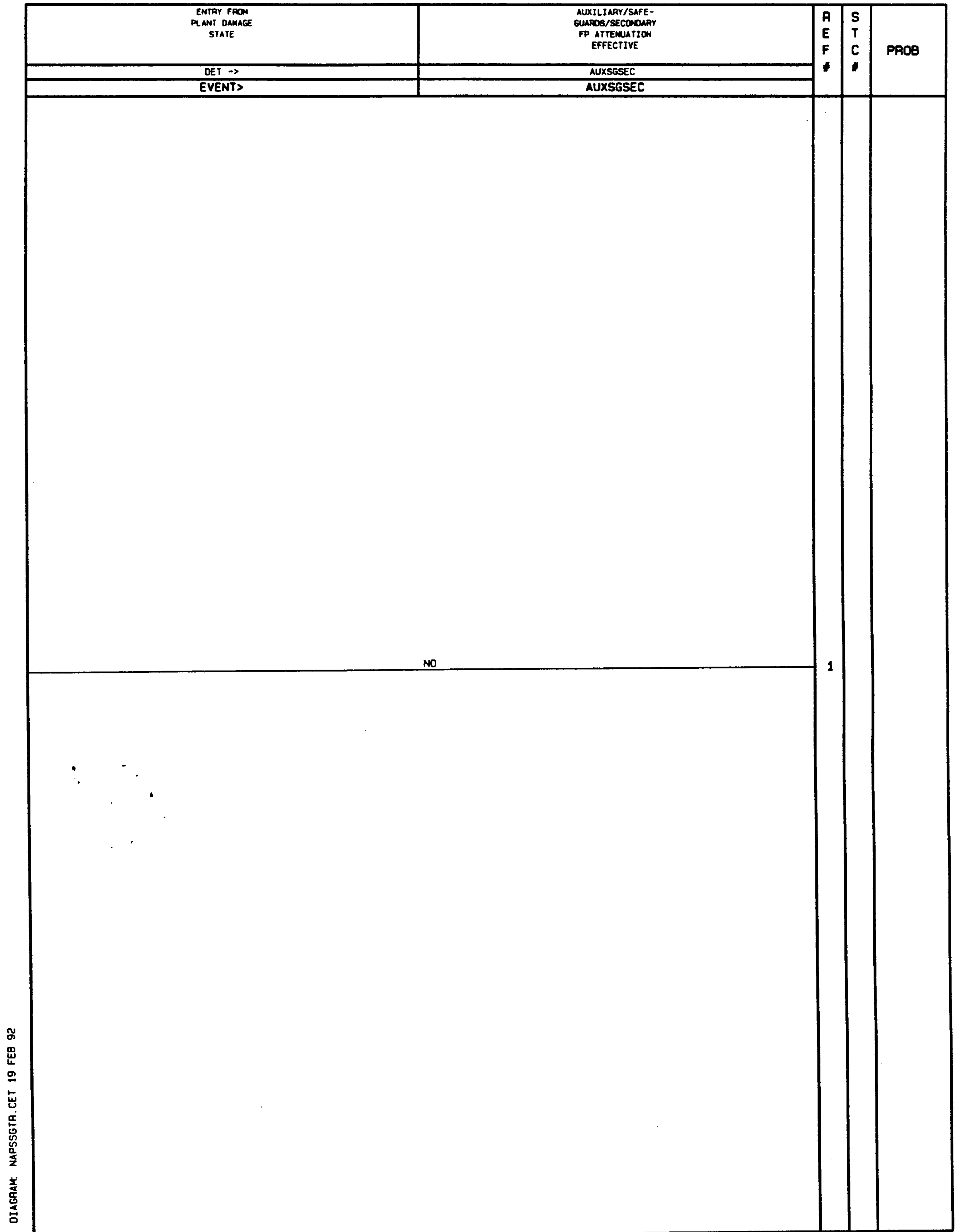


FIGURE 4.5.1-5
LEVEL 2 "CONTAINMENT" EVENT TREE
FOR STEAM GENERATOR TUBE RUPTURE
CONTAINMENT BYPASS SEQUENCES

VIRGINIA ELECTRIC POWER COMPANY NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
LEVEL 2 "CONTAINMENT" EVENT TREE FOR STEAM GENERATOR TUBE RUPTURE CONTAINMENT BYPASS SEQUENCES REV. 0

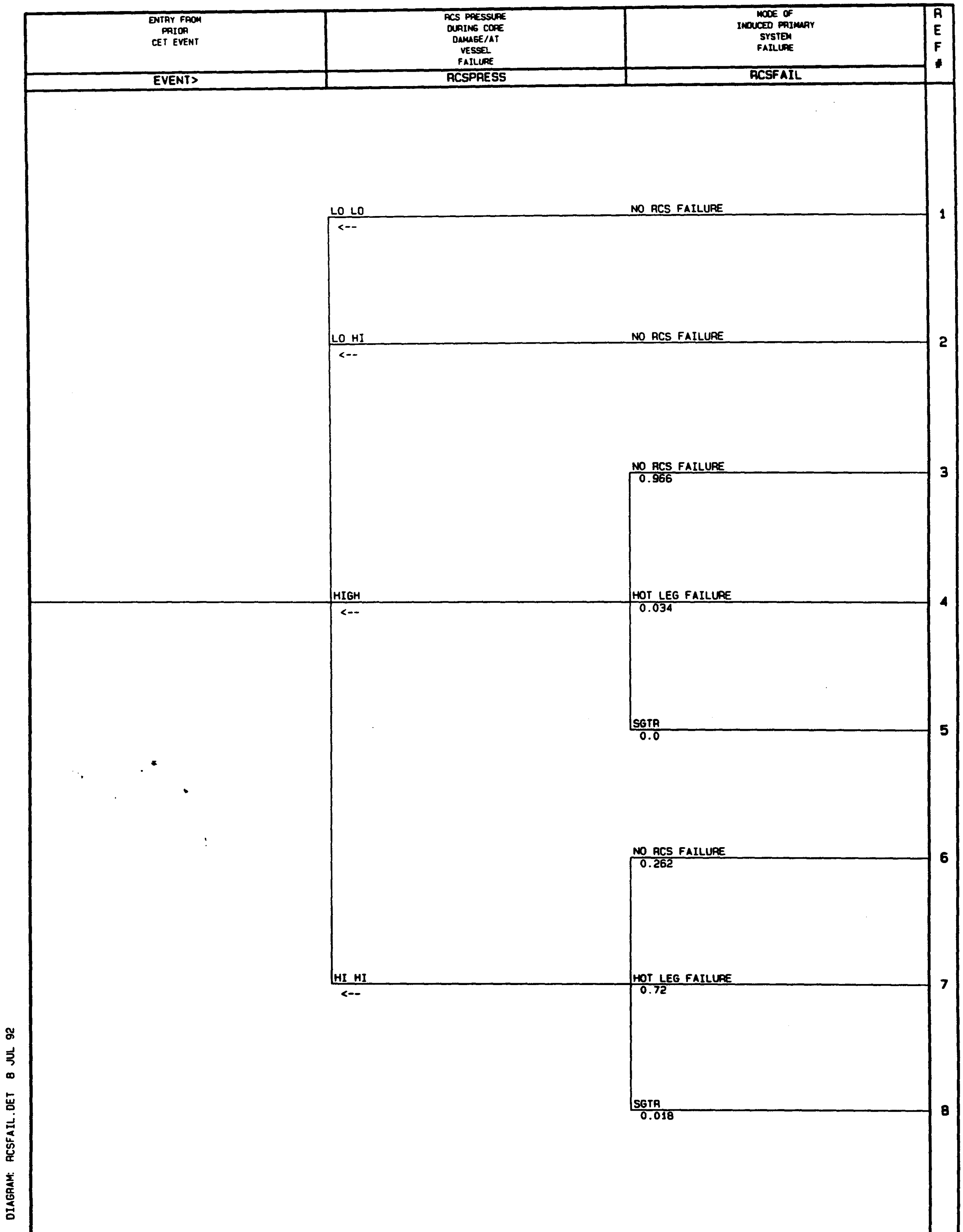


FIGURE 4.5.2-1
MODE OF INDUCED PRIMARY SYSTEM FAILURE DECOMPOSITION EVENT TREE

VIRGINIA ELECTRIC POWER COMPANY
NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
MODE OF INDUCED PRIMARY SYSTEM FAILURE
DECOMPOSITION EVENT TREE
REV. 0

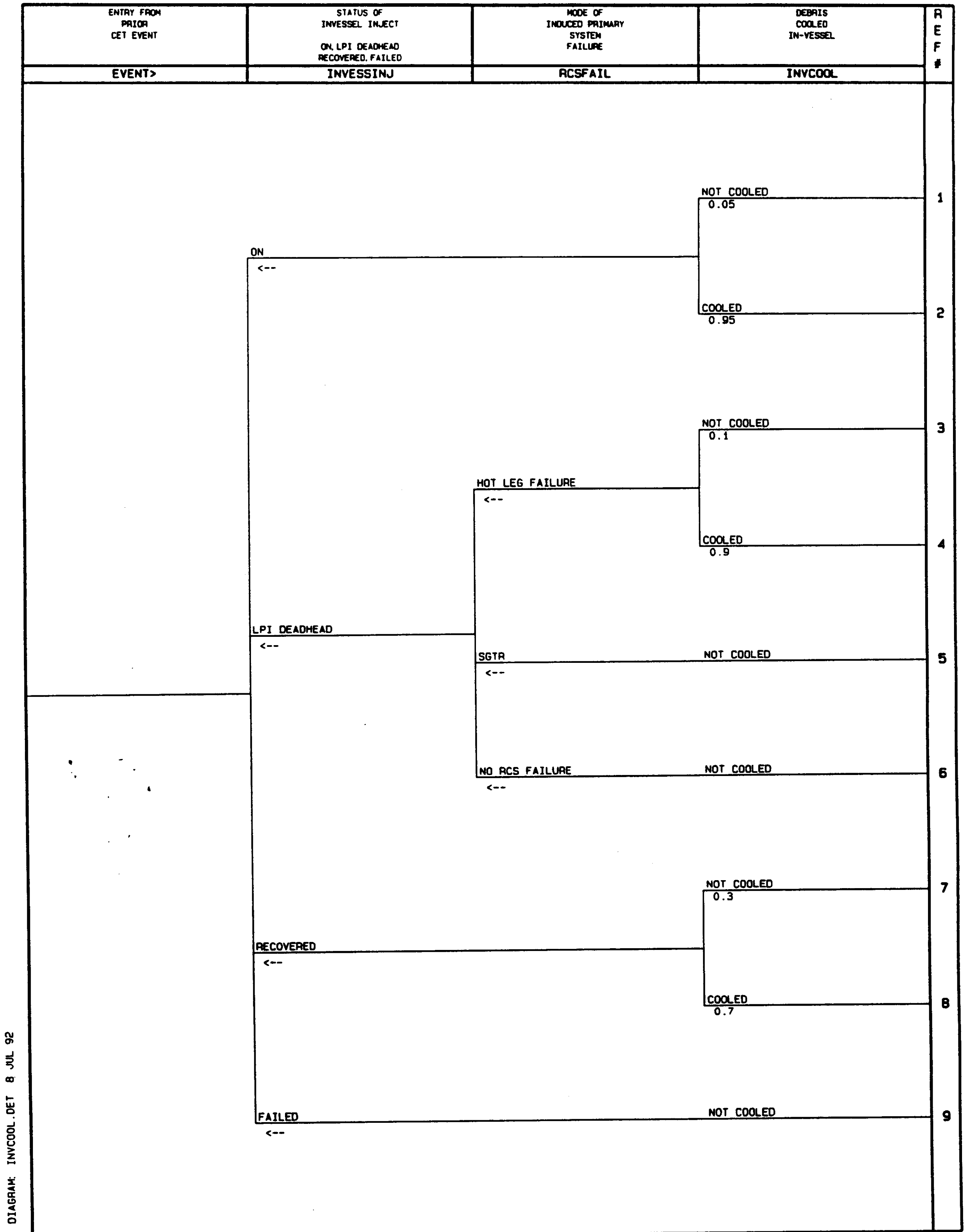


FIGURE 4.5.2-2
DEBRIS COOLED IN-VESSEL
DECOMPOSITION EVENT TREE

VIRGINIA ELECTRIC POWER COMPANY NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
DEBRIS COOLED IN-VESSEL DECOMPOSITION EVENT TREE REV. 0

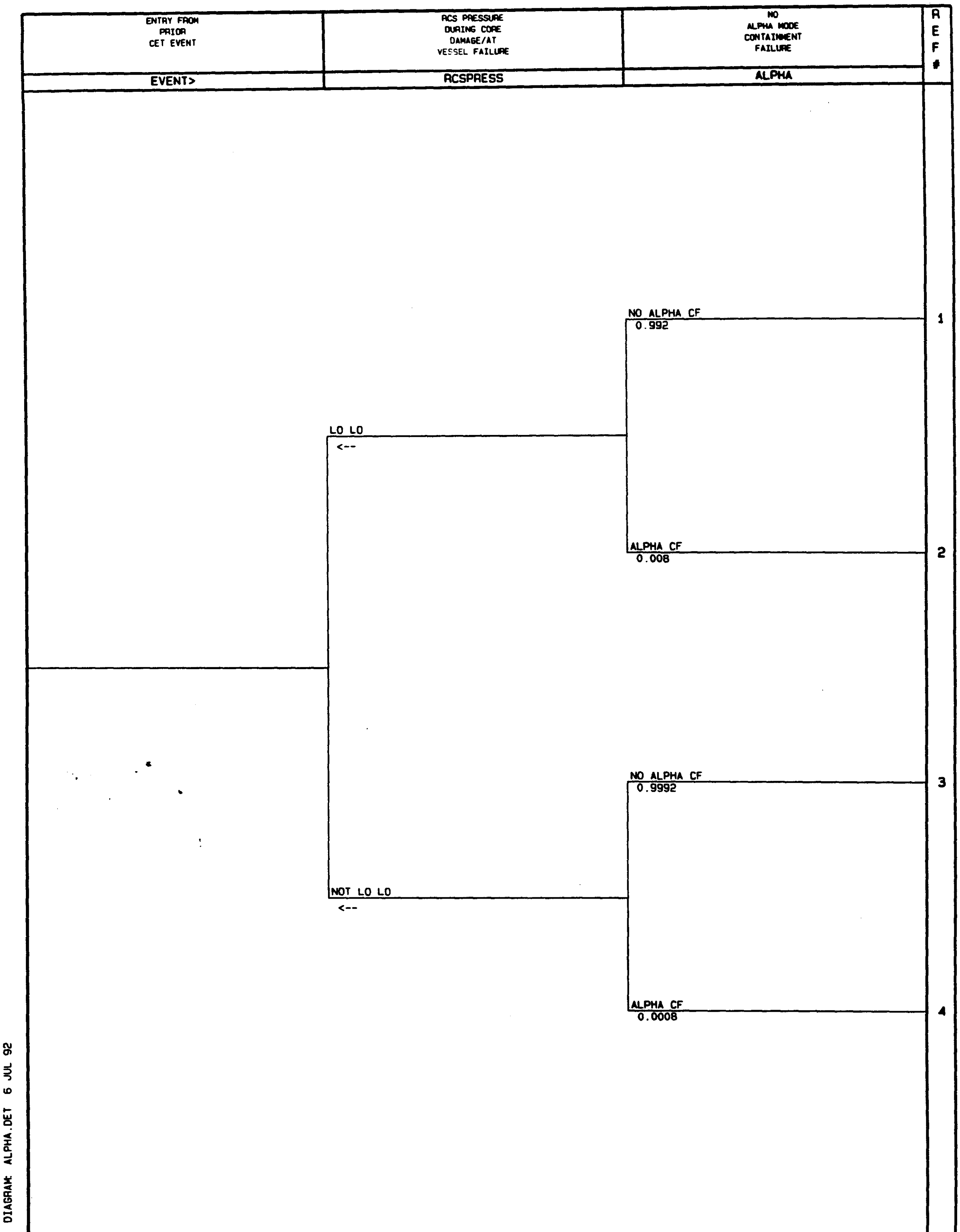


FIGURE 4.5.2-3
NO ALPHA MODE CONTAINMENT FAILURE DECOMPOSITION EVENT TREE

VIRGINIA ELECTRIC POWER COMPANY NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
NO ALPHA MODE CONTAINMENT FAILURE DECOMPOSITION EVENT TREE REV. 0

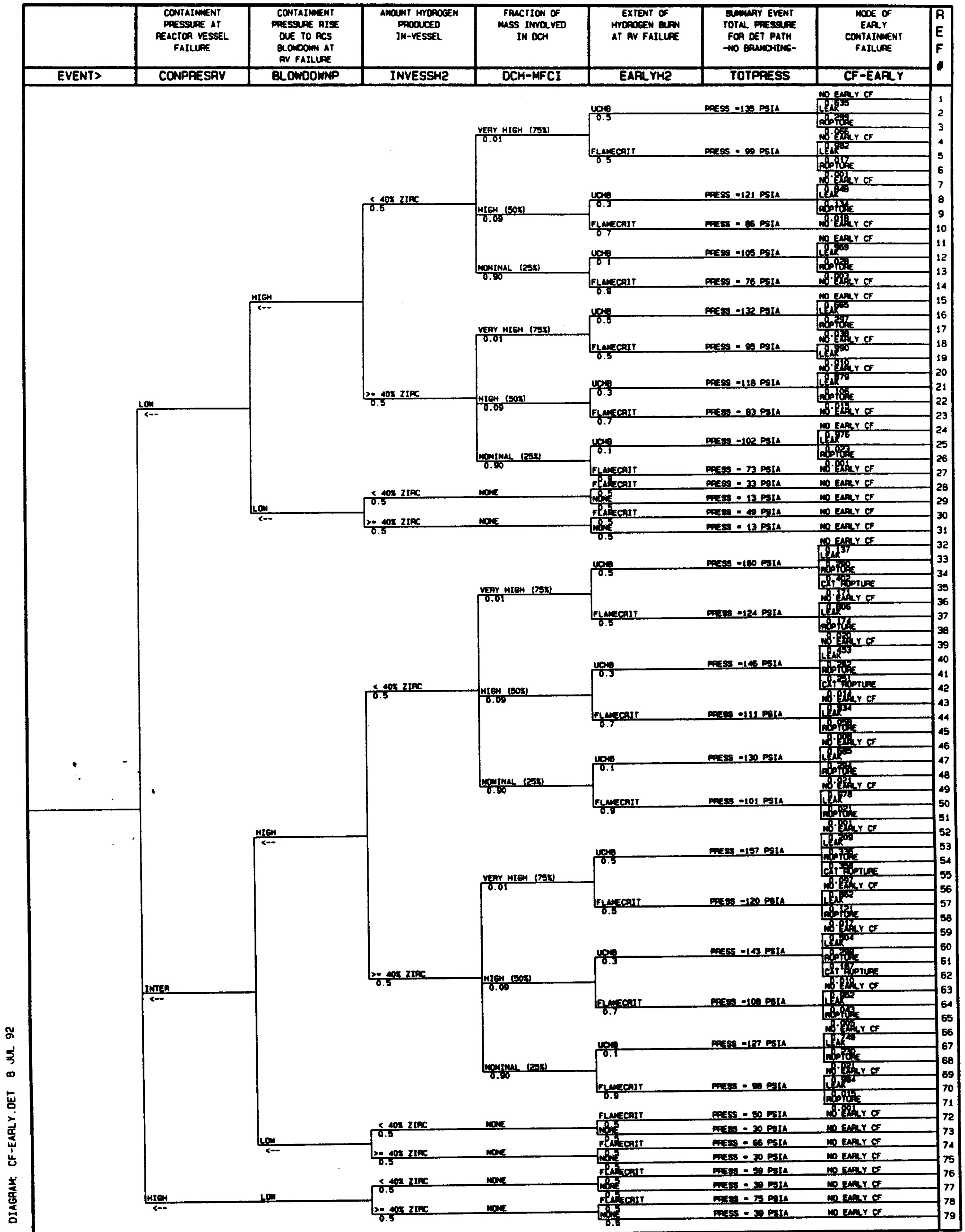


FIGURE 4.5.2-4
MODE OF EARLY CONTAINMENT FAILURE DECOMPOSITION EVENT TREE

VIRGINIA ELECTRIC POWER COMPANY
NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
MODE OF EARLY CONTAINMENT FAILURE
DECOMPOSITION EVENT TREE
REV. 0

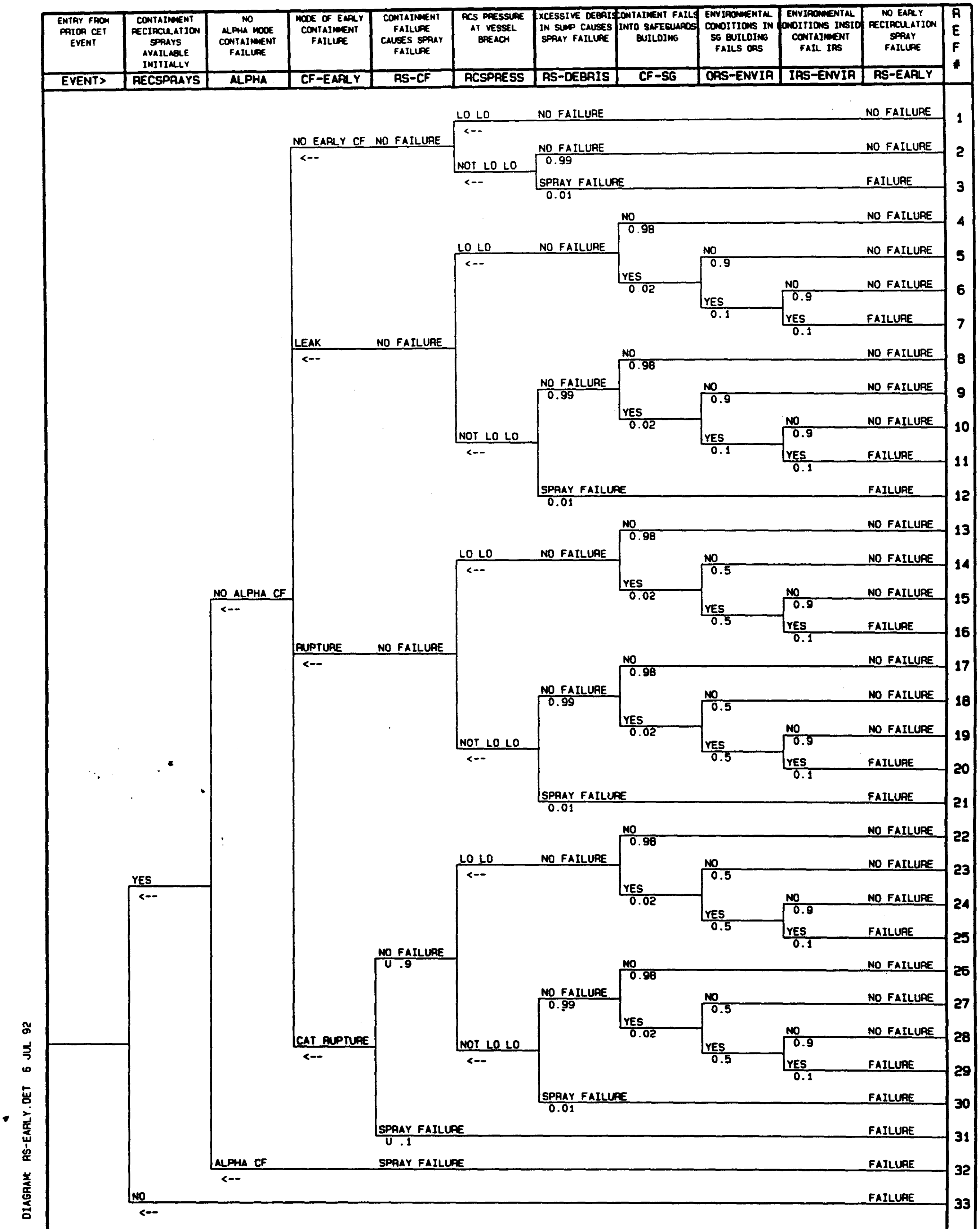


FIGURE 4.5.2-5
NO EARLY RECIRCULATION SPRAY FAILURE DECOMPOSITION EVENT TREE

VIRGINIA ELECTRIC POWER COMPANY
NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
NO EARLY RECIRCULATION SPRAY FAILURE DECOMPOSITION EVENT TREE
REV. 0

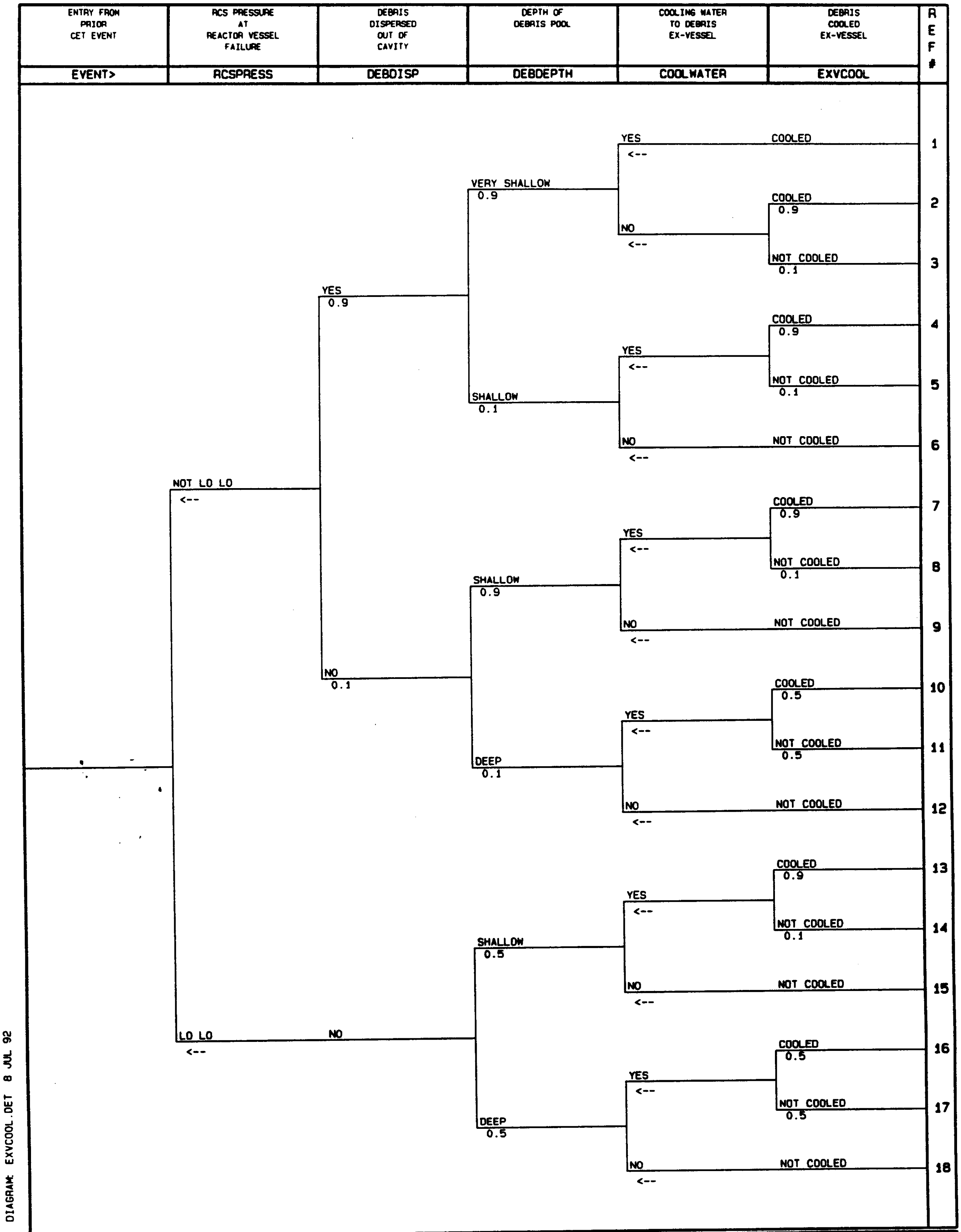


FIGURE 4.5.2-6
DEBRIS COOLED EX-VESSEL
DECOMPOSITION EVENT TREE

VIRGINIA ELECTRIC POWER COMPANY NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
DEBRIS COOLED EX-VESSEL DECOMPOSITION EVENT TREE REV. 0

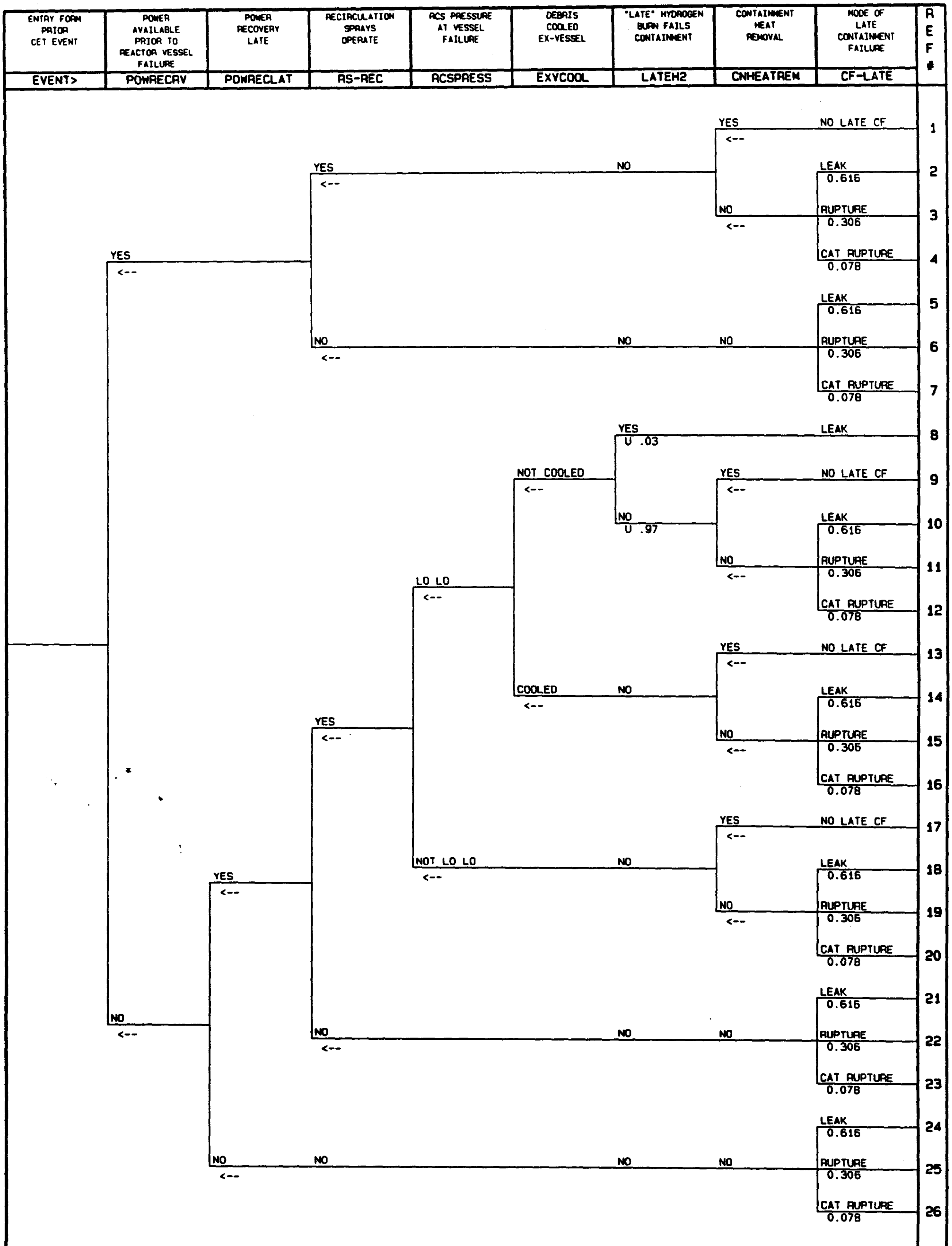


DIAGRAM: CF-LATE.DET 7 JUL 92

FIGURE 4.5.2-7
MODE OF LATE CONTAINMENT FAILURE DECOMPOSITION TREE

VIRGINIA ELECTRIC POWER COMPANY
NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
MODE OF LATE CONTAINMENT FAILURE
DECOMPOSITION TREE
REV. 0

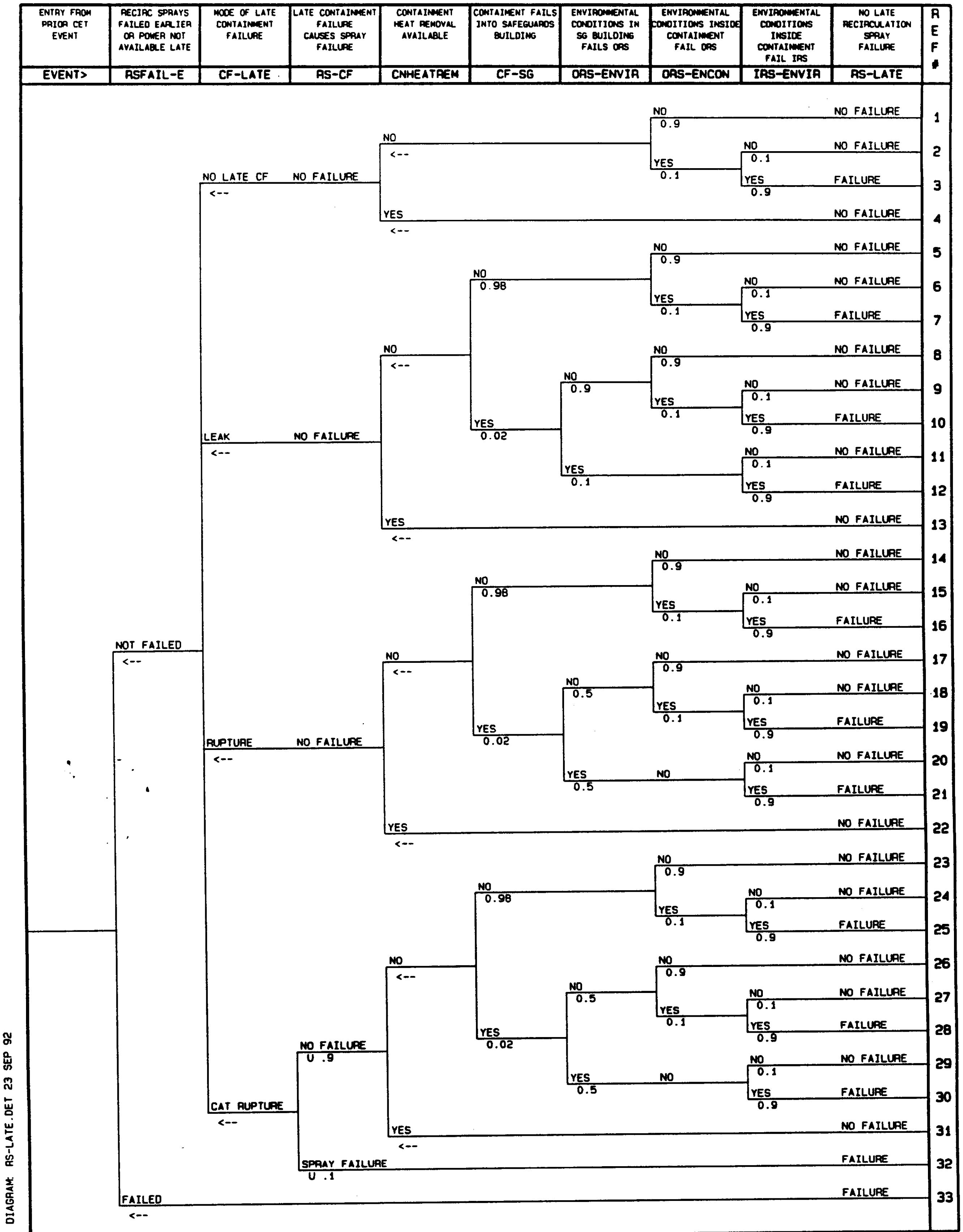


FIGURE 4.5.2-8
NO LATE RECIRCULATION SPRAY FAILURE DECOMPOSITION EVENT TREE

VIRGINIA ELECTRIC POWER COMPANY
NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
NO LATE RECIRCULATION SPRAY FAILURE DECOMPOSITION EVENT TREE
REV. 0

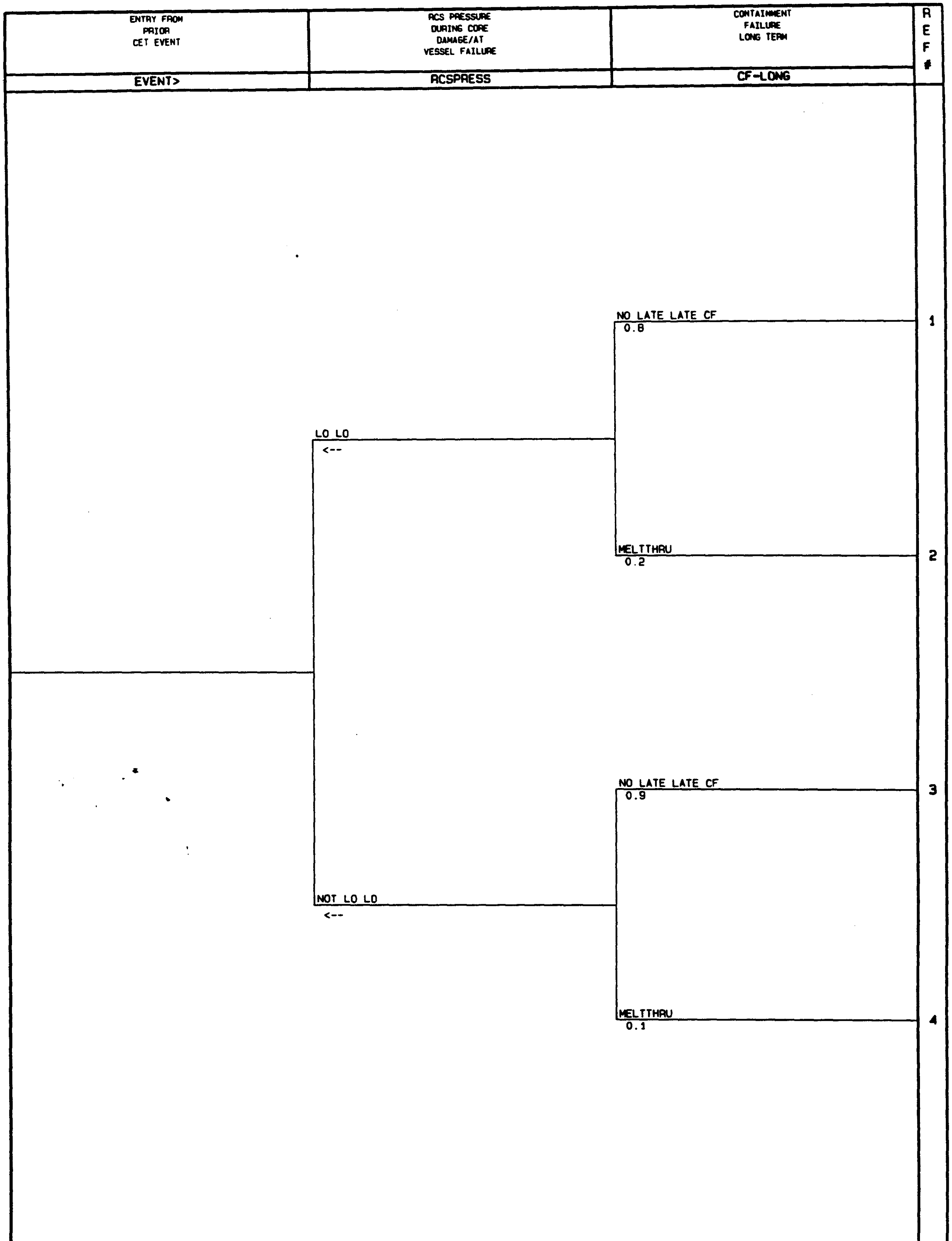


DIAGRAM: CF-LONG.DET 9 JUL 92

FIGURE 4.5.2-9
CONTAINMENT FAILURE LONG TERM DECOMPOSITION EVENT TREE

VIRGINIA ELECTRIC POWER COMPANY
NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
CONTAINMENT FAILURE LONG TERM
DECOMPOSITION EVENT TREE
REV. 0

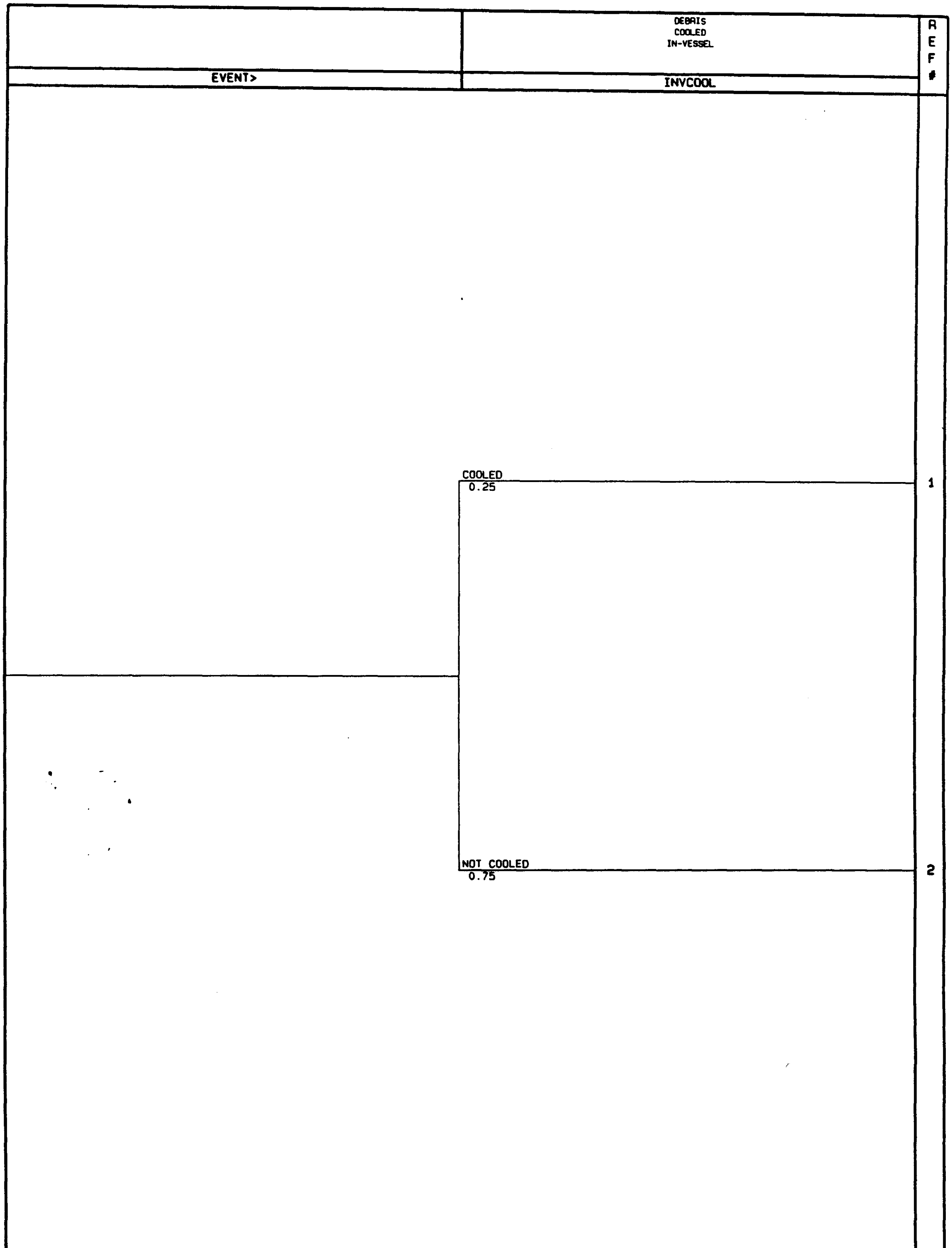


DIAGRAM: NAPSIS01.DET 7 JUL 92

FIGURE 4.5.2-10
IN-VESSEL COOLING FOR LOSS OF ISOLATION SEQUENCES DECOMPOSITION EVENT TREE

VIRGINIA ELECTRIC POWER COMPANY
NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
IN-VESSEL COOLING FOR LOSS OF ISOLATION SEQUENCES DECOMPOSITION EVENT TREE
REV. 0

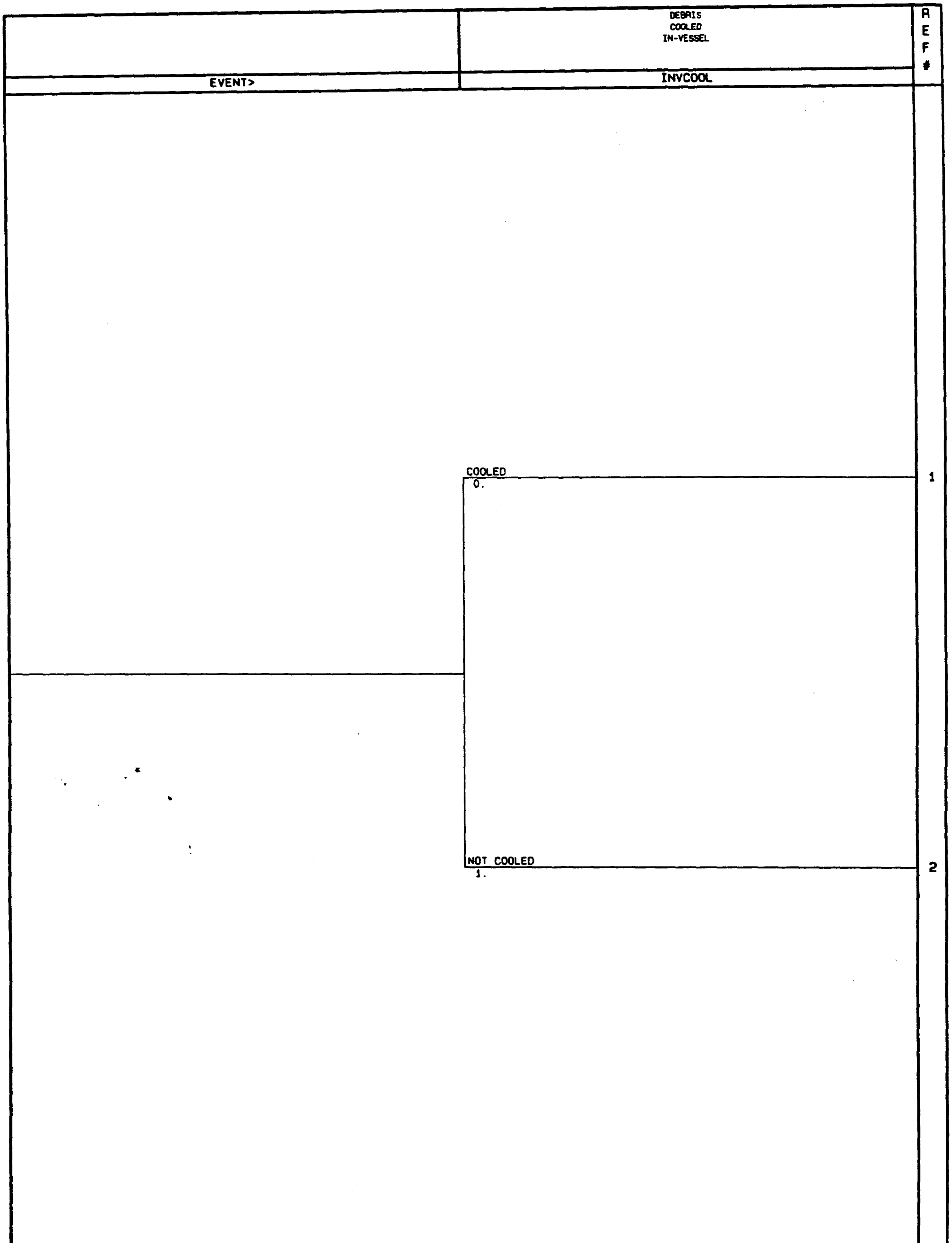


DIAGRAM: NAPSIS02.DET 8 JUL 92

FIGURE 4.5.2-11
IN-VESSEL COOLING FOR LOSS OF ISOLATION SEQUENCES DECOMPOSITION EVENT TREE

VIRGINIA ELECTRIC POWER COMPANY NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
IN-VESSEL COOLING FOR LOSS OF ISOLATION SEQUENCES DECOMPOSITION EVENT TREE REV. 0

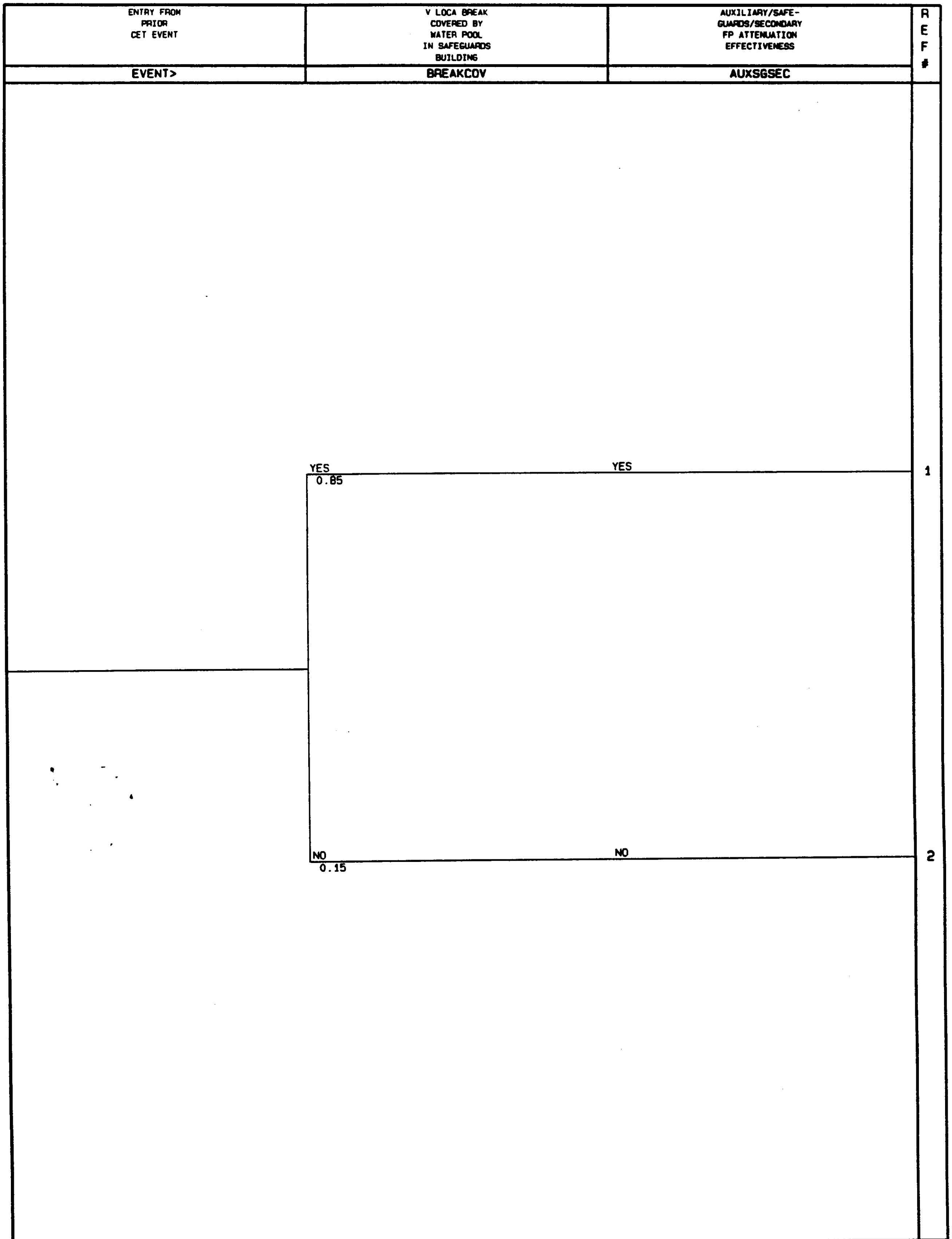


DIAGRAM: AUXSGSEC.DET 5 JUL 92

FIGURE 4.5.2-12
EVENT V SAFEGUARDS BUILDING FP RETENTION EFFECTIVENESS DECOMPOSITION EVENT TREE

VIRGINIA ELECTRIC POWER COMPANY NORTH ANNA POWER STATION INDIVIDUAL PLANT EXAMINATION PRA
EVENT V SAFEGUARDS BUILDING FP RETENTION EFFECTIVENESS DECOMPOSITION EVENT TREE REV. 0

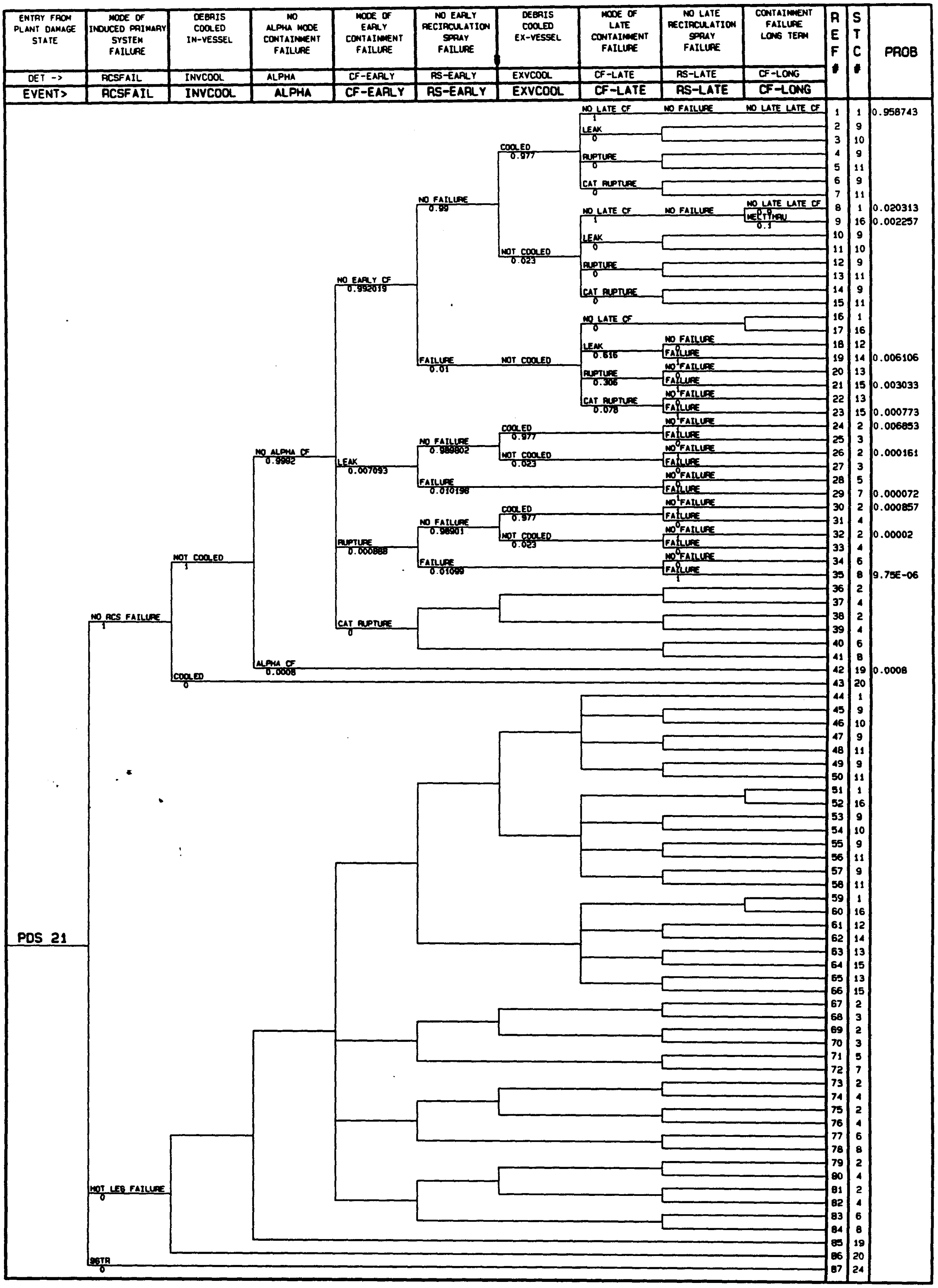


DIAGRAM: NAPSGEN .CET 19 FEB 92 DATA FILE: F4531.CDB Quantified: 27 SEP 92 Sum = 1.000E+000 PDS: REV2NAPS.PDD

FIGURE 4.5.3-1 Containment Event Tree for PDS 21

DIAGRAM: NAPSGEN .CET 19 FEB 92 DATA FILE: F4532.CDB Quantified: 27 SEP 92 Sum = 1.000E+000 PDS: REV2NAPS.PDD

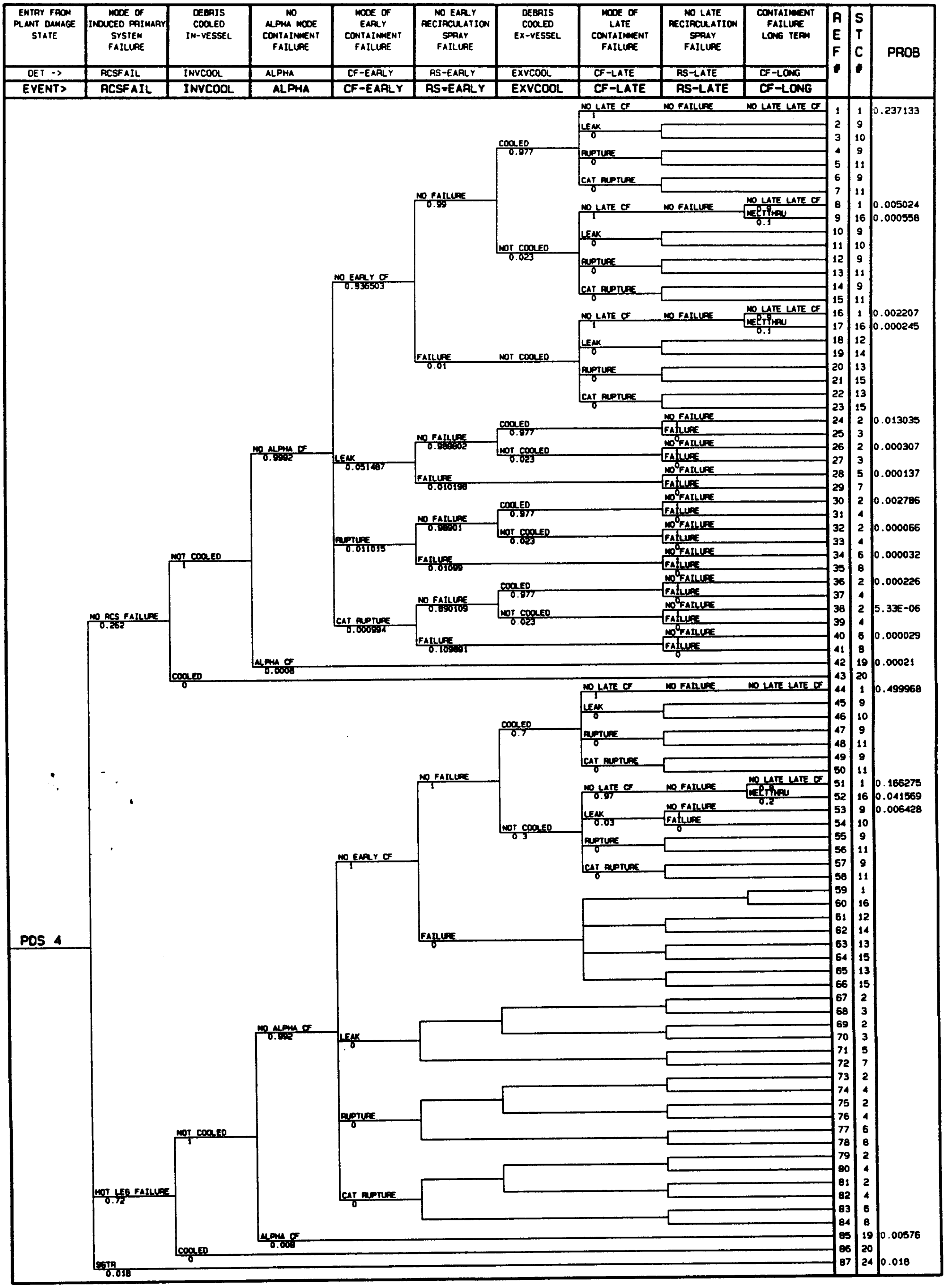


FIGURE 4.5.3-2 Containment Event Tree for PDS 4

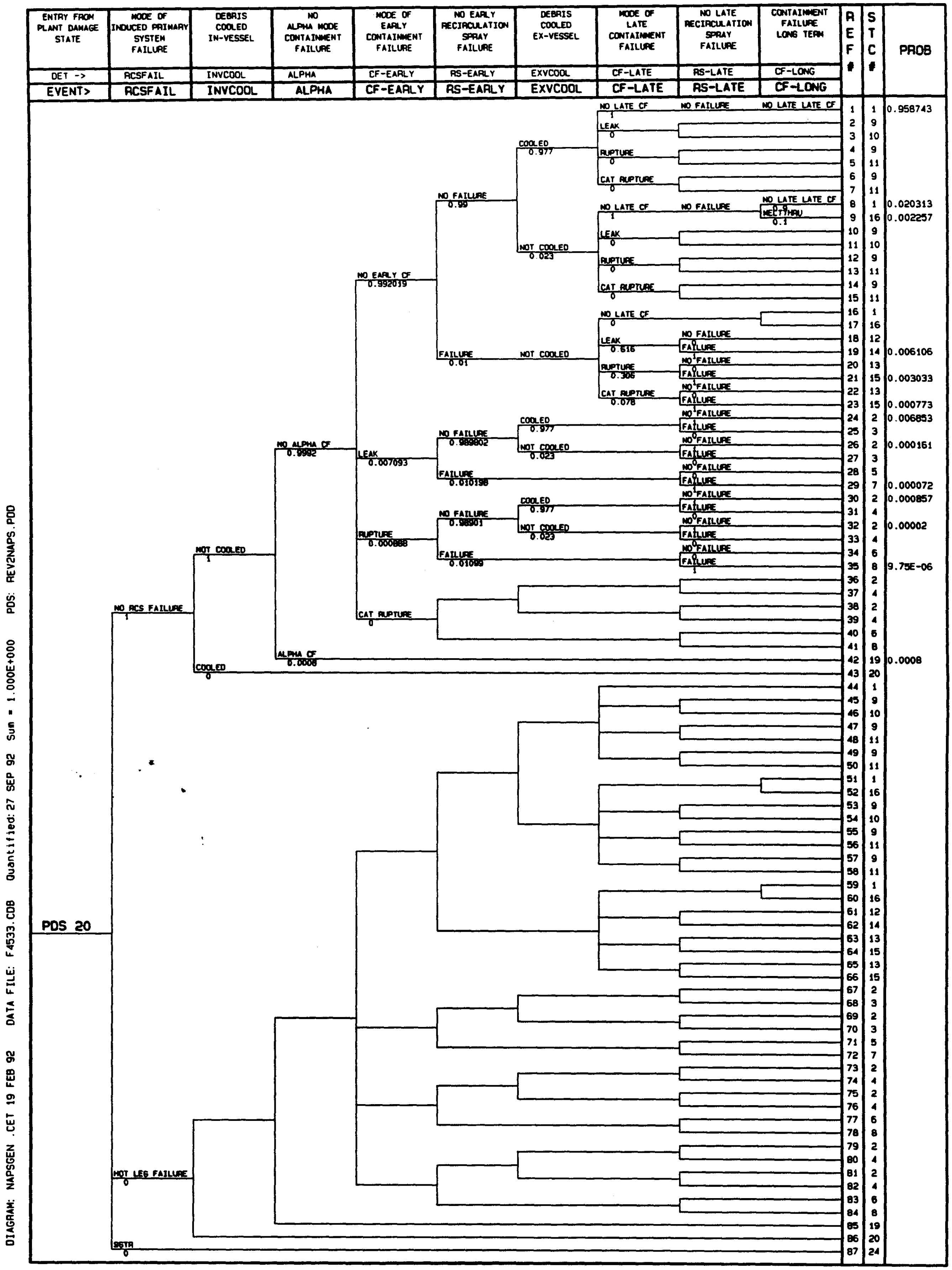


FIGURE 4.5.3-3 Containment Event Tree for PDS 20

ENTRY FROM PLANT DAMAGE STATE	AUXILIARY/SAFE-GUARDS/SECONDARY FP ATTENUATION EFFECTIVE	REF #	STC #	PROB
DET ->	AUXSGSEC			
EVENT>	AUXSGSEC			
PDS 25	NO	1	24	1.0

DIAGRAM: NAPSSGTR.CET 19 FEB 92 DATA FILE: F4534.CDB Quantified: 27 SEP 92 Sum = 1.000E+000 PDS: REV2NAPS.PDD

FIGURE 4.5.3-4 Containment Event Tree for PDS 25

DIAGRAM: NAPSGEN .CET 19 FEB 92 DATA FILE: F4535.CDB Quantified: 27 SEP 92 Sum = 1.000E+000 PDS: REV2NAPS.PDD

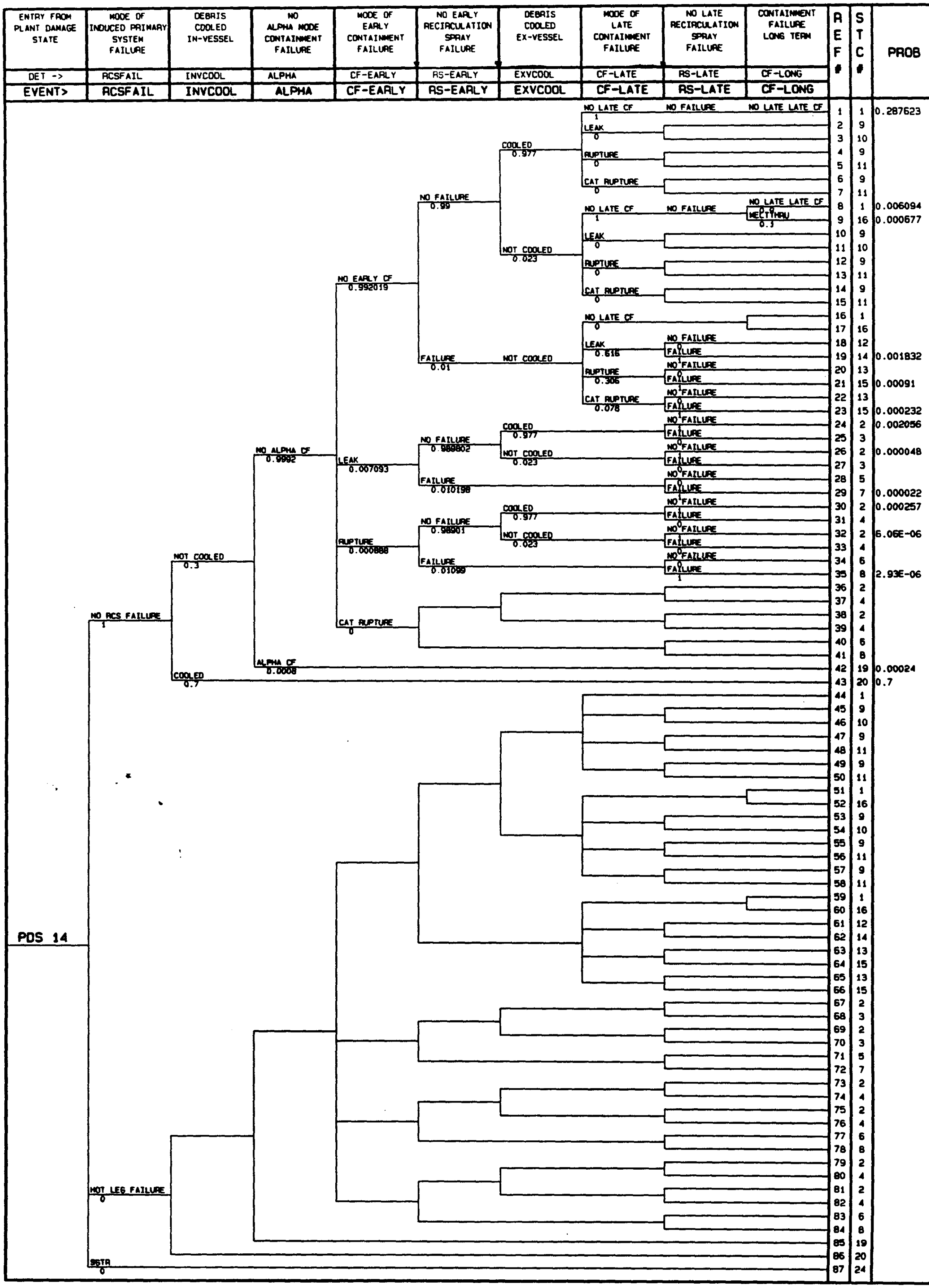


FIGURE 4.5.3-5 Containment Event Tree for PDS 14

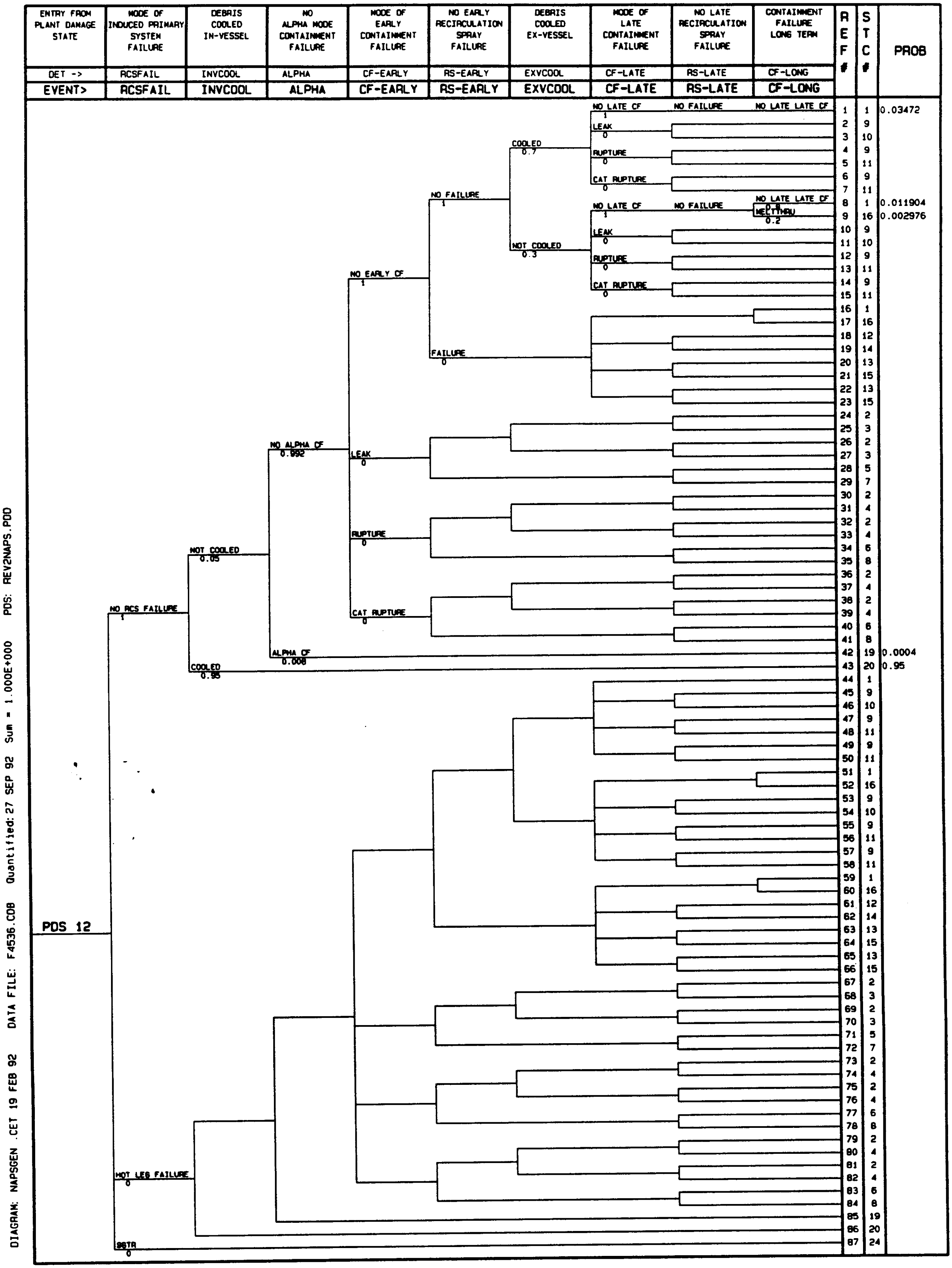


FIGURE 4.5.3-6 Containment Event Tree for PDS 12

Figure 4.6.2-1
Primary System Pressure:
6" LOCA

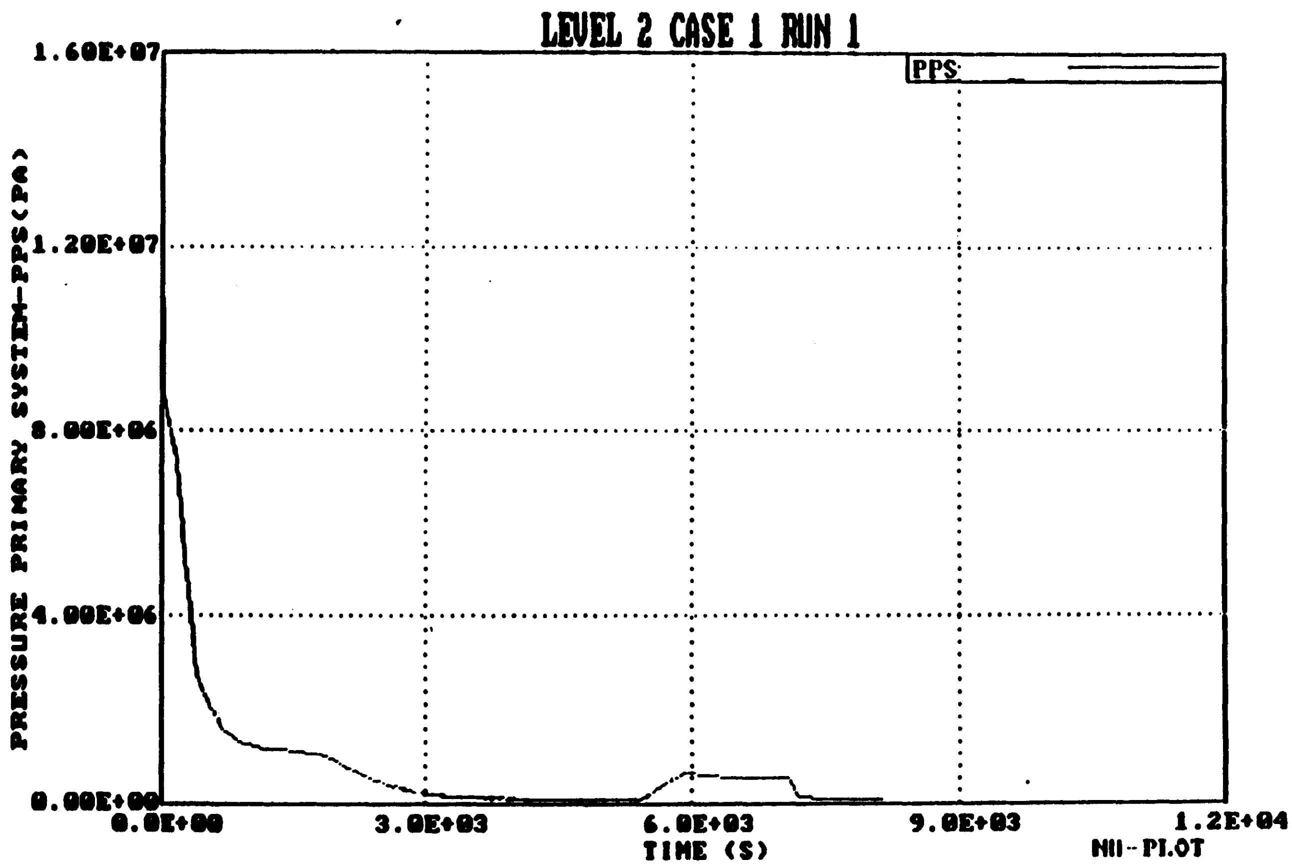


Figure 4.6.2-2
Primary System Pressure:
4" LOCA

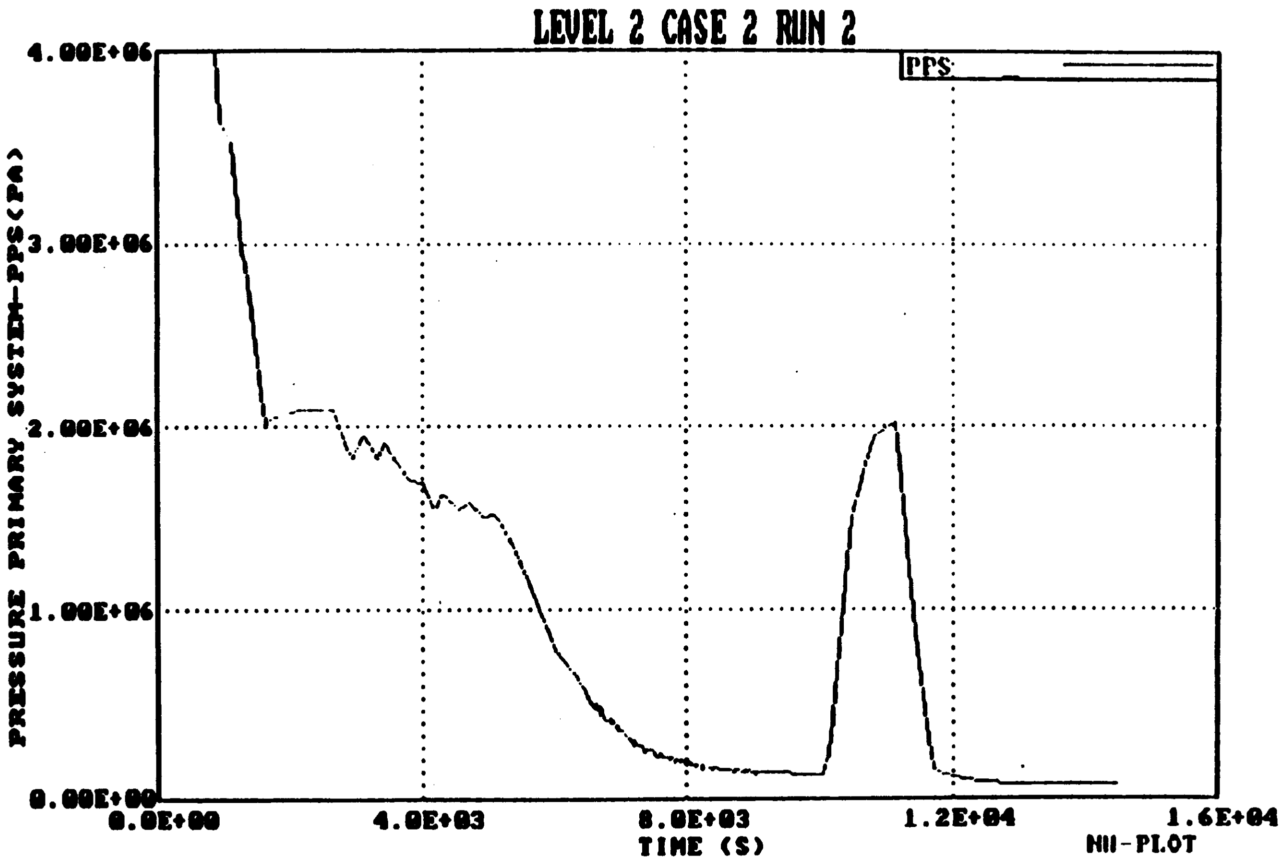


Figure 4.6.2-3
Primary System Pressure:
2" LOCA

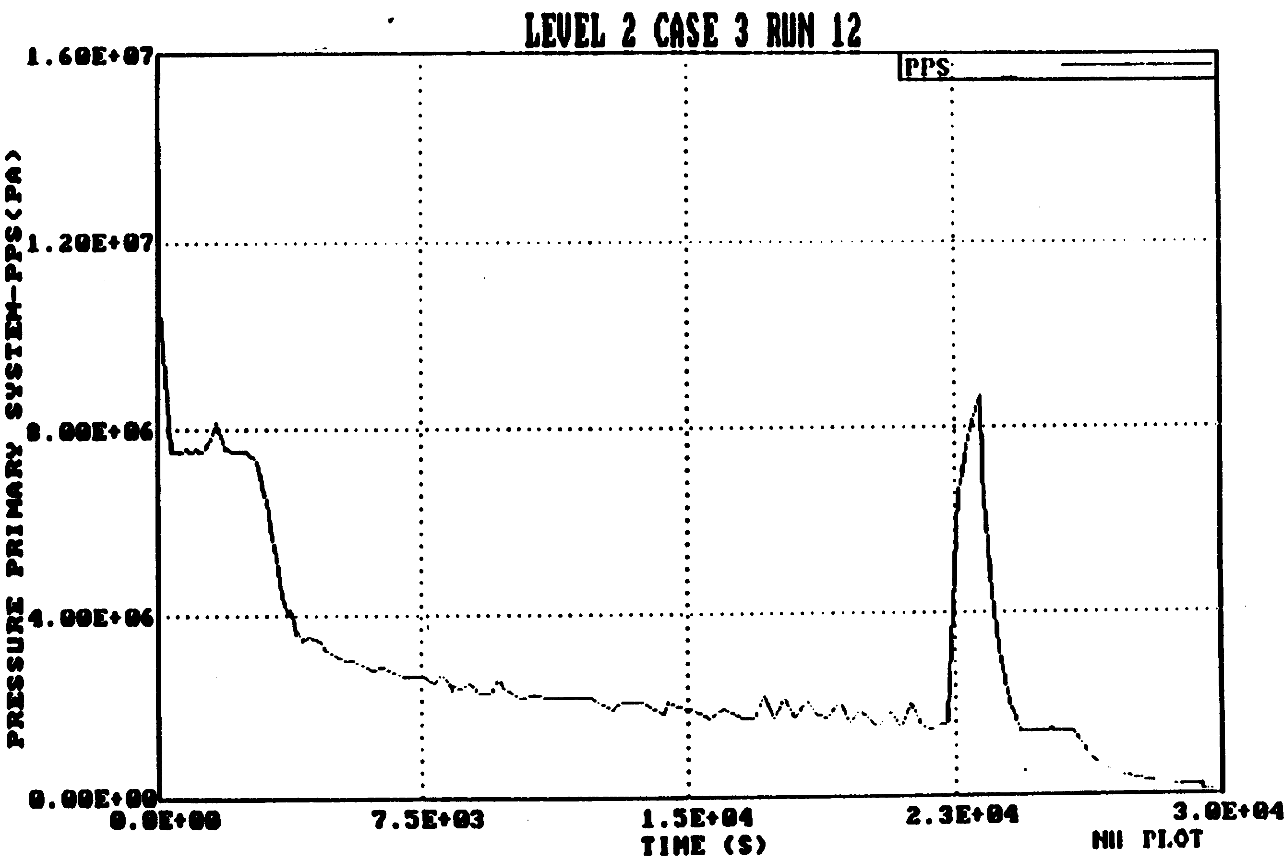


Figure 4.6.2-4
Primary System Pressure:
SBO Event

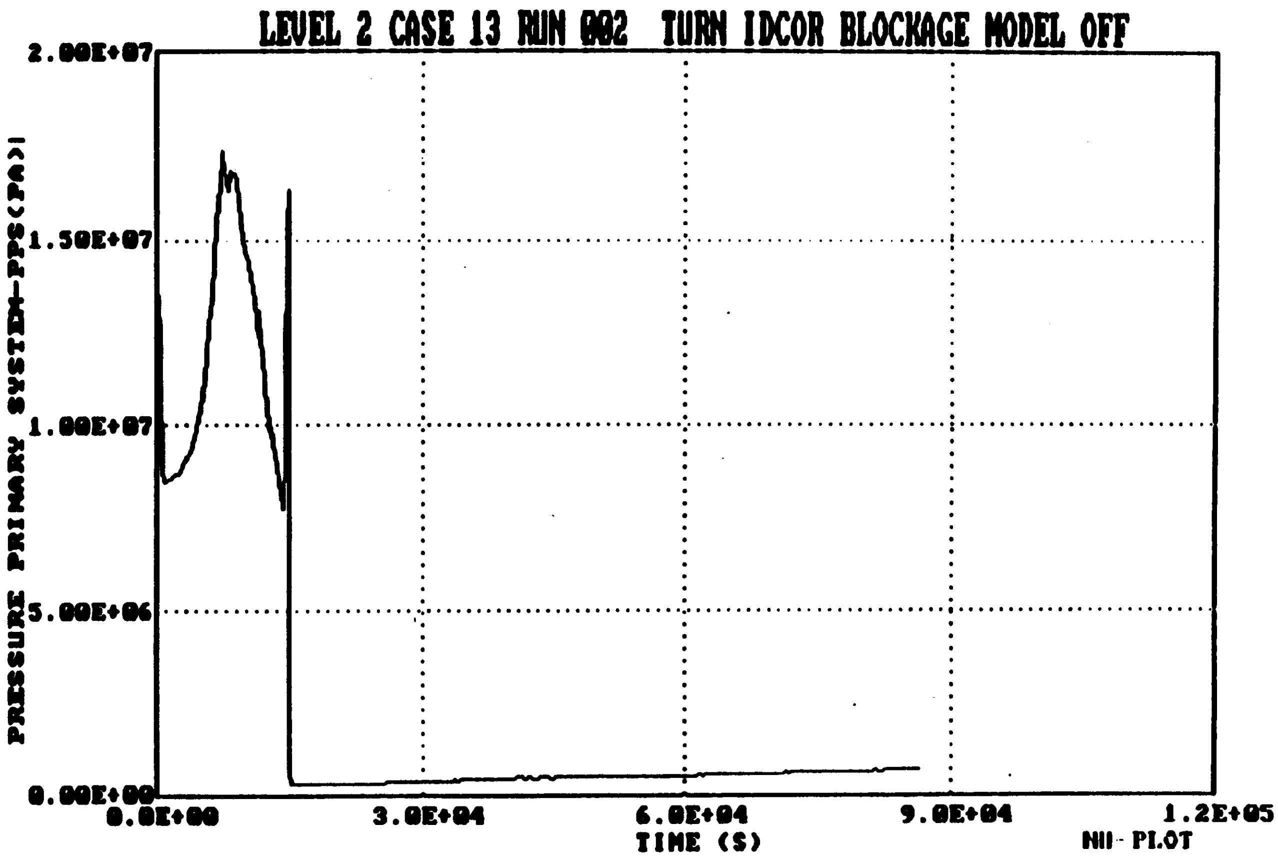
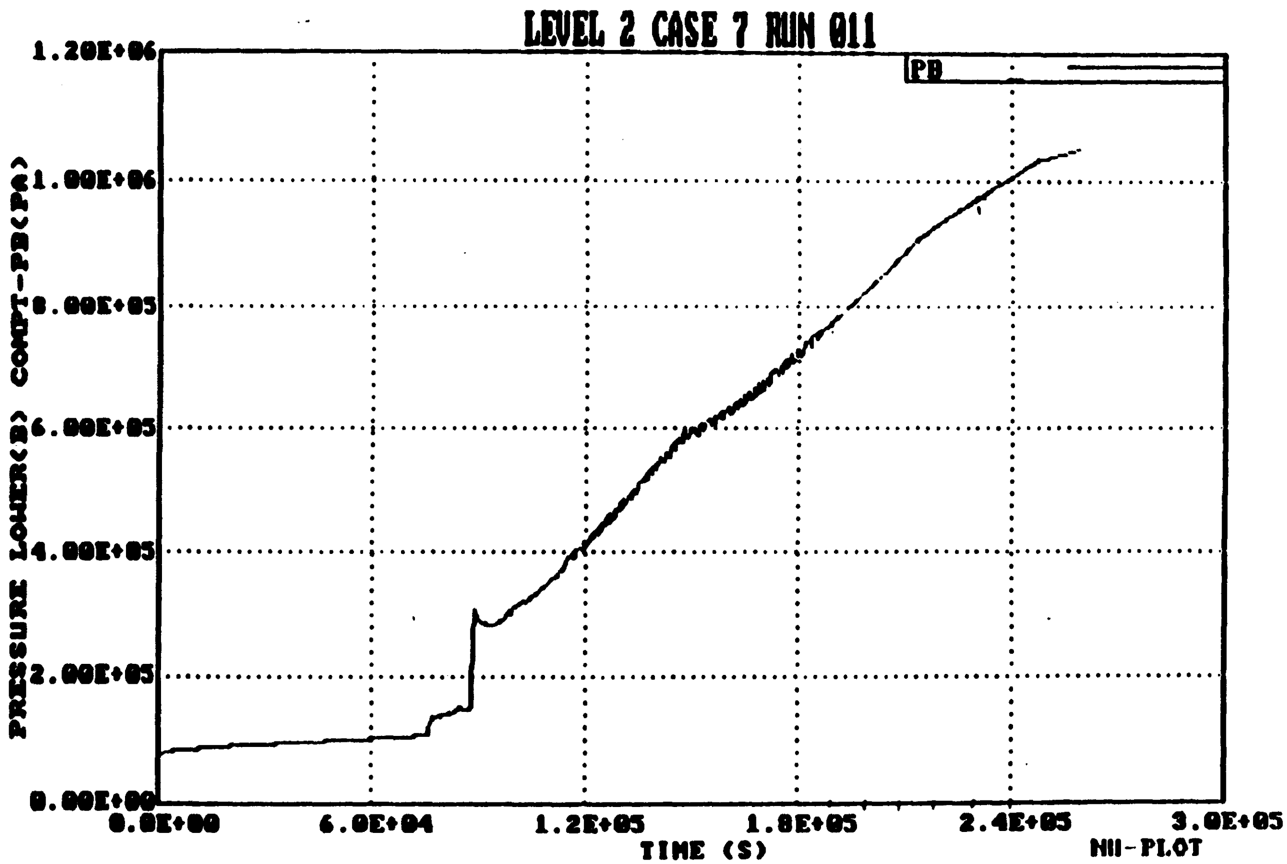


FIGURE 4.6.2-5

CONTAINMENT PRESSURE: LT SBO With Seal LOCA

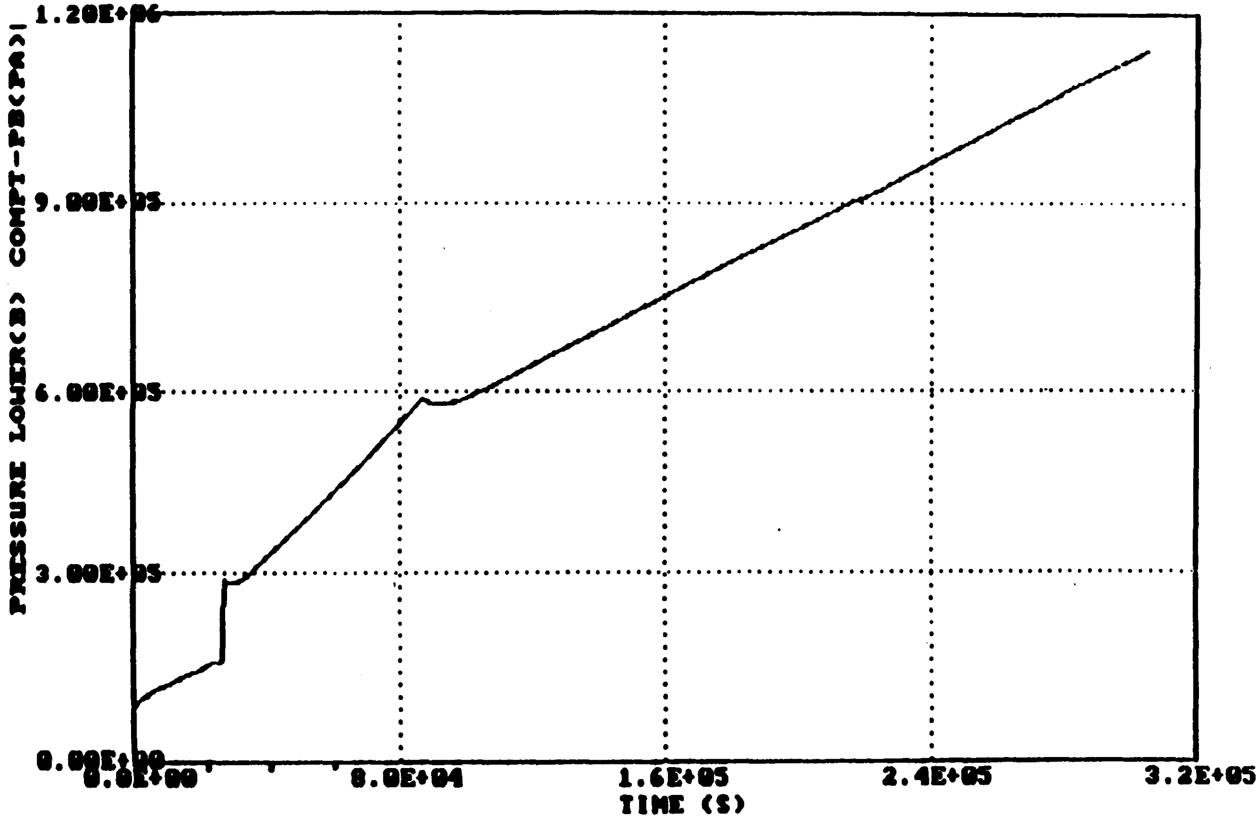


Core at 2200°F peak: 82000 s

Vessel Failure: 88157 s

FIGURE 4.6.2-6

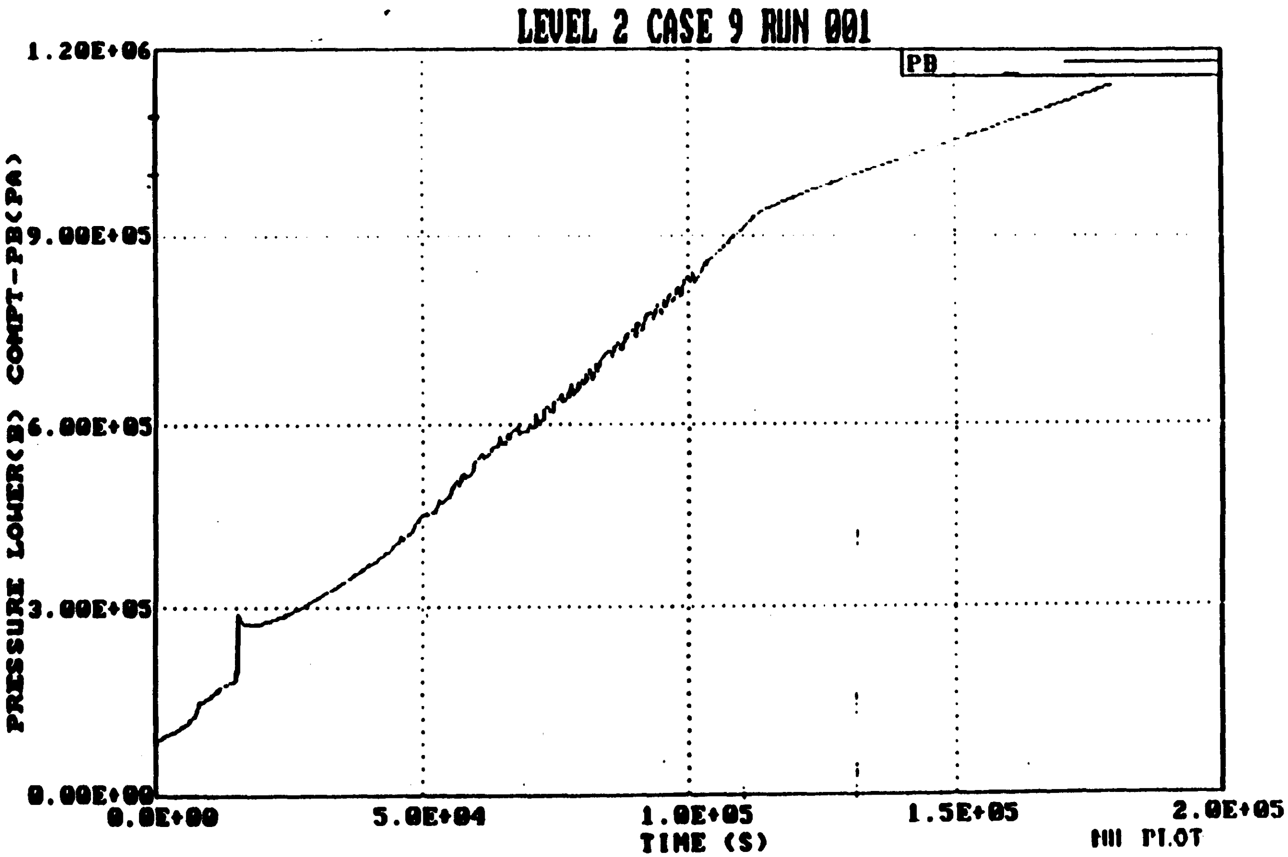
CONTAINMENT PRESSURE: LT SBO without Seal LOCA



Core at 2200^{o}F peak: 20,000 s
Vessel Failure: 26517 s

FIGURE 4.6.2-7

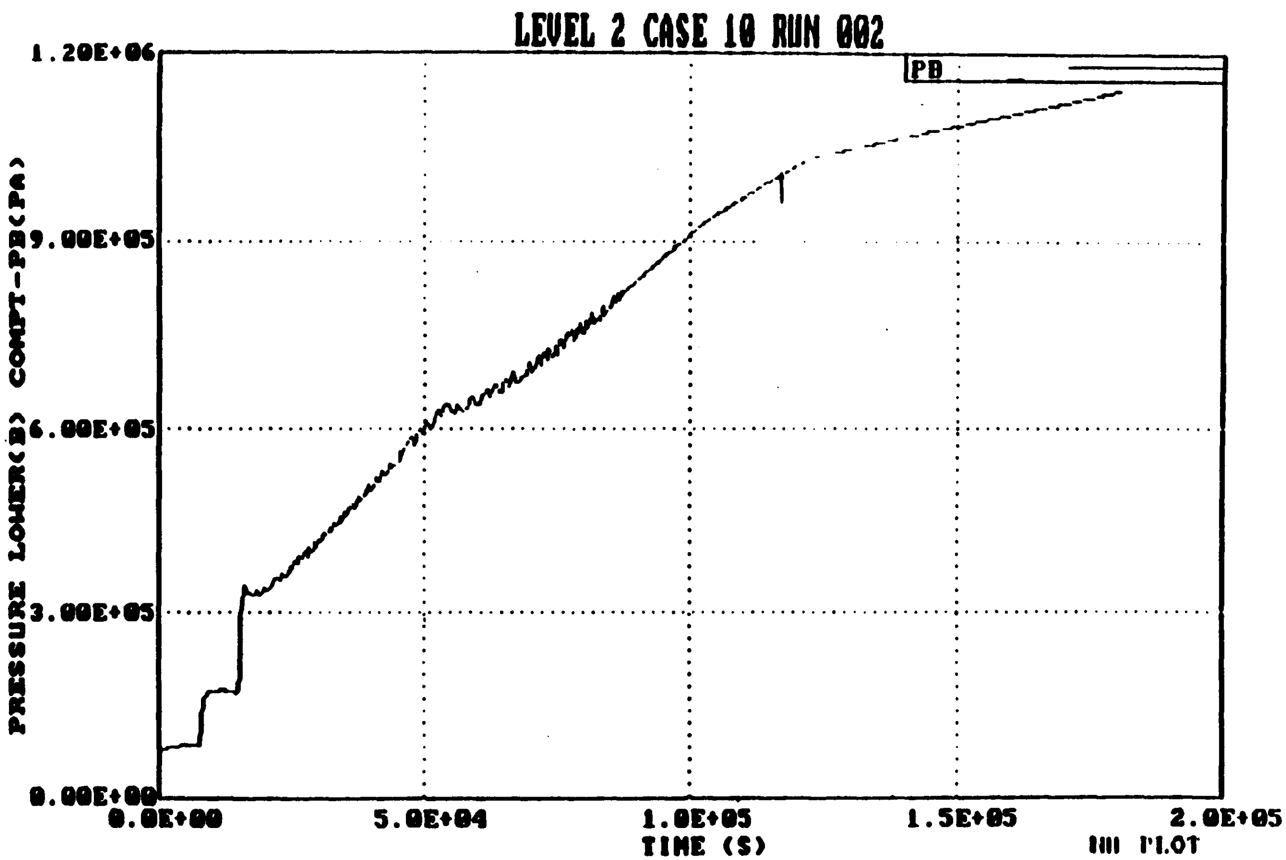
CONTAINMENT PRESSURE: ST SBO With Seal LOCA



Core at 2200°F peak: 10000 s
Vessel Failure: 15420 s

FIGURE 4.6.2-8

CONTAINMENT PRESSURE: ST SBO without Seal LOCA



Core at 2200°F peak: 11300 s

Vessel Failure: 15272 s

FIGURE 4.6.2-9

CONTAINMENT PRESSURE: 6" LOCA With Sprays and Heat Removal

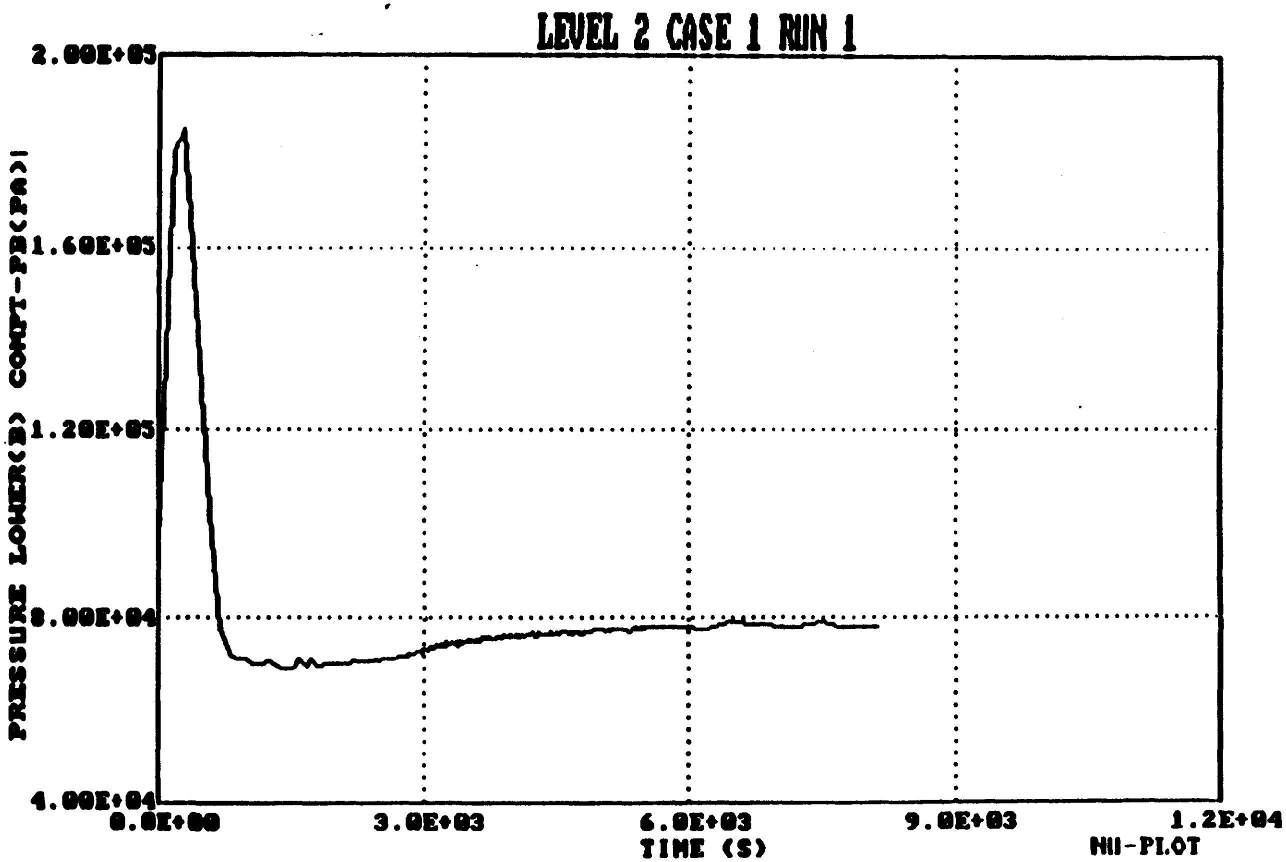


FIGURE 4.6.2-10

CONTAINMENT PRESSURE: 2" LOCA with Sprays and Heat Removal

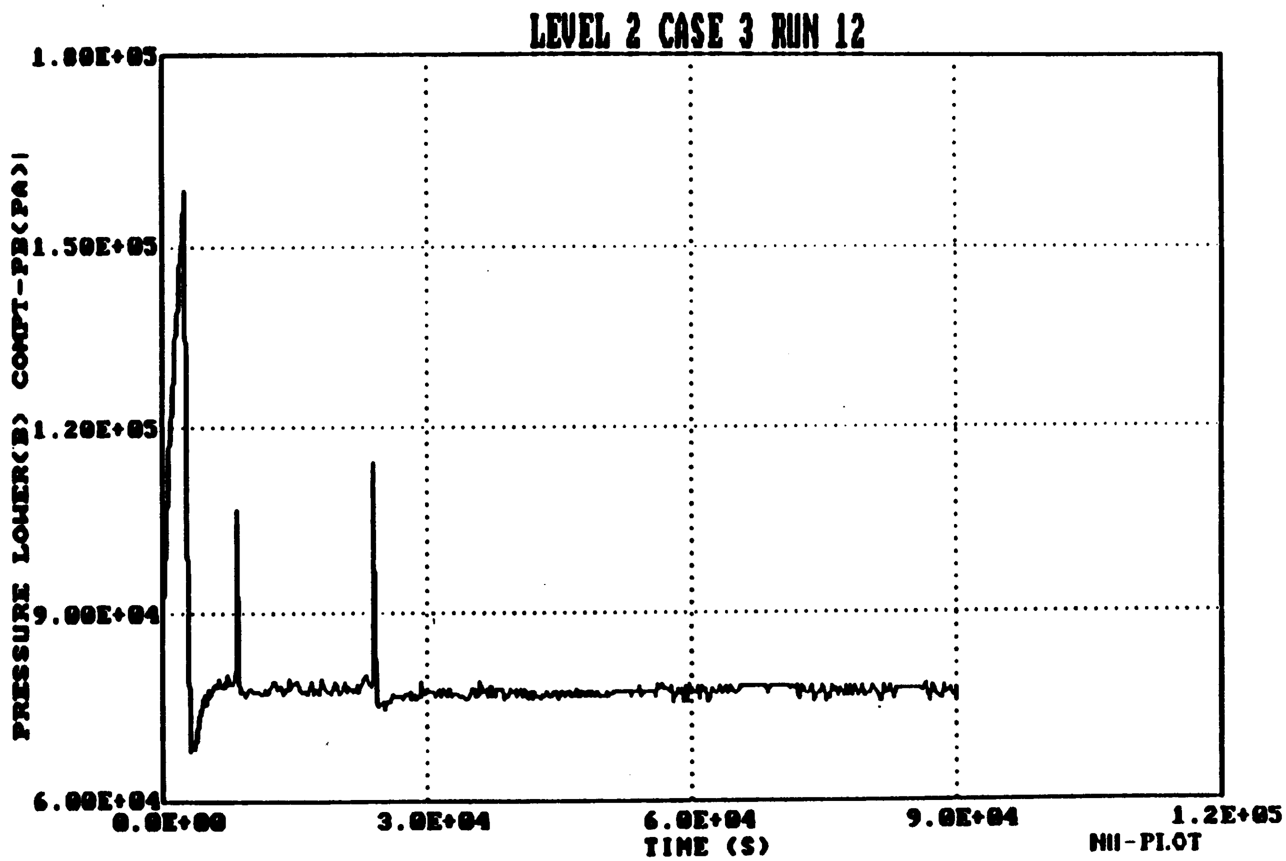


FIGURE 4.6.2-11

CONTAINMENT PRESSURE: Large LOCA with No Sprays

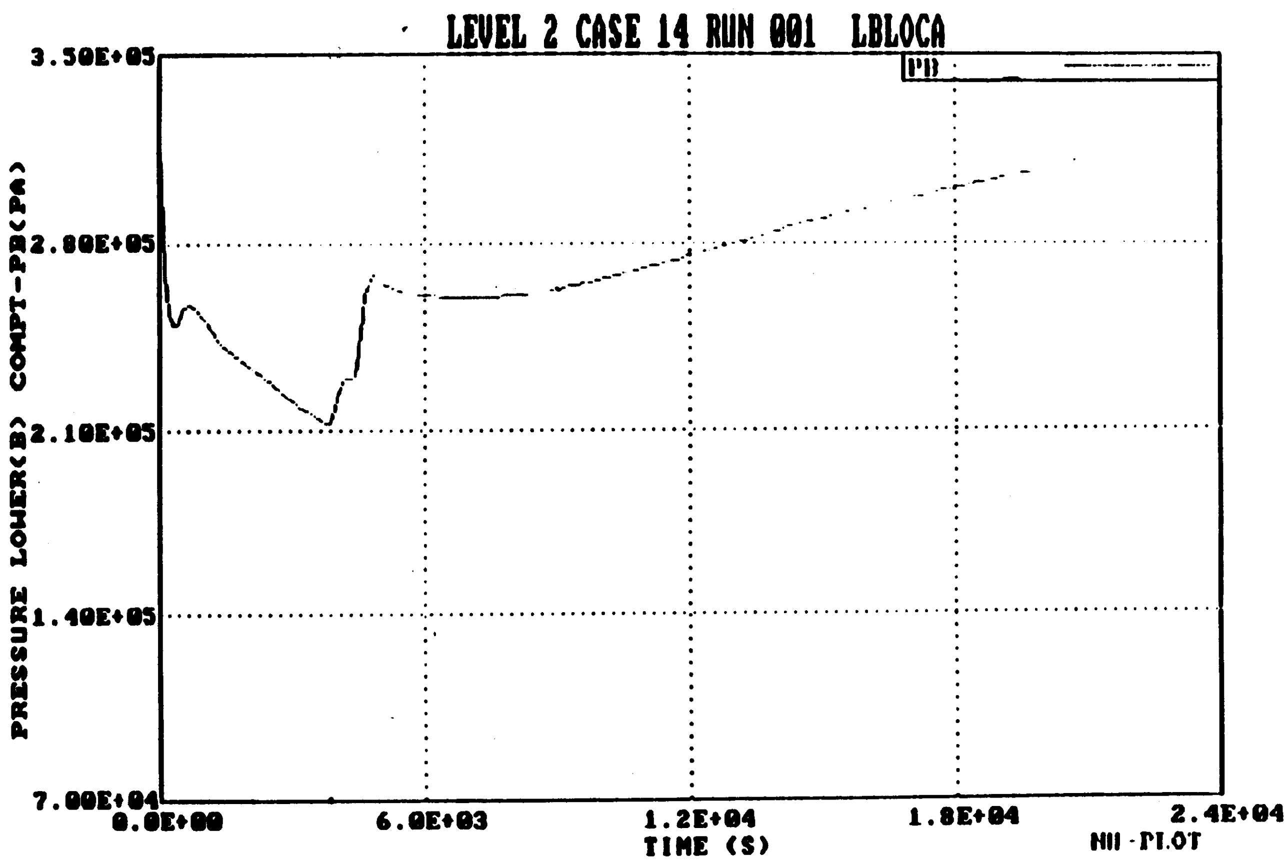


FIGURE 4.6.2-12

CONTAINMENT PRESSURE: 2" LOCA without Sprays and Heat Removal

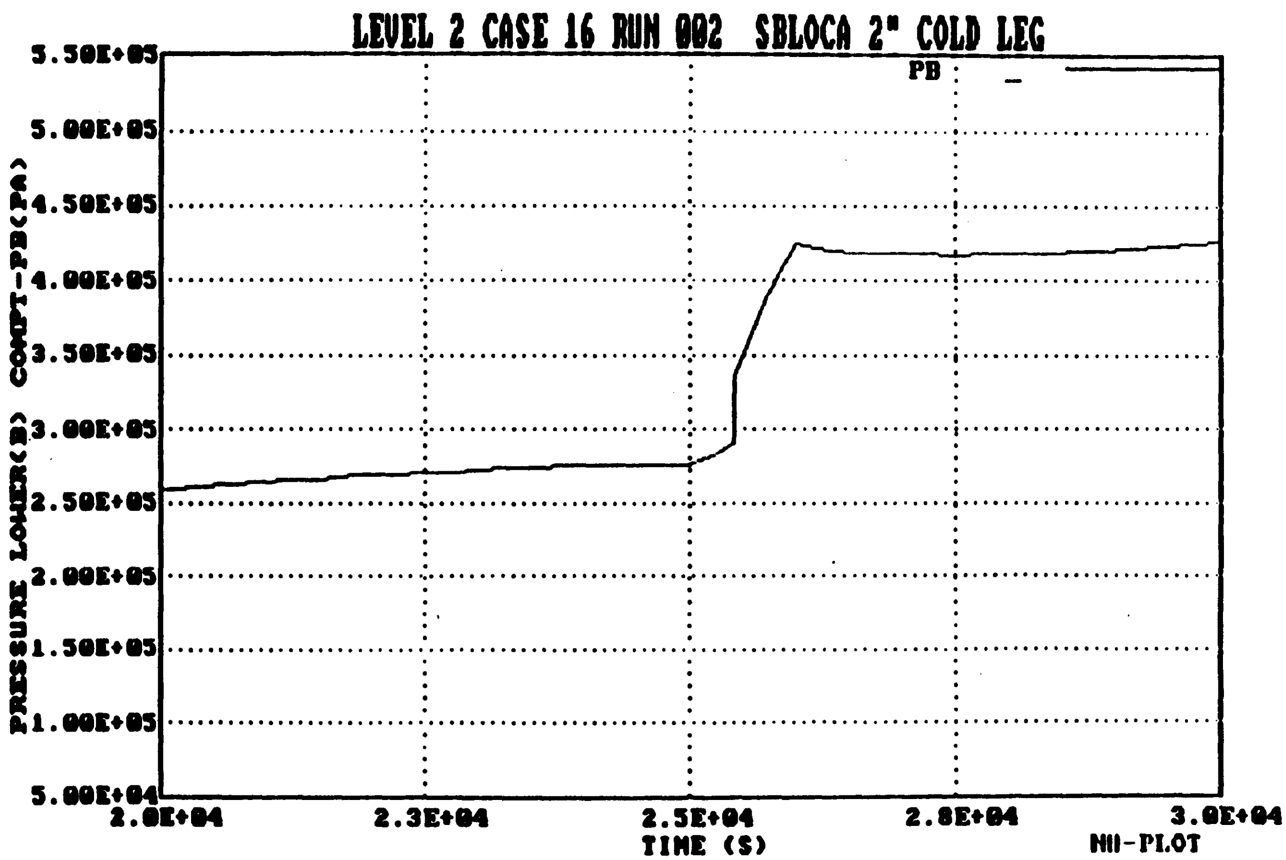


FIGURE 4.6.2-13

CONTAINMENT PRESSURE: 6" LOCA without Sprays and Heat Removal

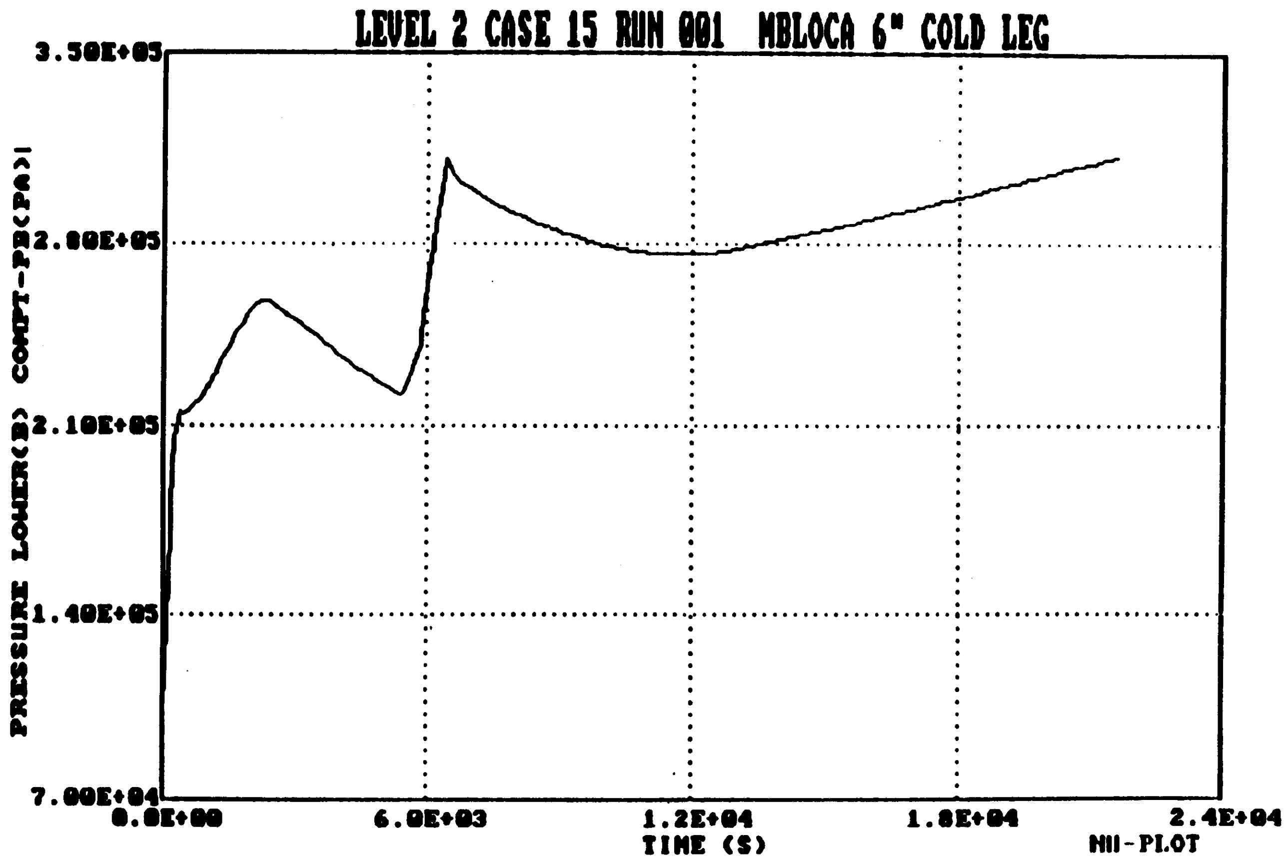


Figure 4.6.3-1 LTSBO Steam Generator Water Level

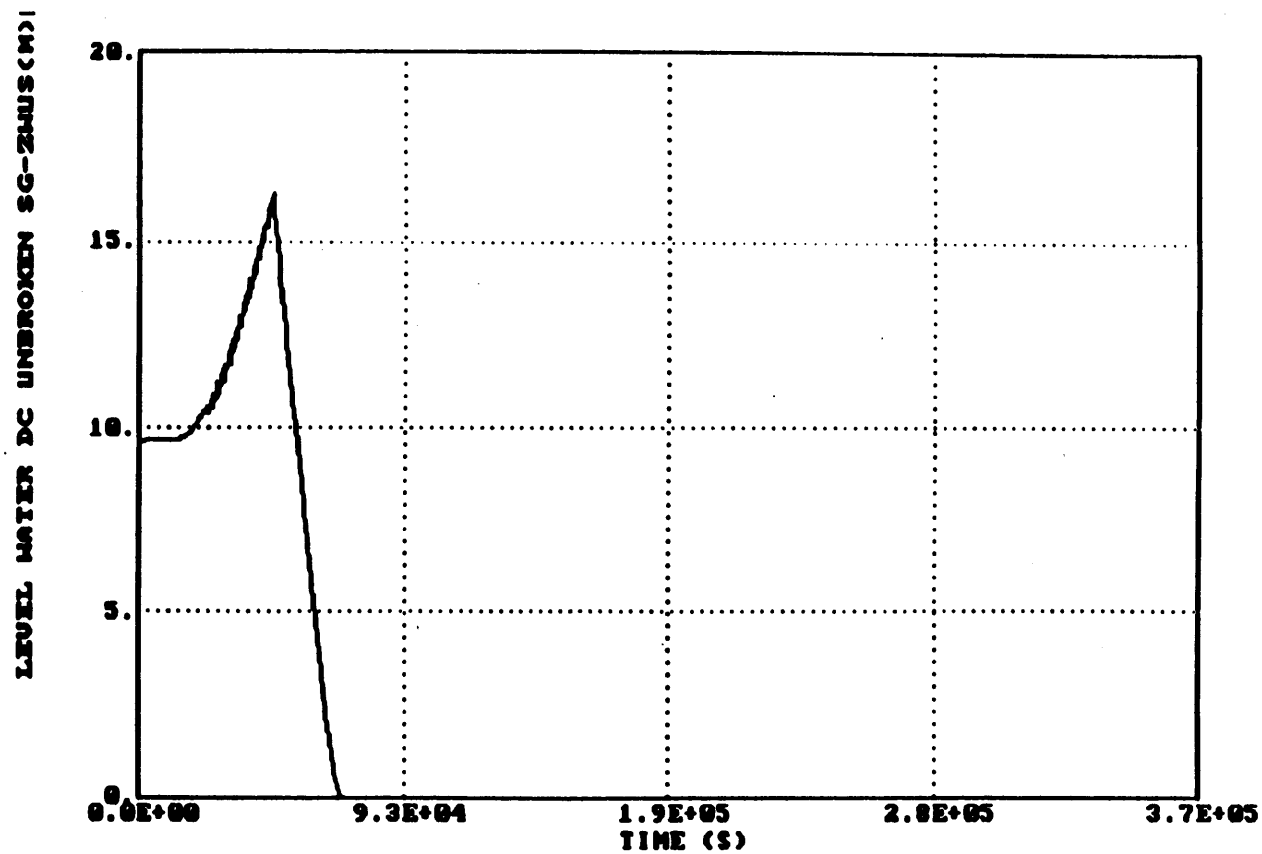


Figure 4.6.3-2 LTSBO RCS Pressure

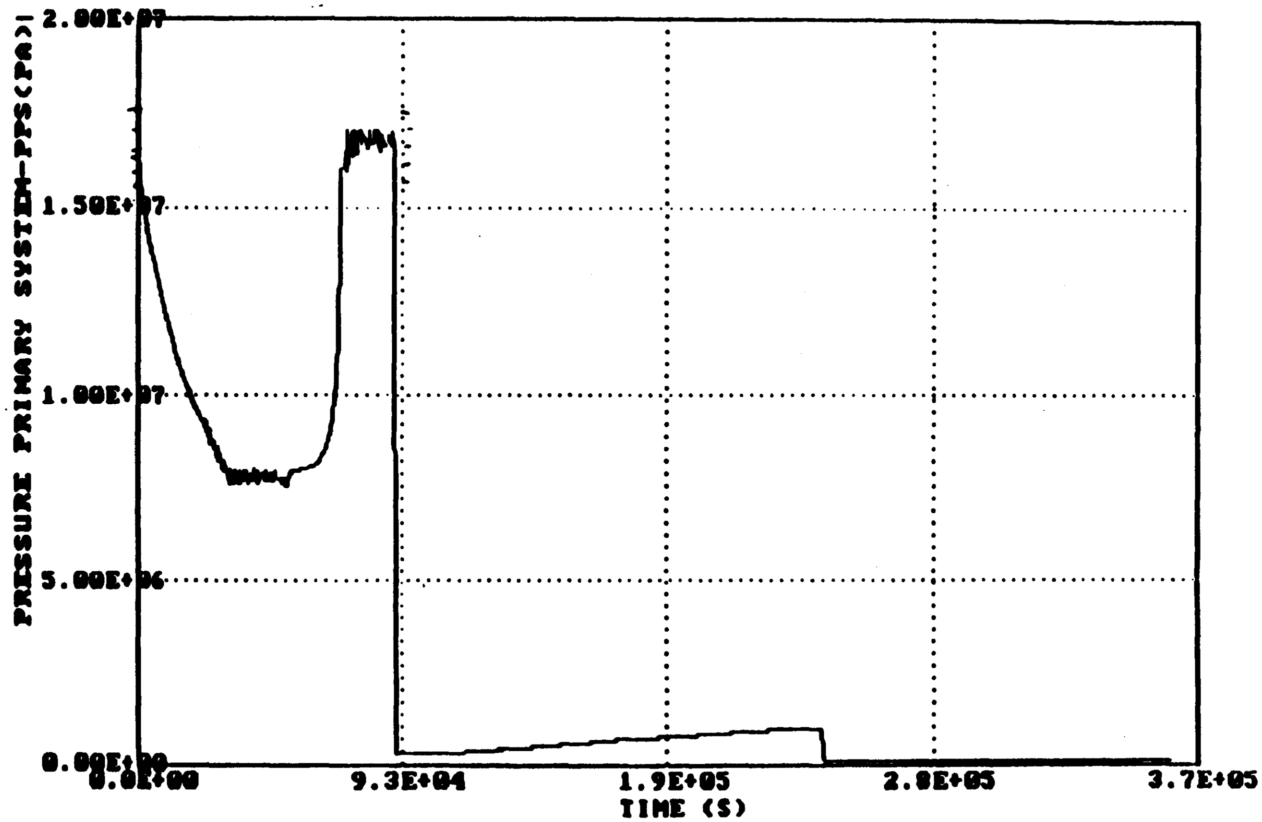


Figure 4.6.3-3 LTSBO RCS Water Temperature

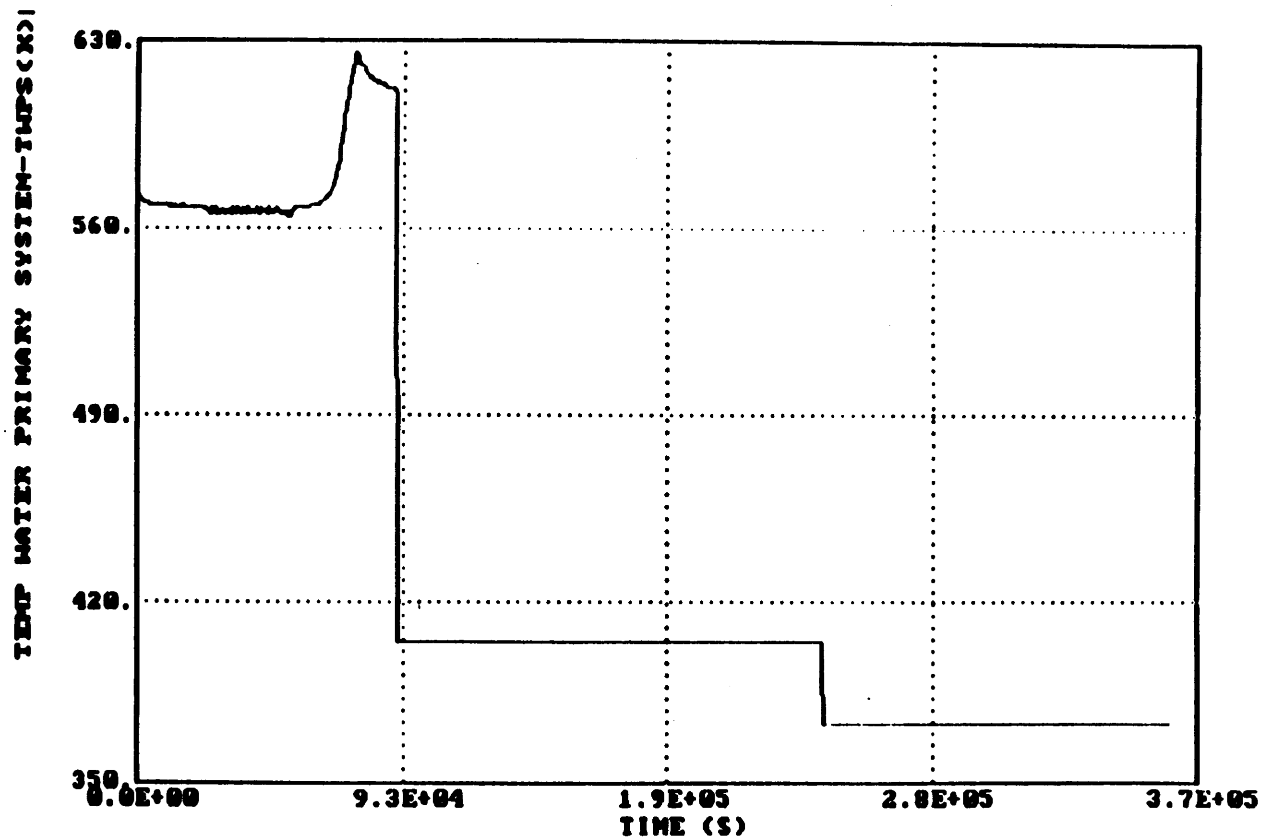


Figure 4.6.3-4 LTSBO Reactor Vessel Water Level

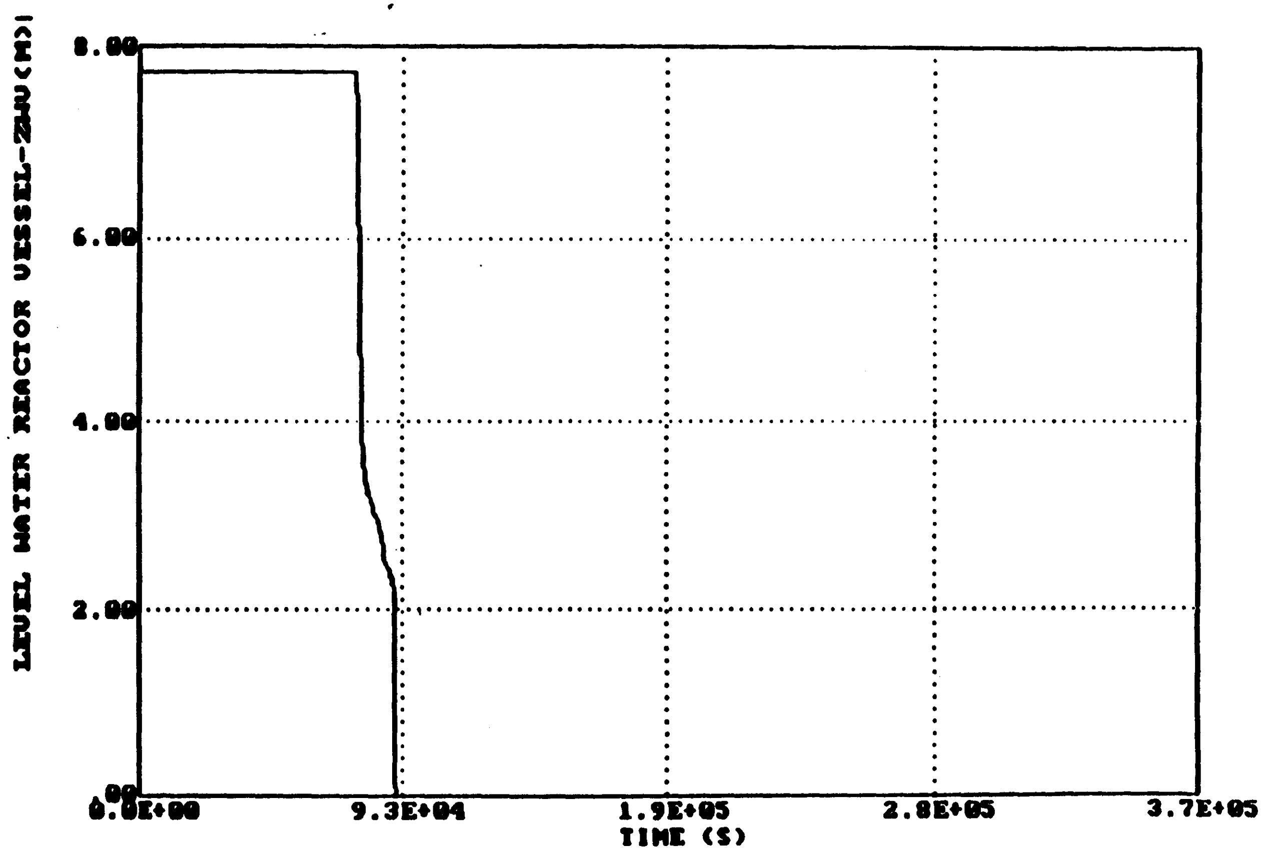


Figure 4.6.3-5 LTSBO In-Vessel Hydrogen Production

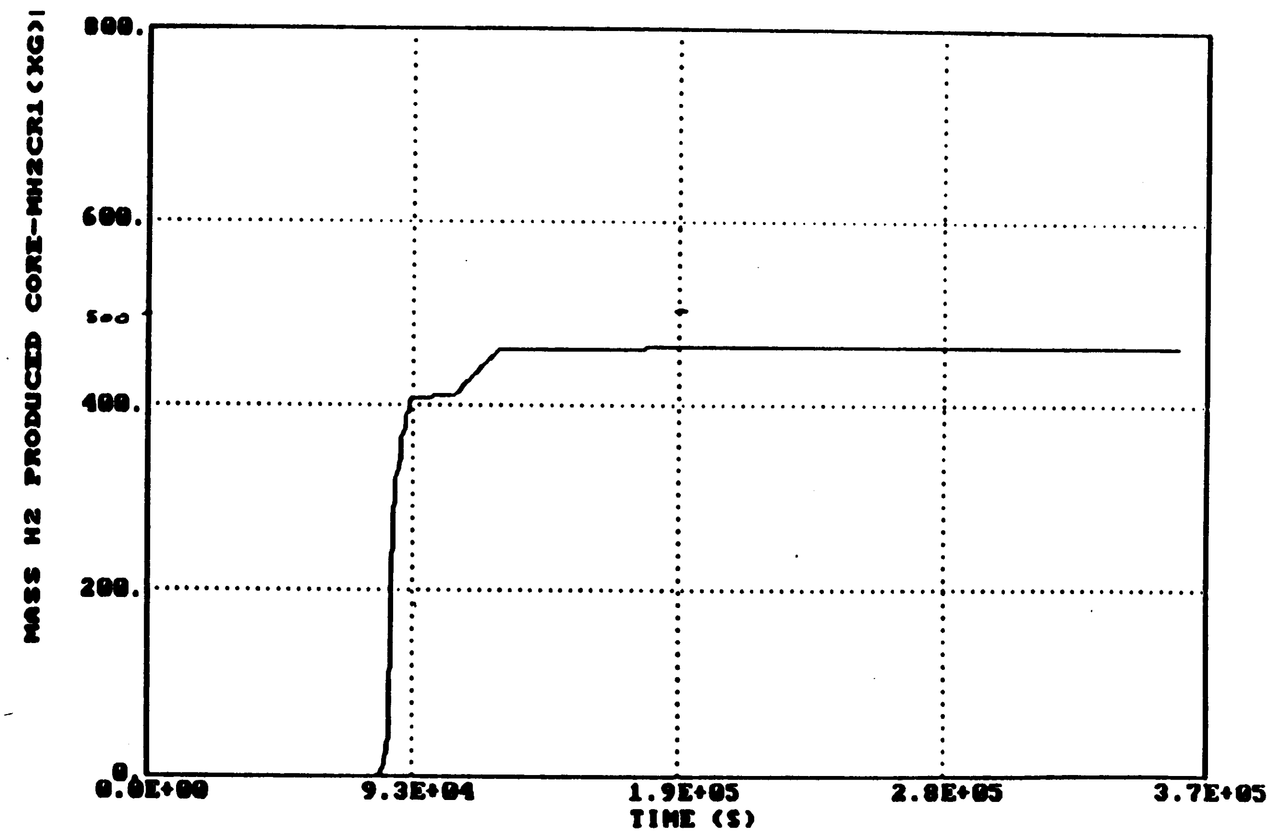


Figure 4.6.3-6 LTSBO Containment Pressure

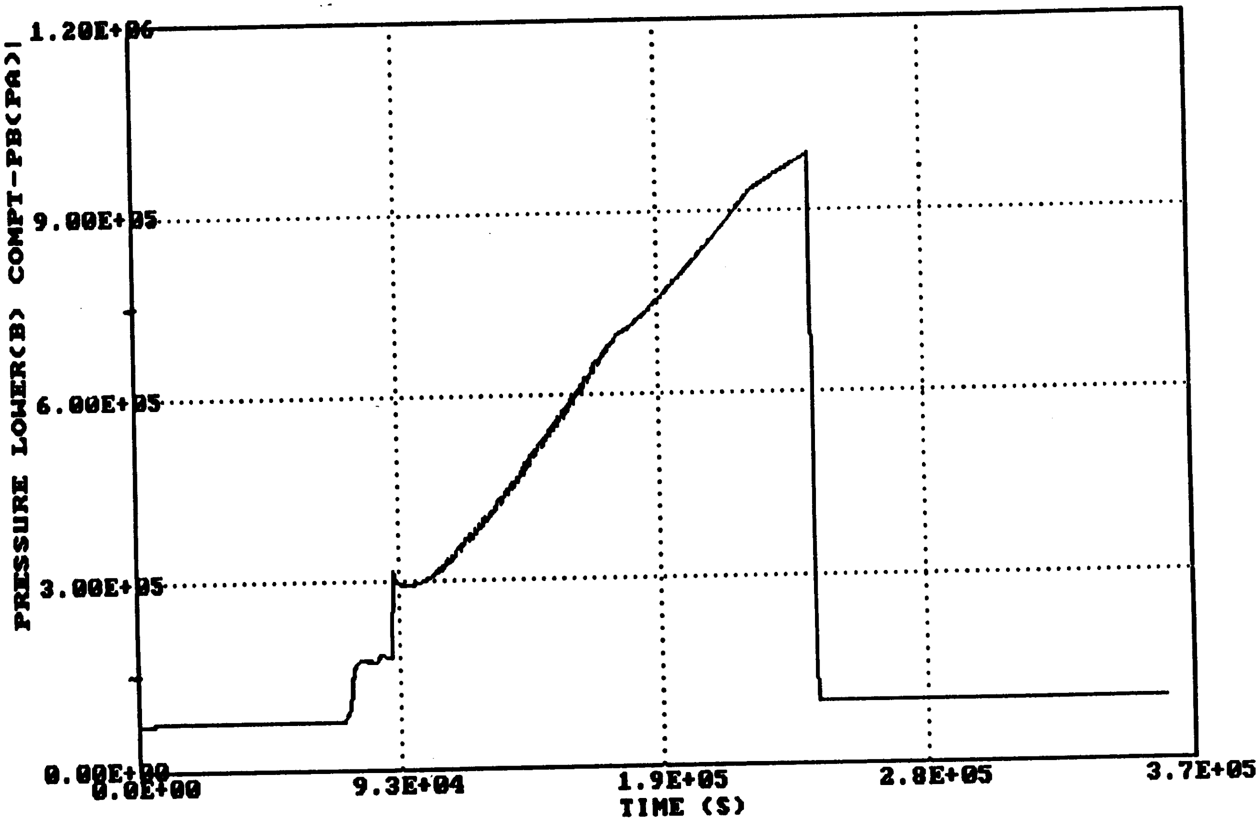


Figure 4.6.3-7 LTSBO CSI Release to Containment

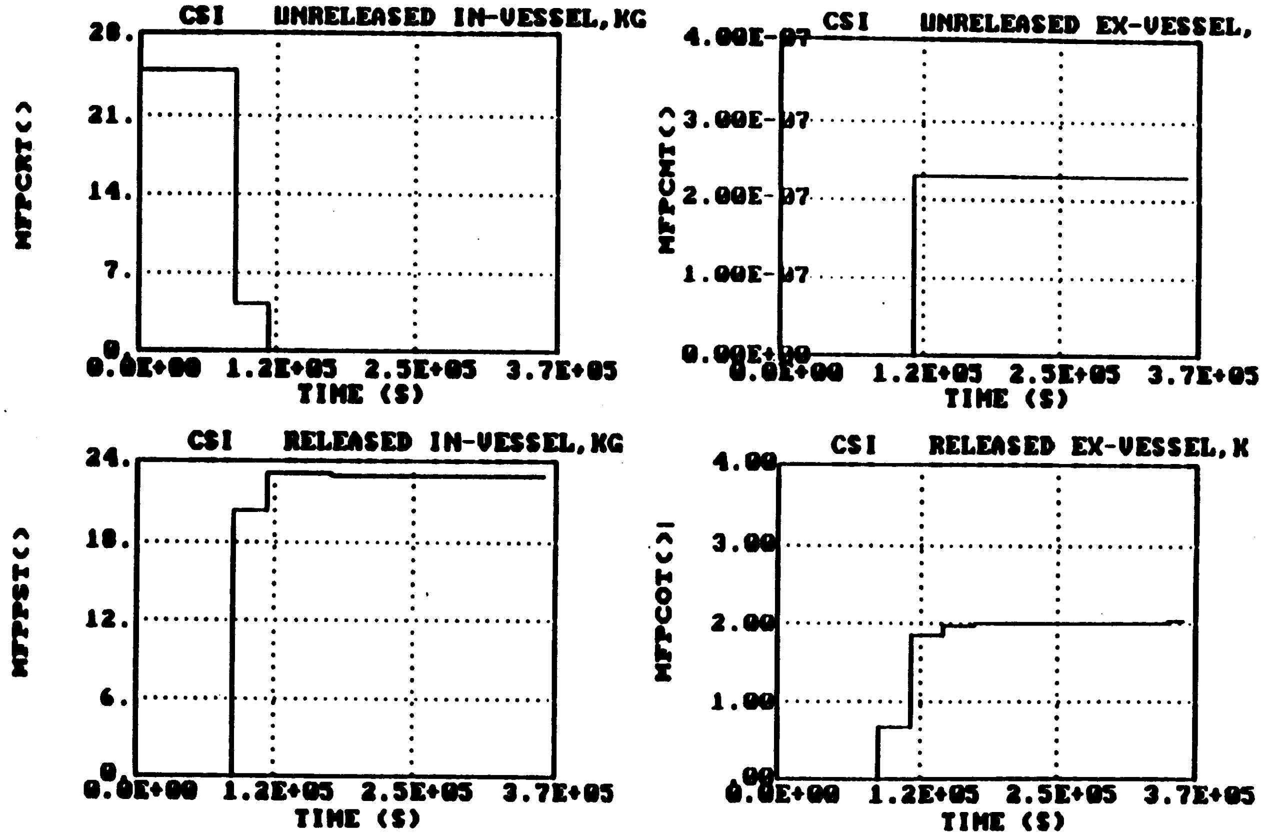


Figure 4.6.3-8 LTSBO CSOH Release to Containment

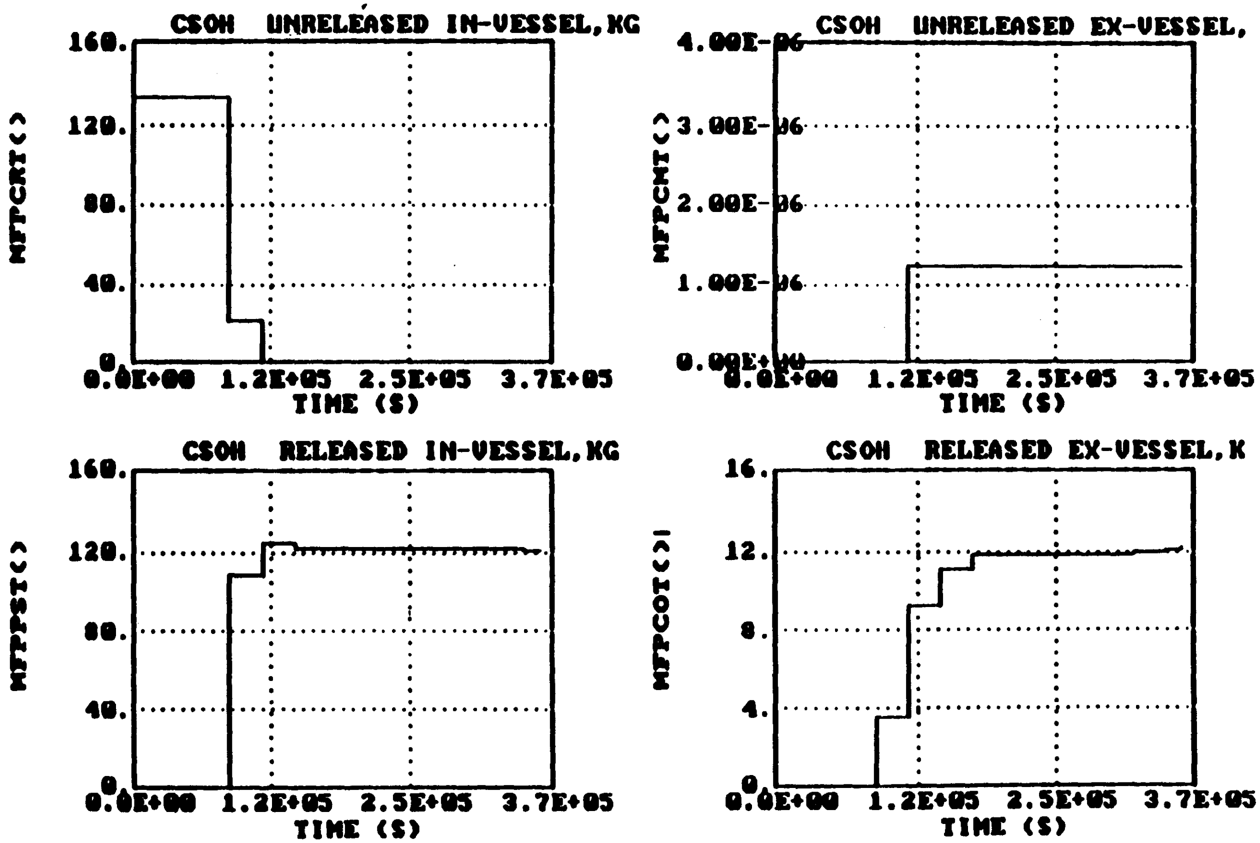


Figure 4.6.3-9 LTSBO LA203 Release to Containment

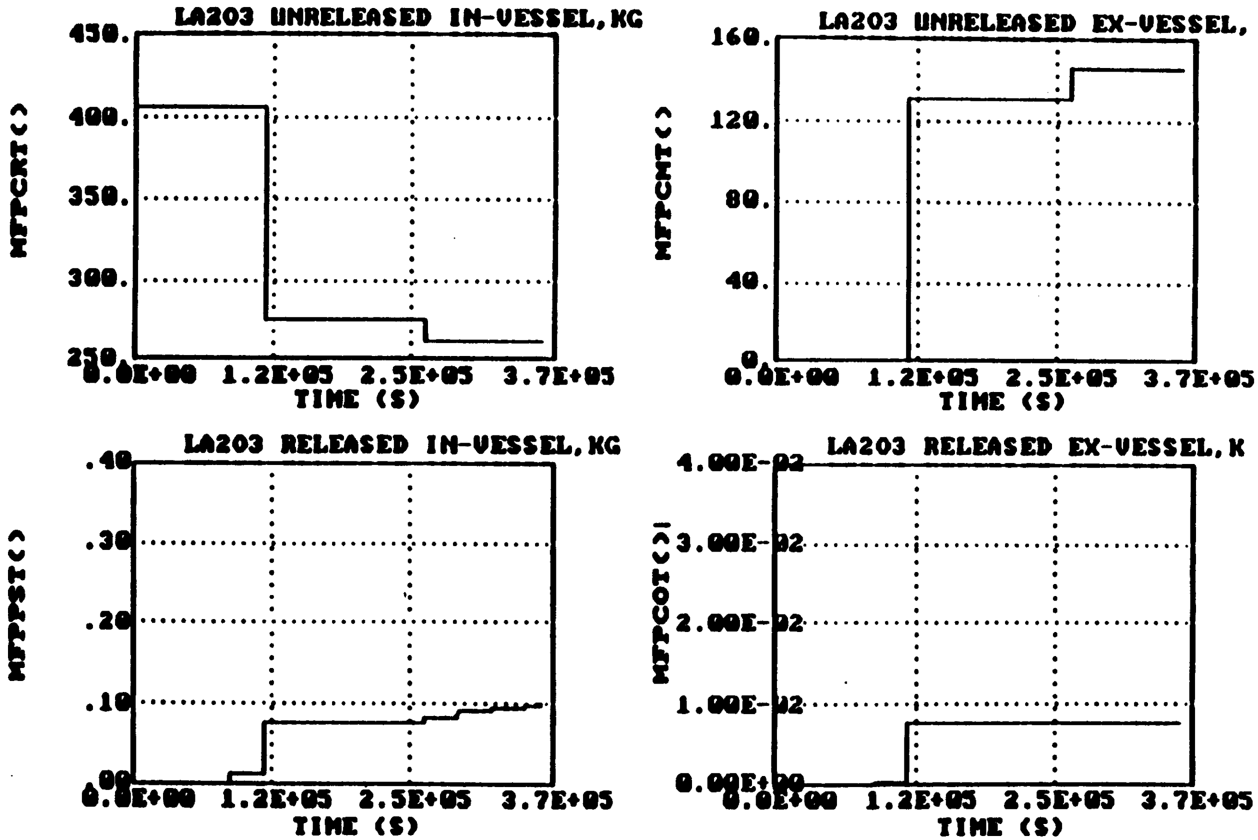


Figure 4.6.3-10 LTSBO SRO Release to Containment

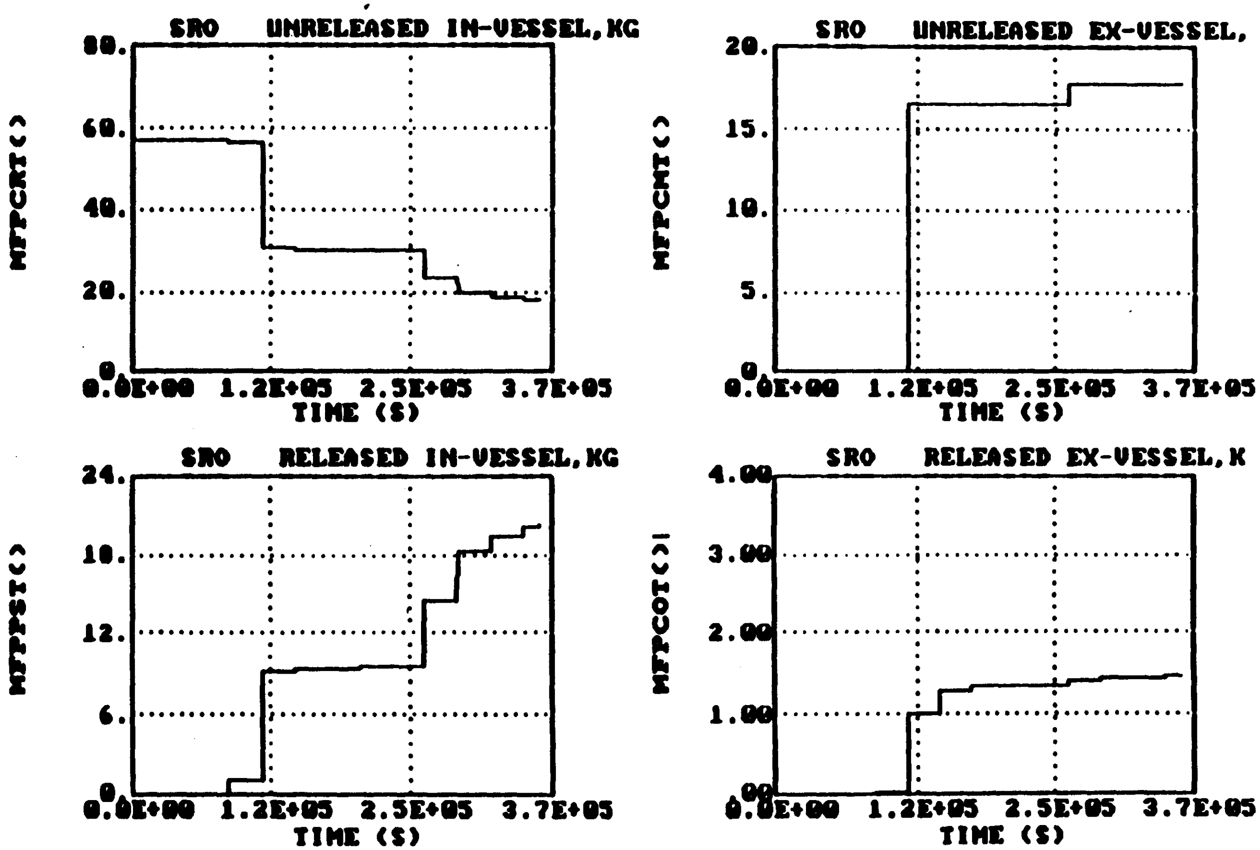


Figure 4.6.3-11 V Sequence Vessel Water Level

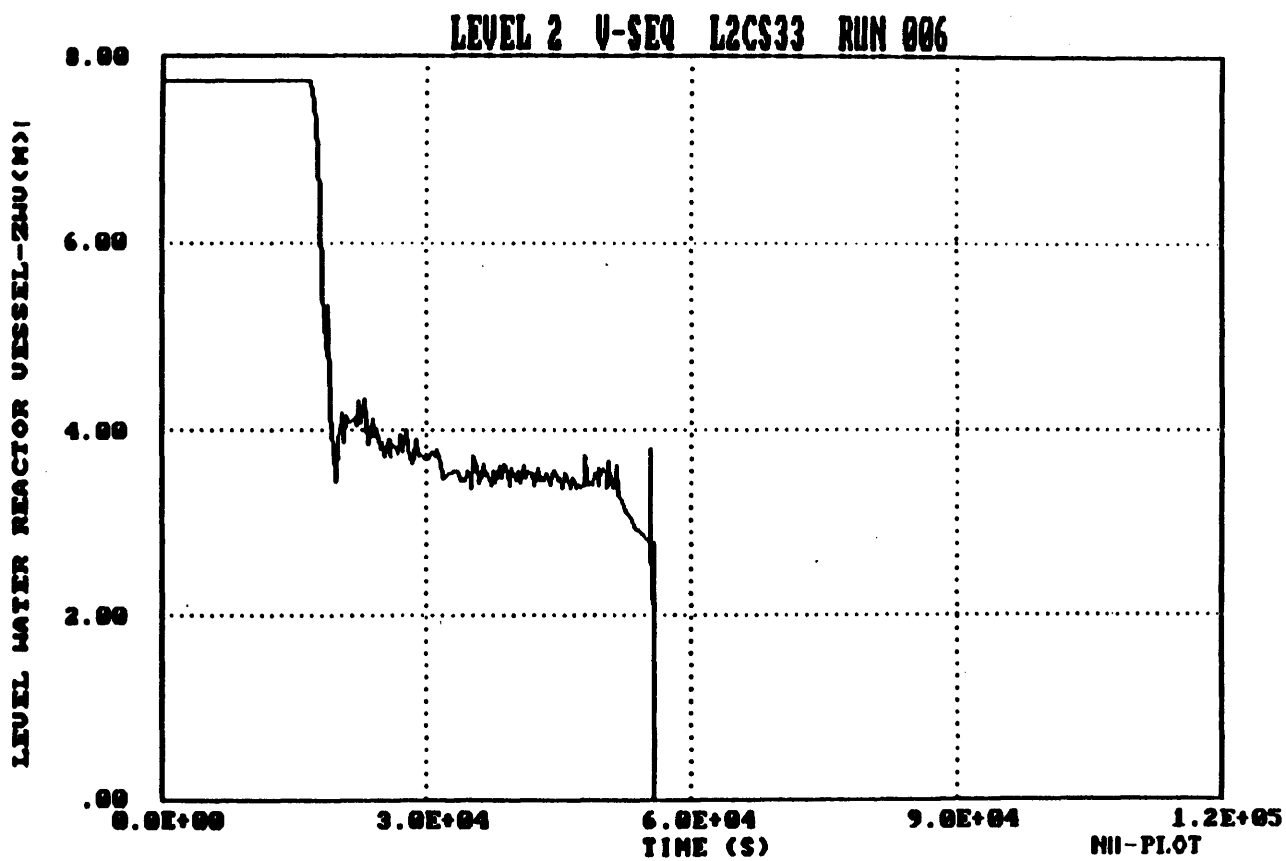


Figure 4.6.3-12 V Sequence RCS Pressure

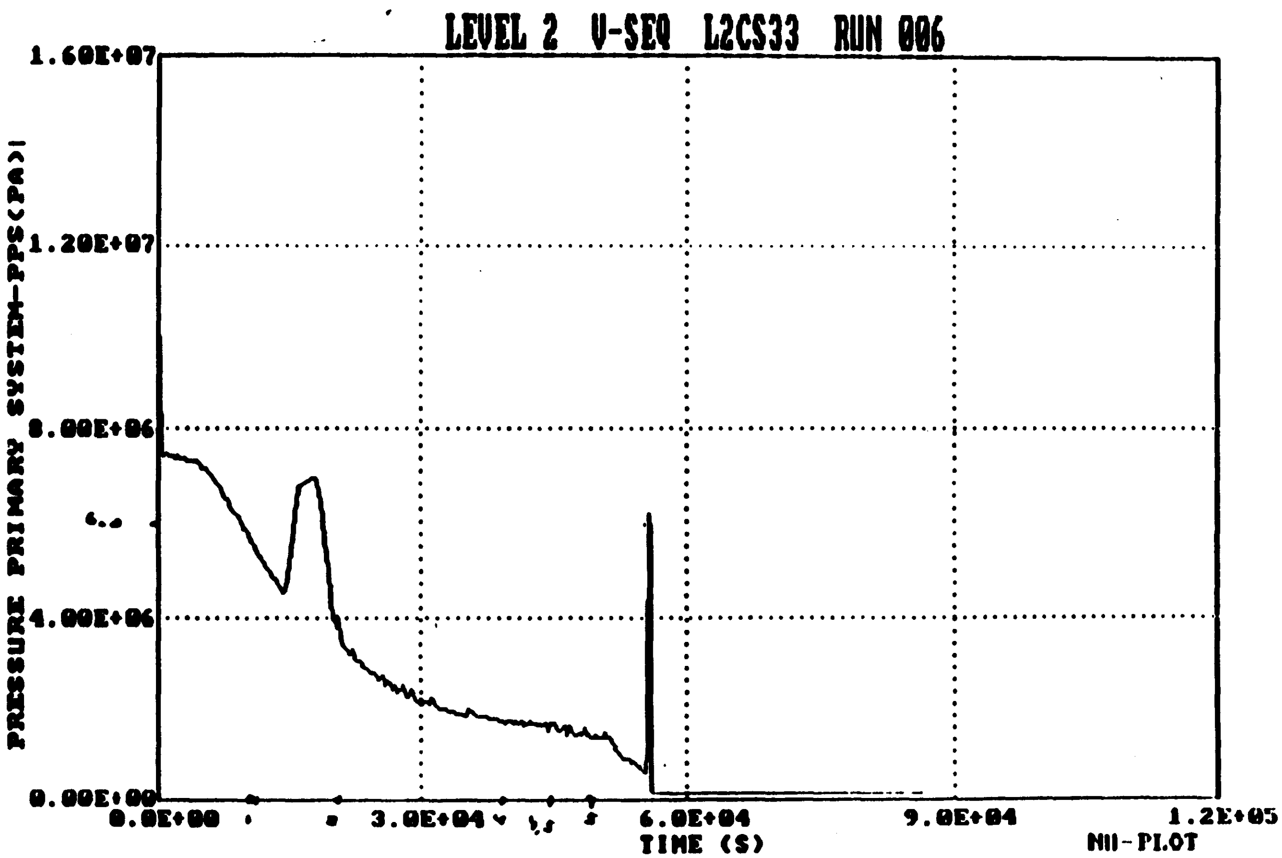


Figure 4.6.3-13 V Sequence RCS Water Temperature

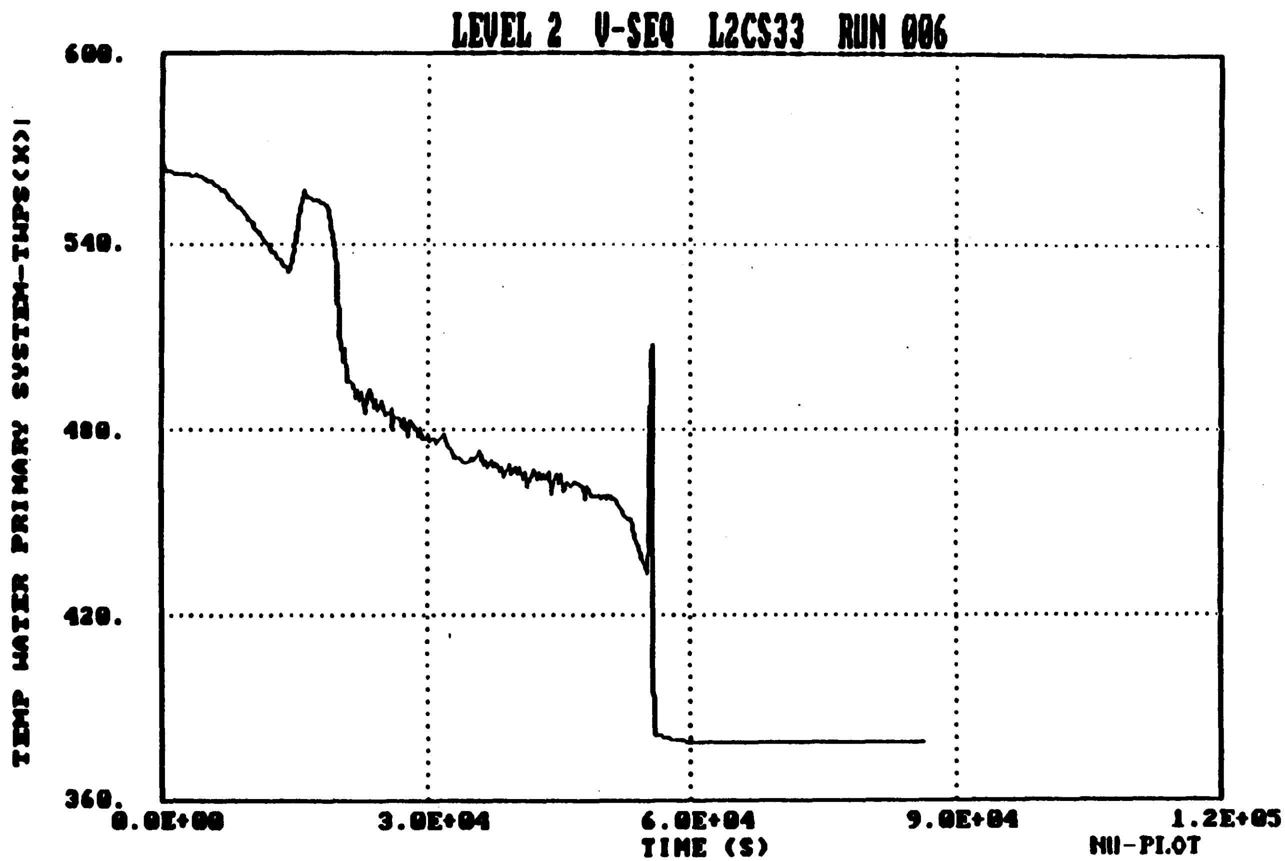


Figure 4.6.3-14 V Sequence Temperature of Hottest Core Node

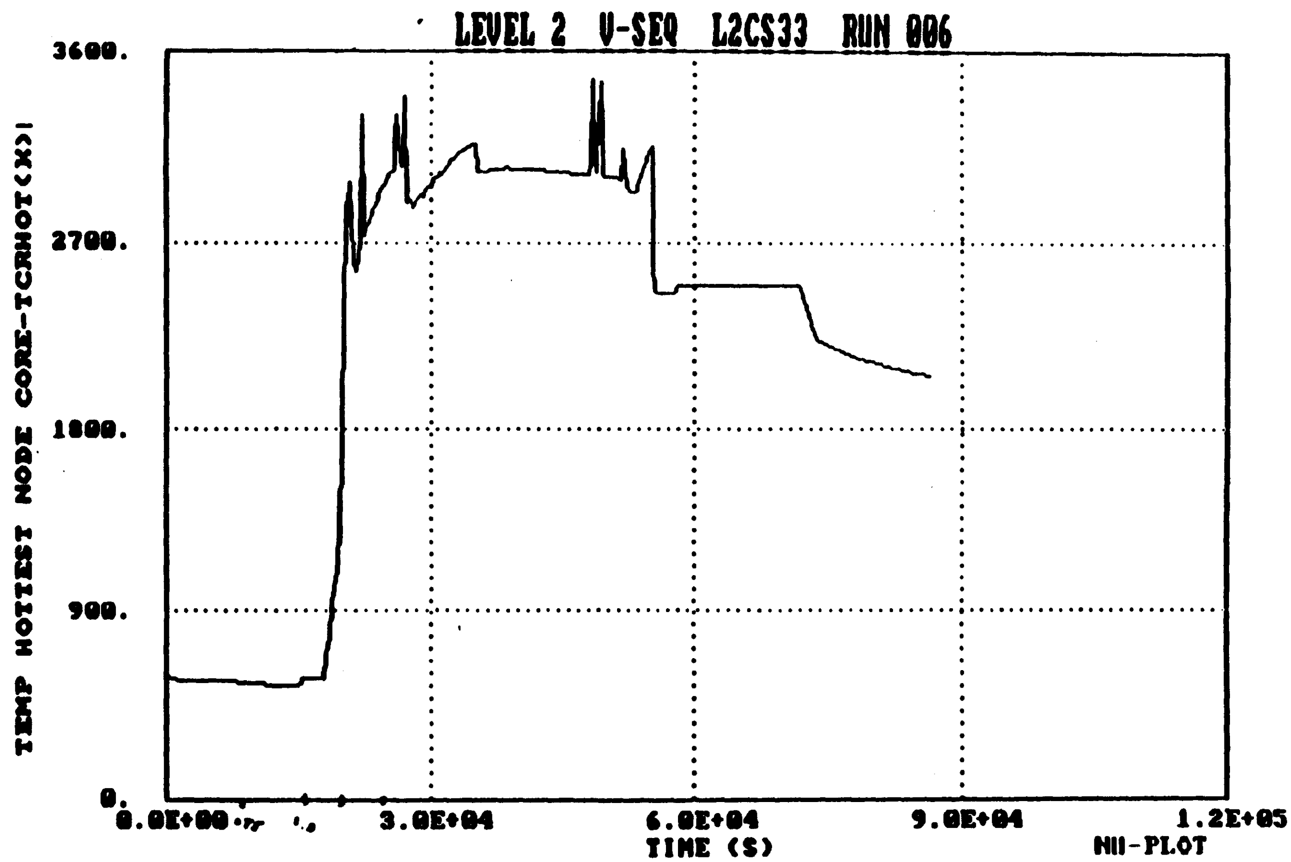


Figure 4.6.3-15 V Sequence Containment Pressure

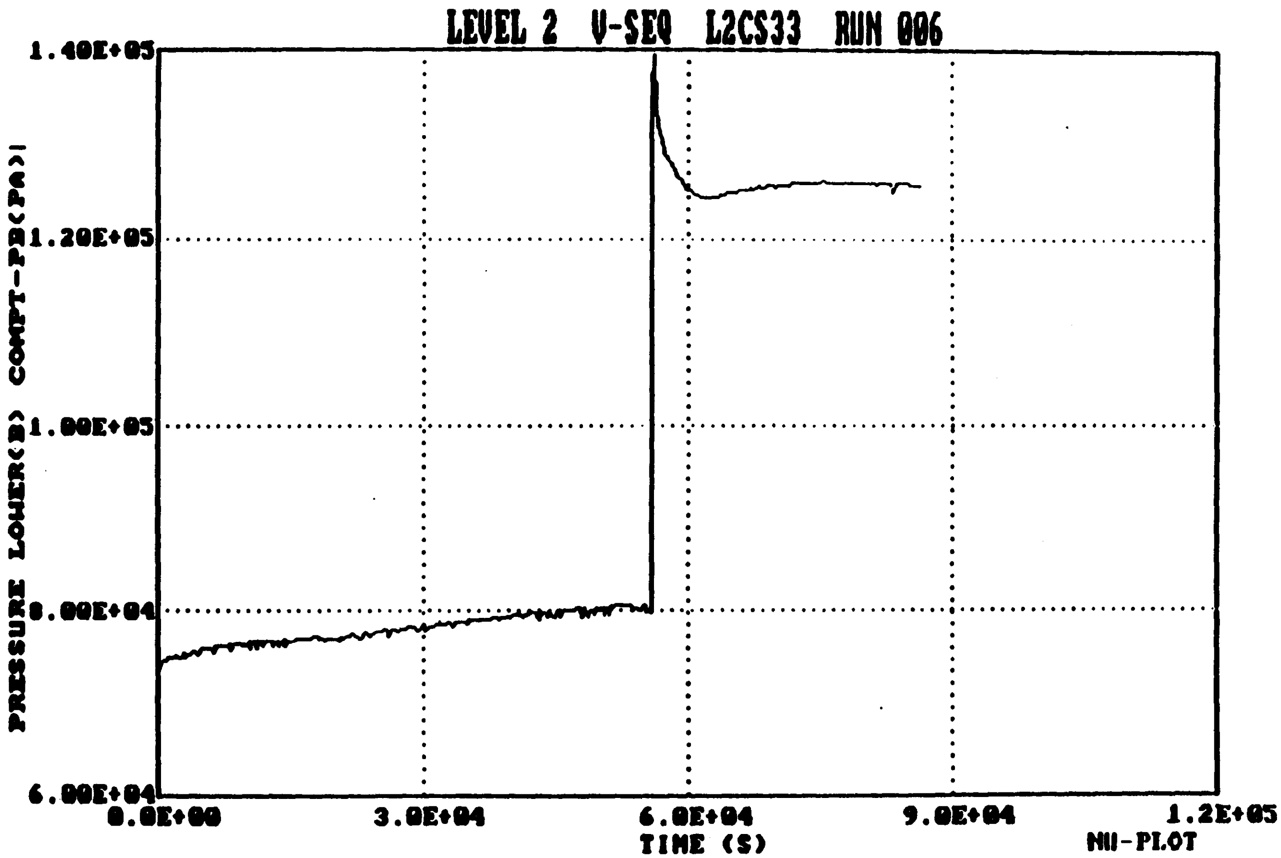


Figure 4.6.3-16 SGTR Broken and Unbroken SG Water Levels

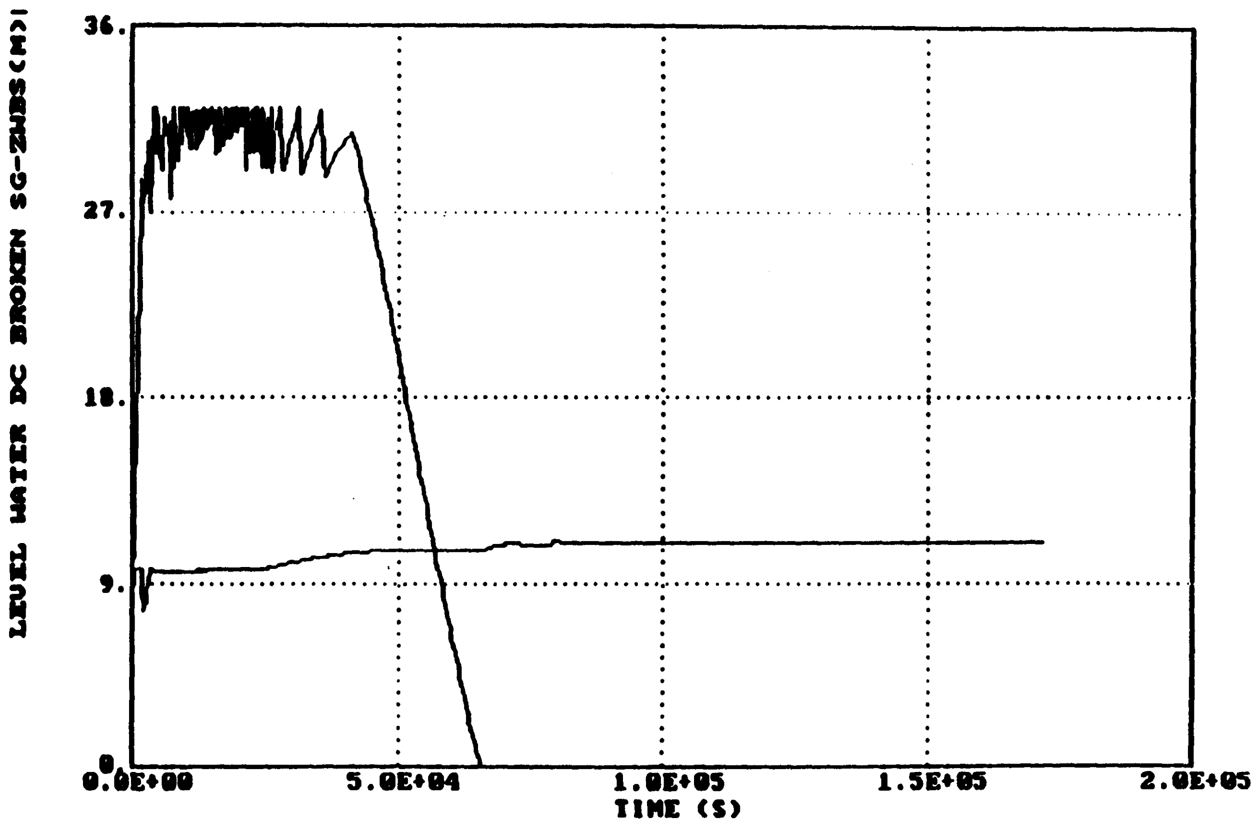


Figure 4.6.3-17 SGTR RCS Pressure

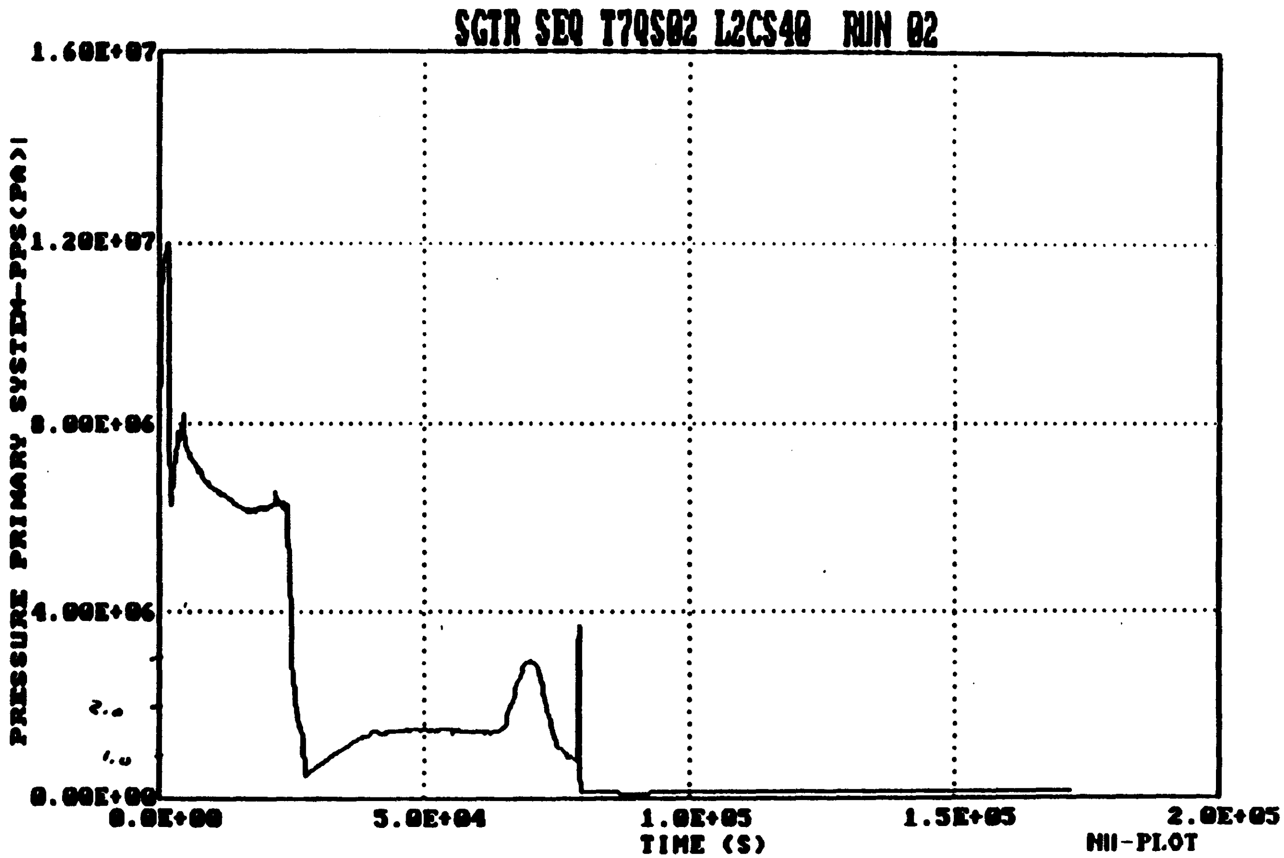


Figure 4.6.3-18 SGTR Vessel Water Temperature

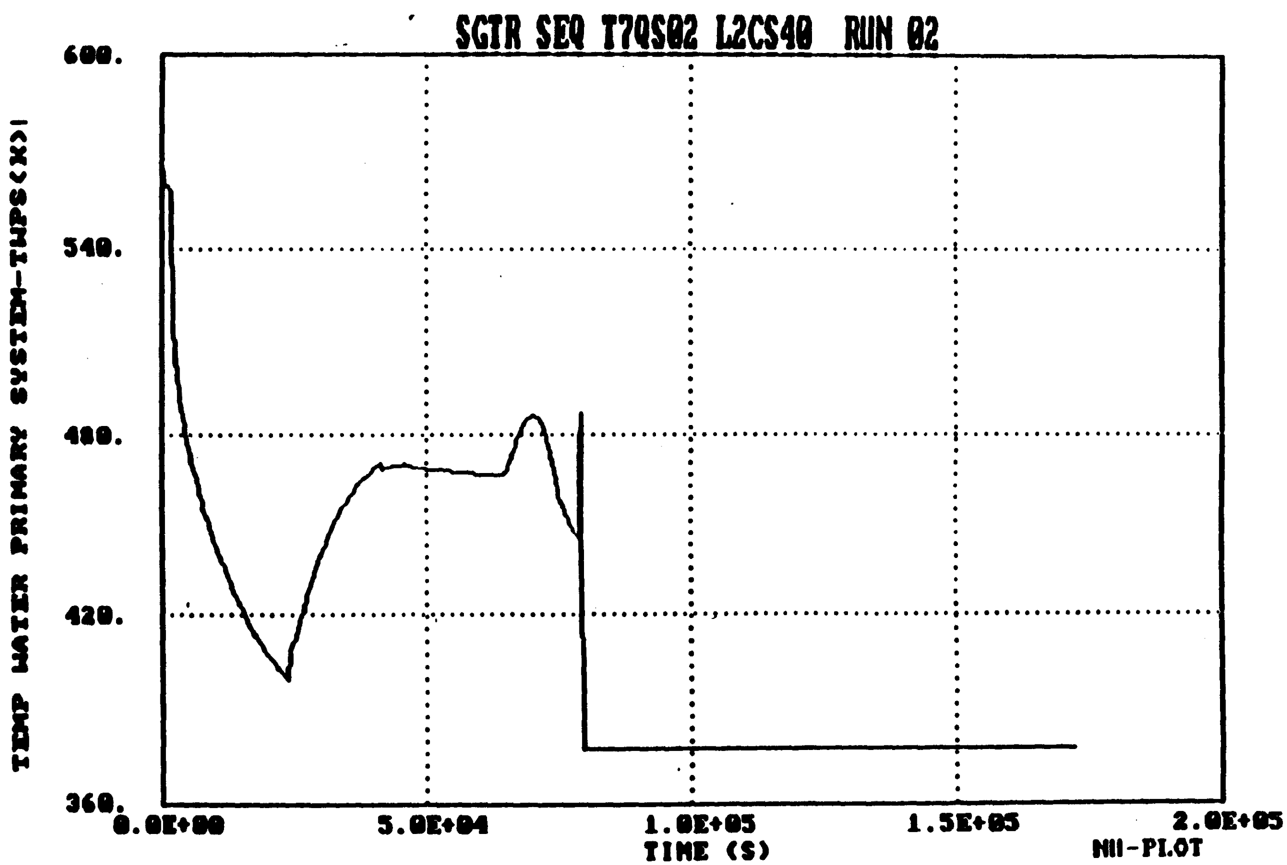


Figure 4.6.3-19 SGTR: Vessel Water Level

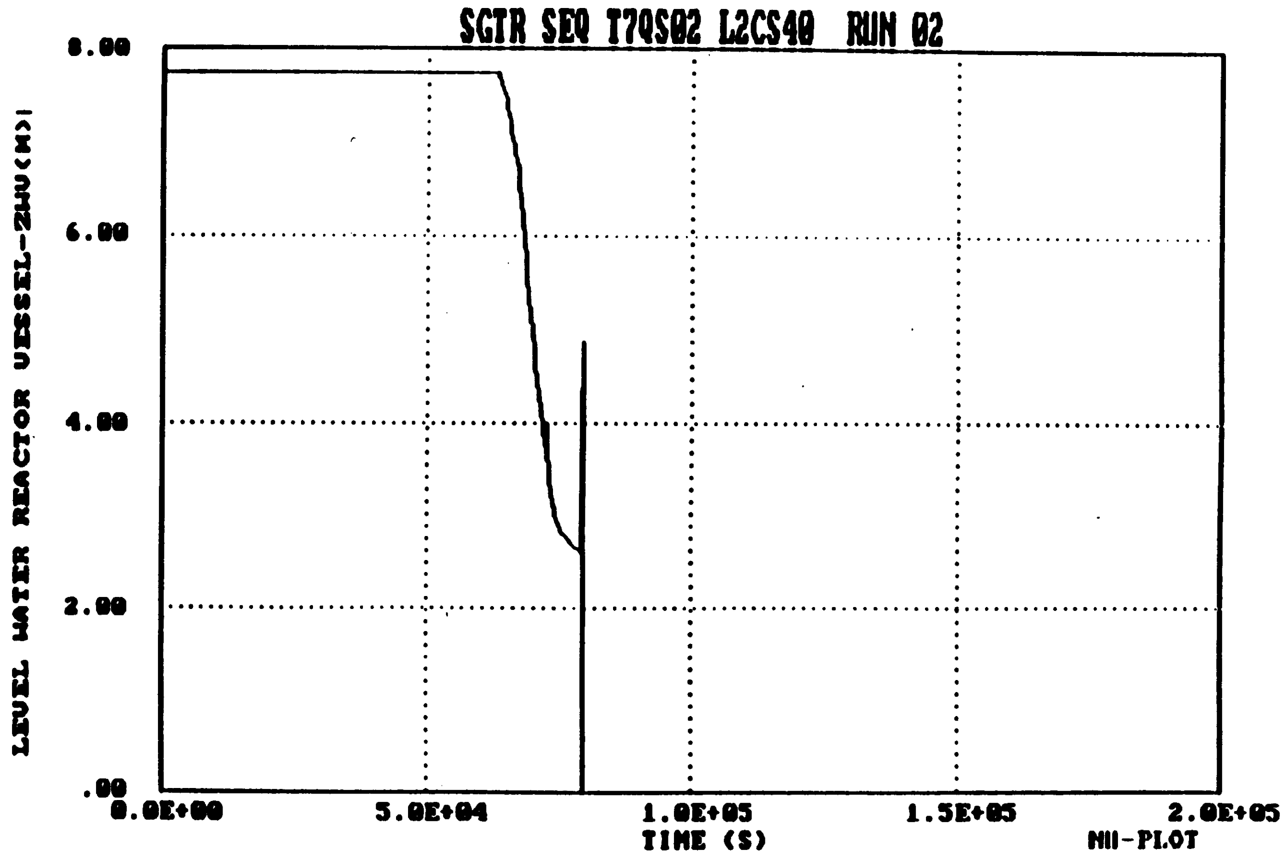


Figure 4.6.3-20 SGTR Broken SG Gas Temperature

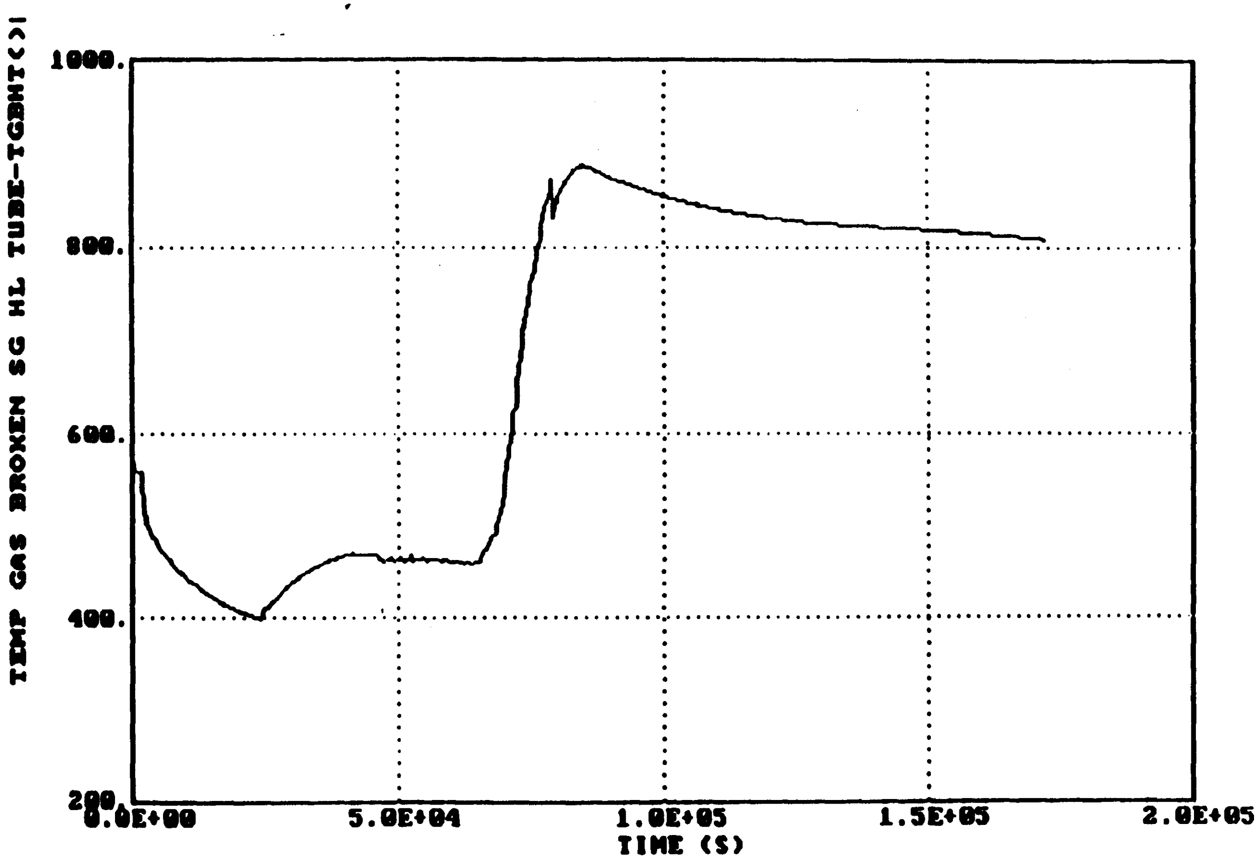
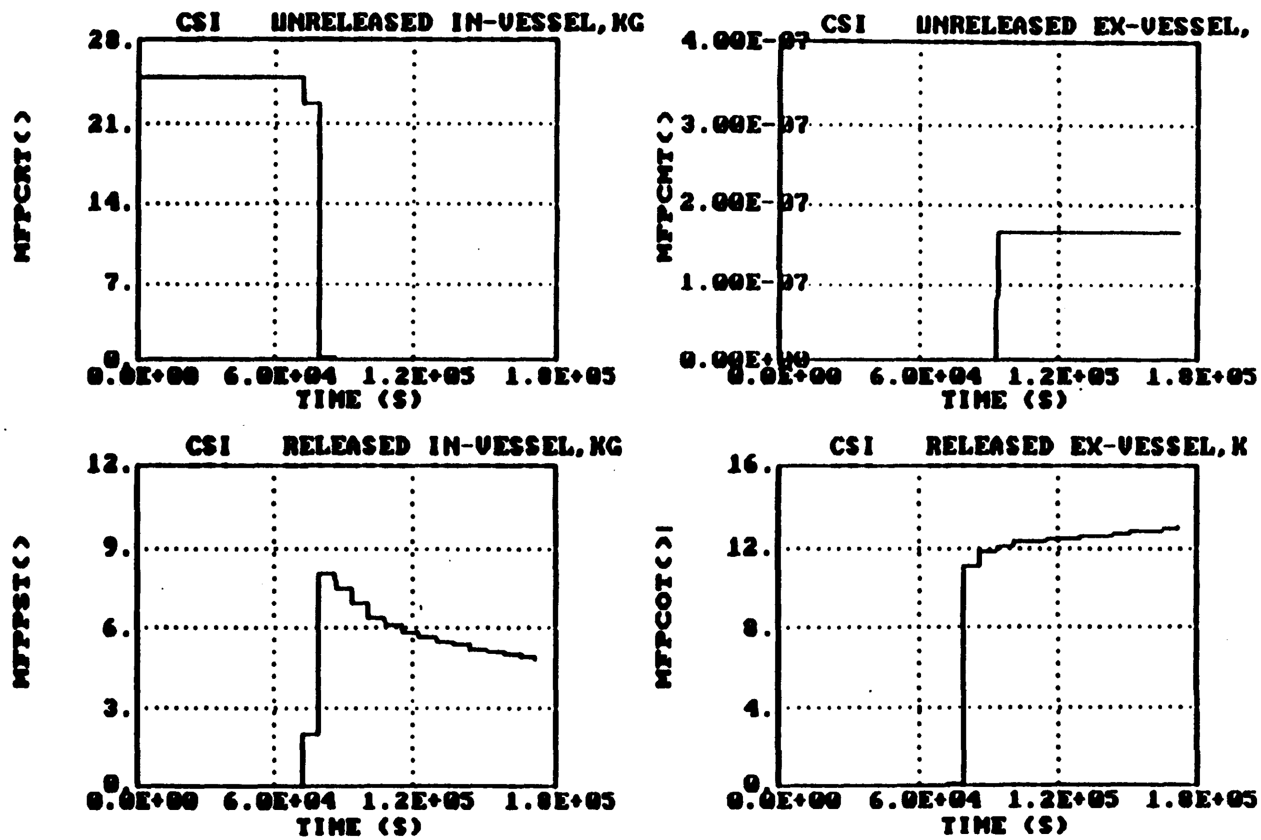


Figure 4.6.3-21 SGTR CSI Release to Containment



Figures 4.6.3-22 SGTR CSOH Release to Containment

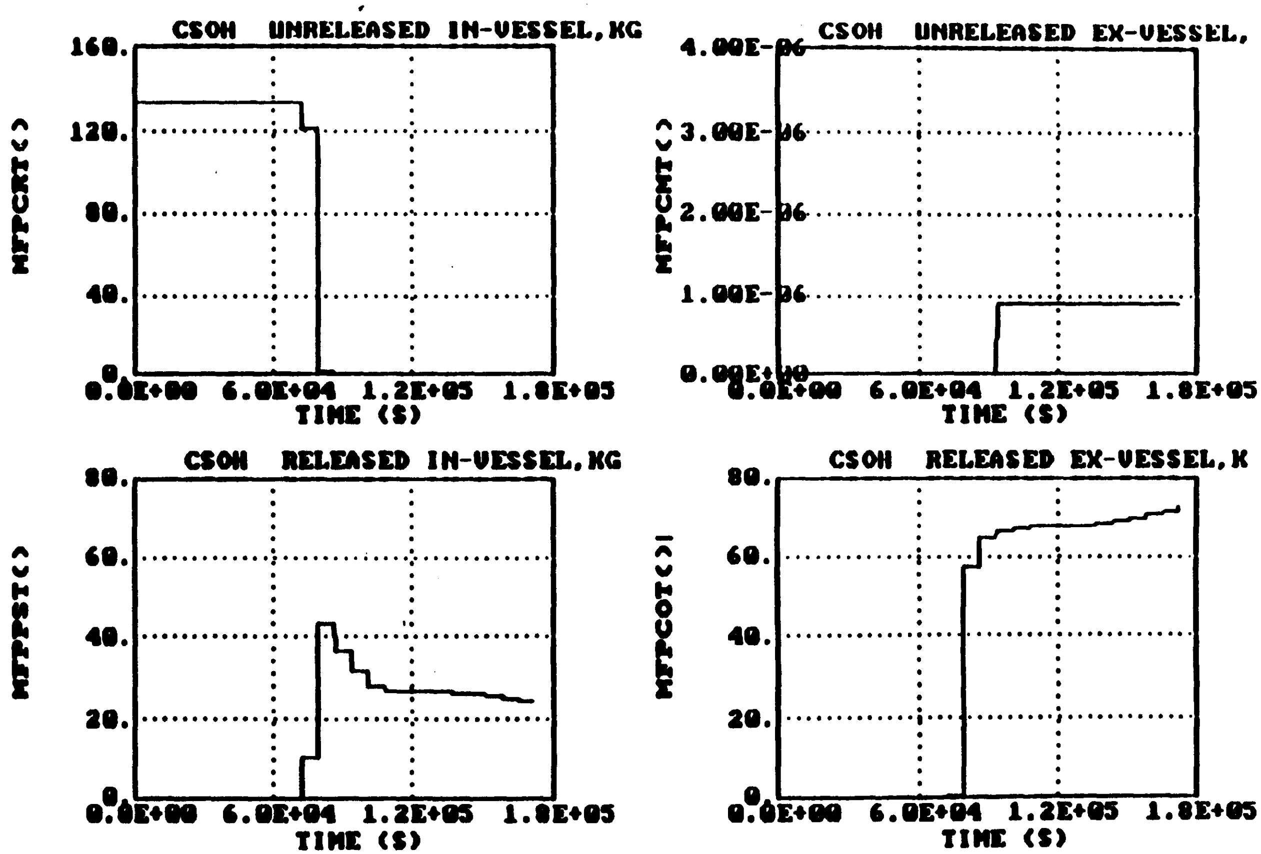


Figure 4.6.3-23 SGTR SRO Release to Containment

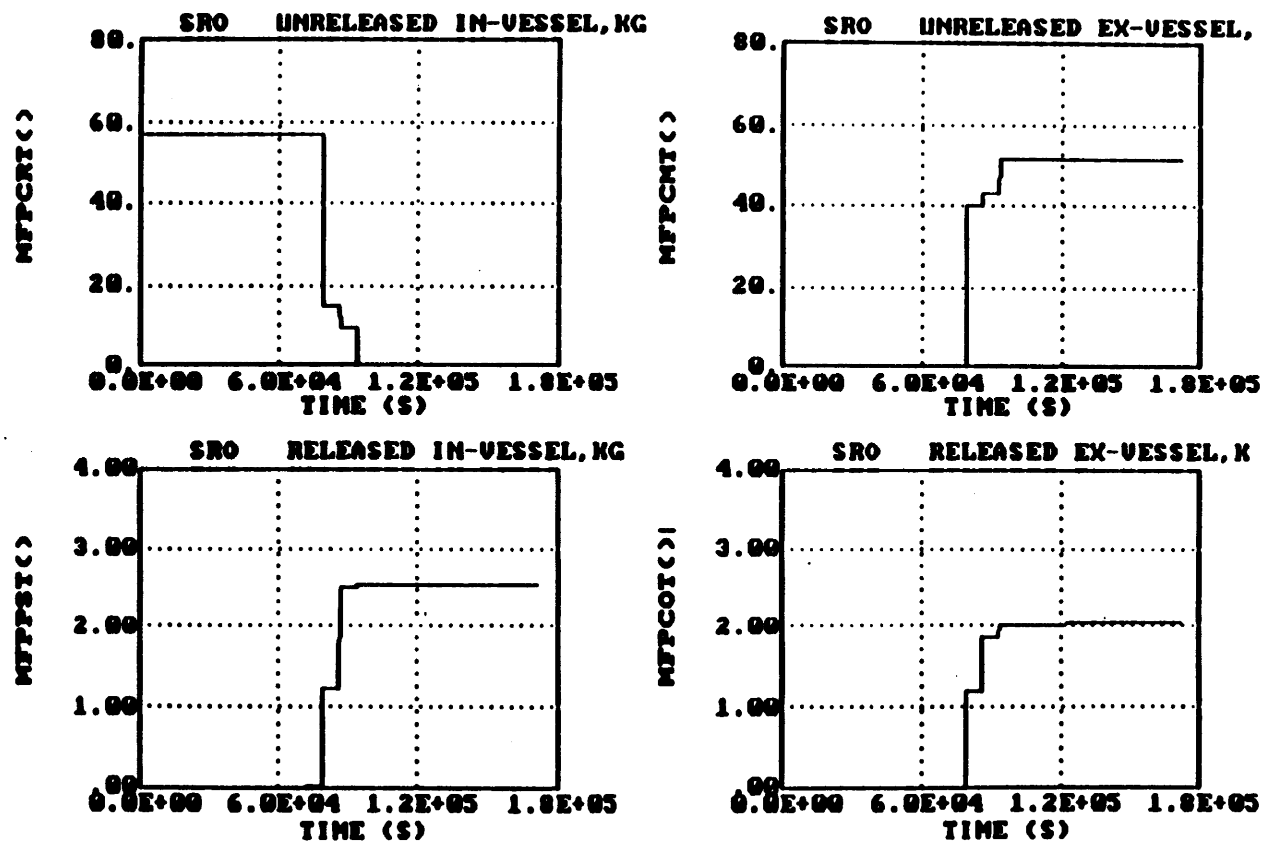


Figure 4.6.3-24 SGTR LA203 Release to Containment

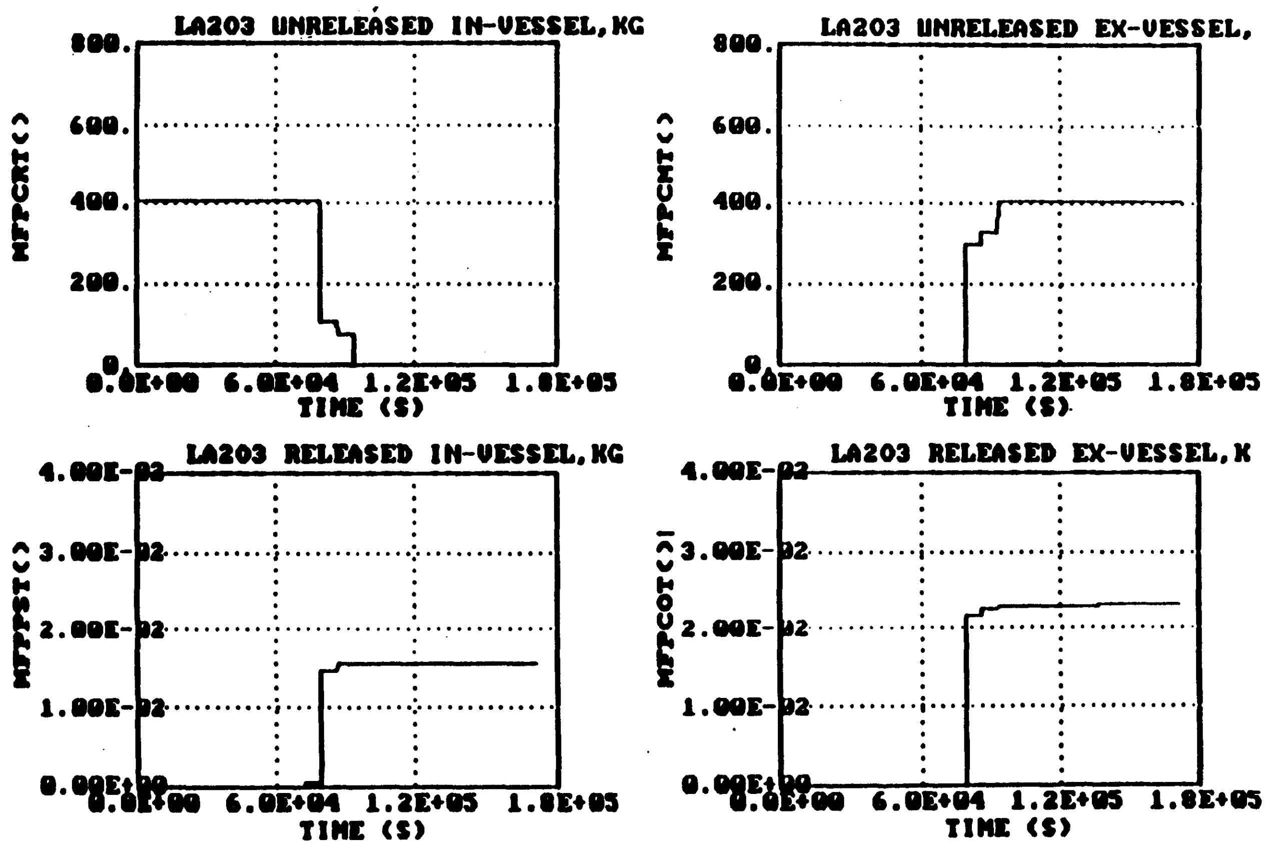


DIAGRAM: REVONAPS.STD 30 SEP 92 DATA FILE: 27 SEP 92 Sum = 6.797E-005

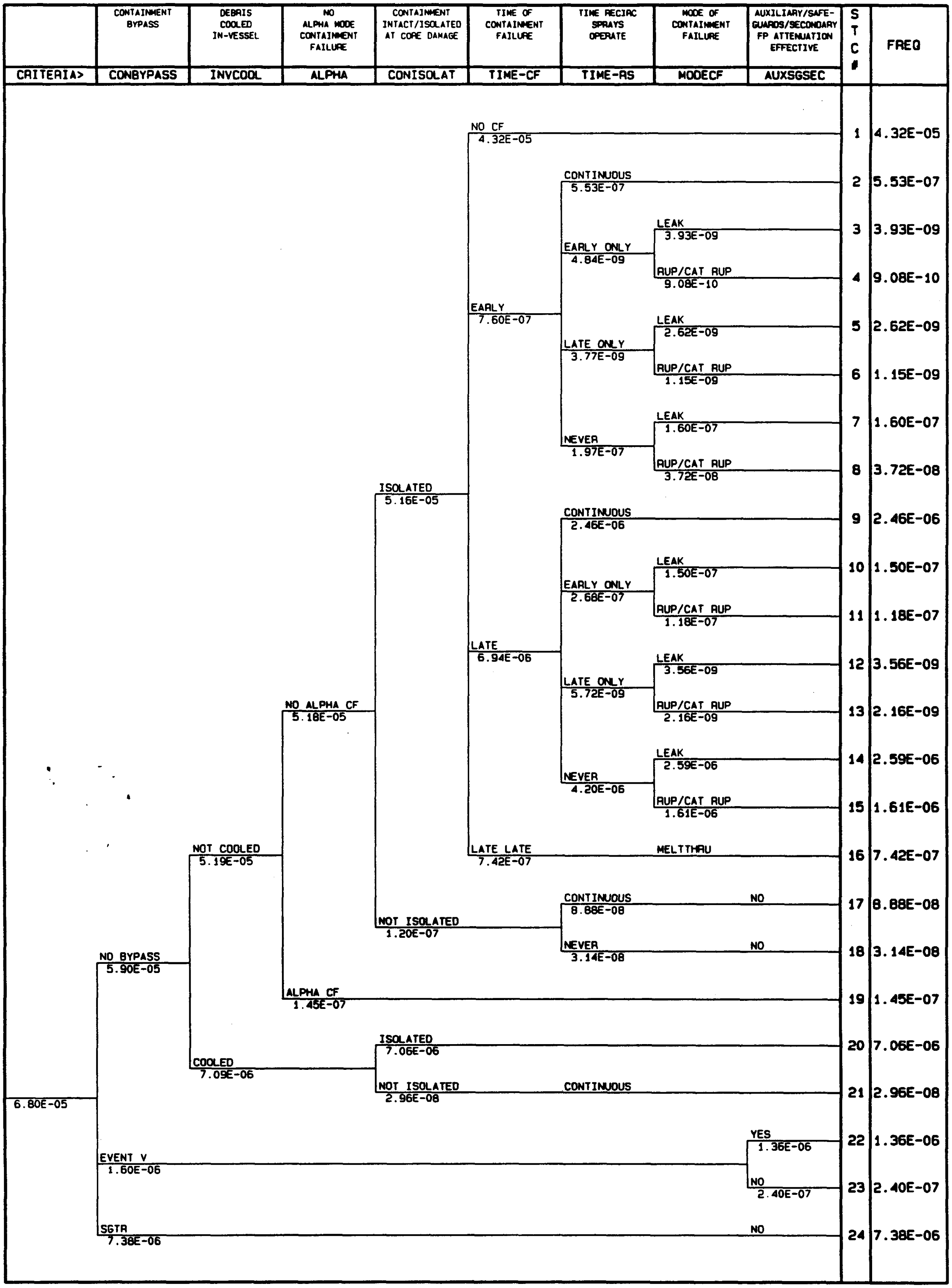


FIGURE 4.7.2-1 Source Term Logic Diagram

Figure 4.7.3-1
Comparison of MAAP and NUREG-1150
Release Fractions for Late Failure

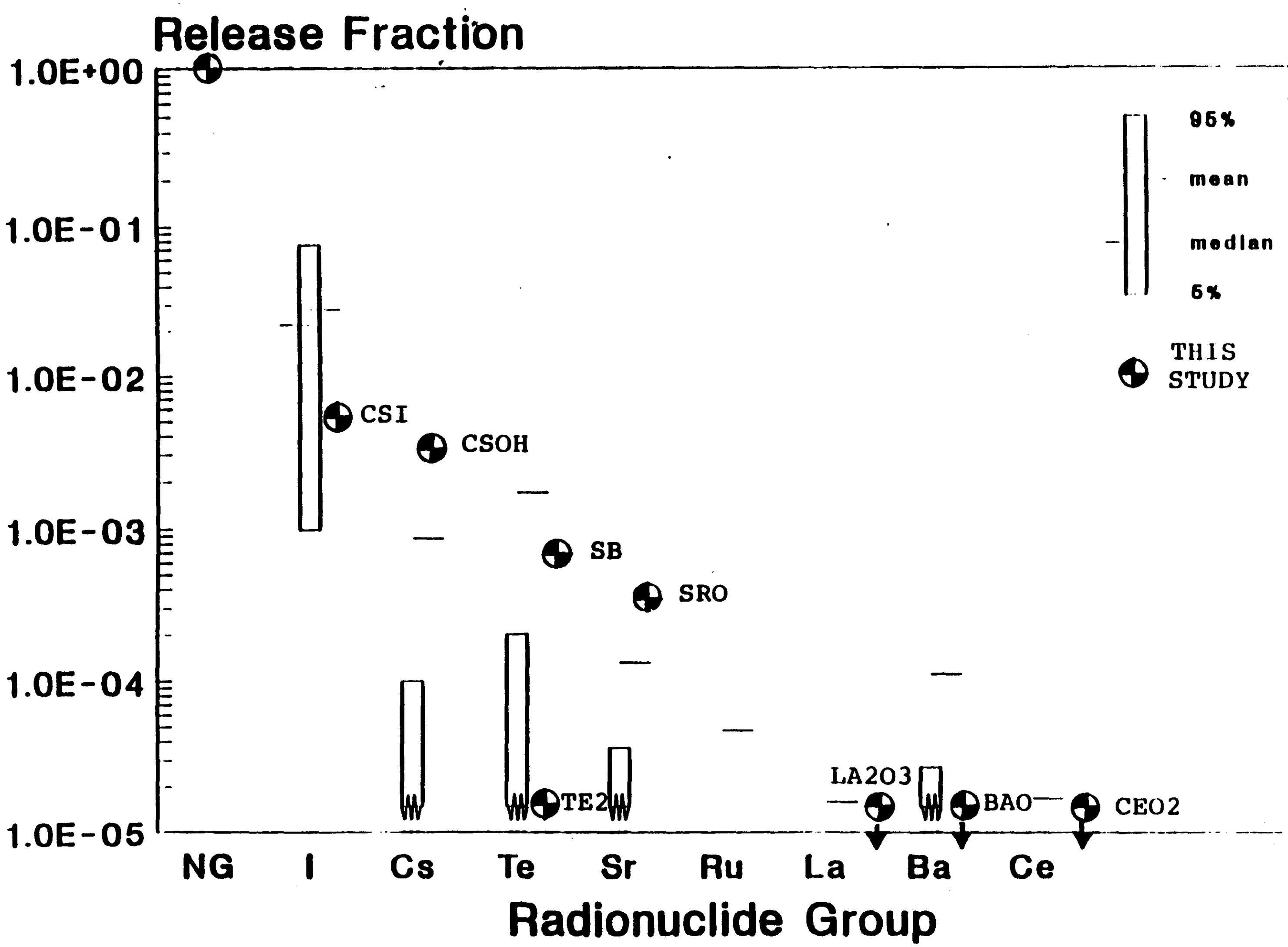
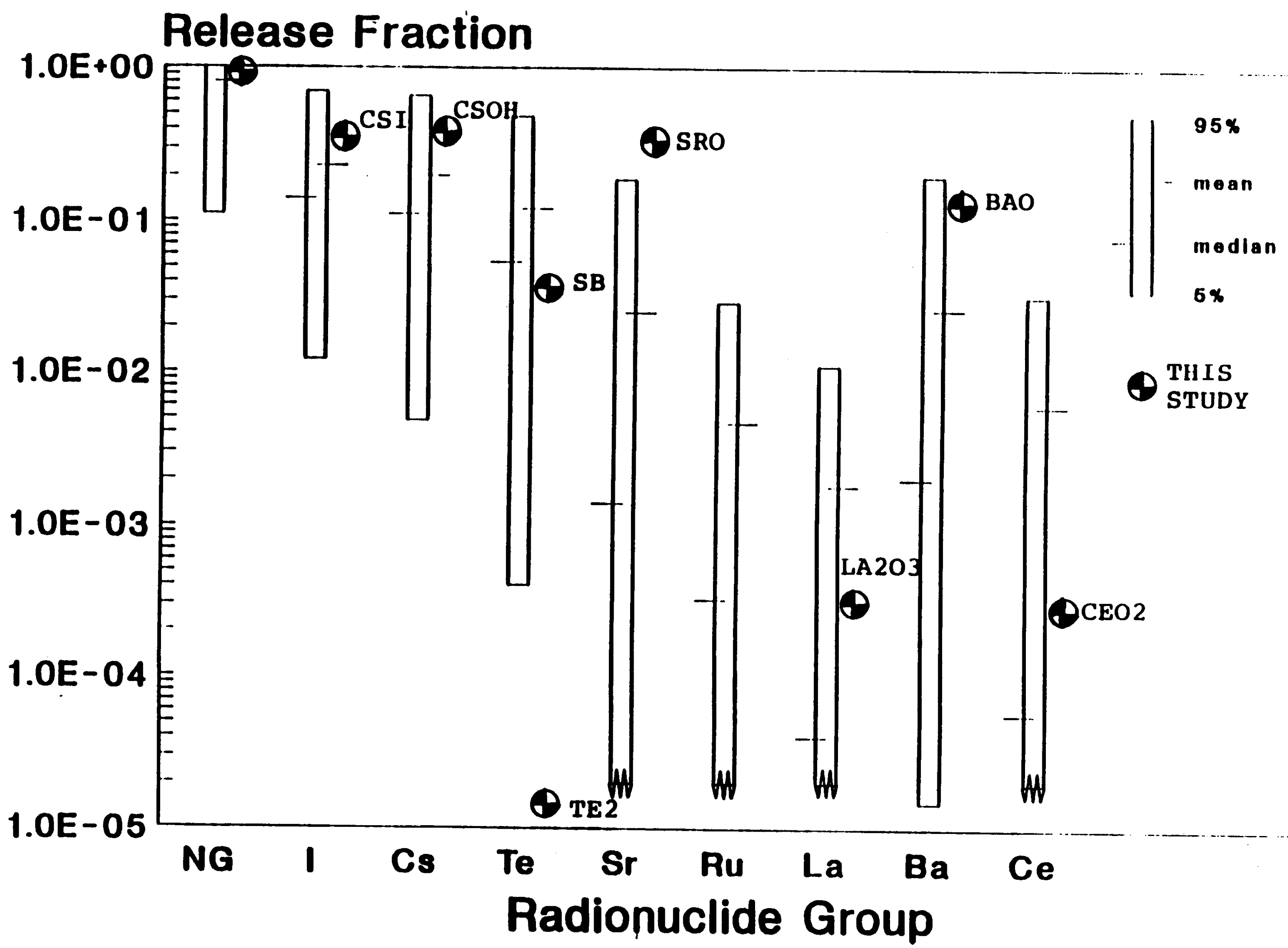


Figure 4.7.3-2
Comparison of MAAP and NUREG-1150
Release Fractions for Containment Bypass



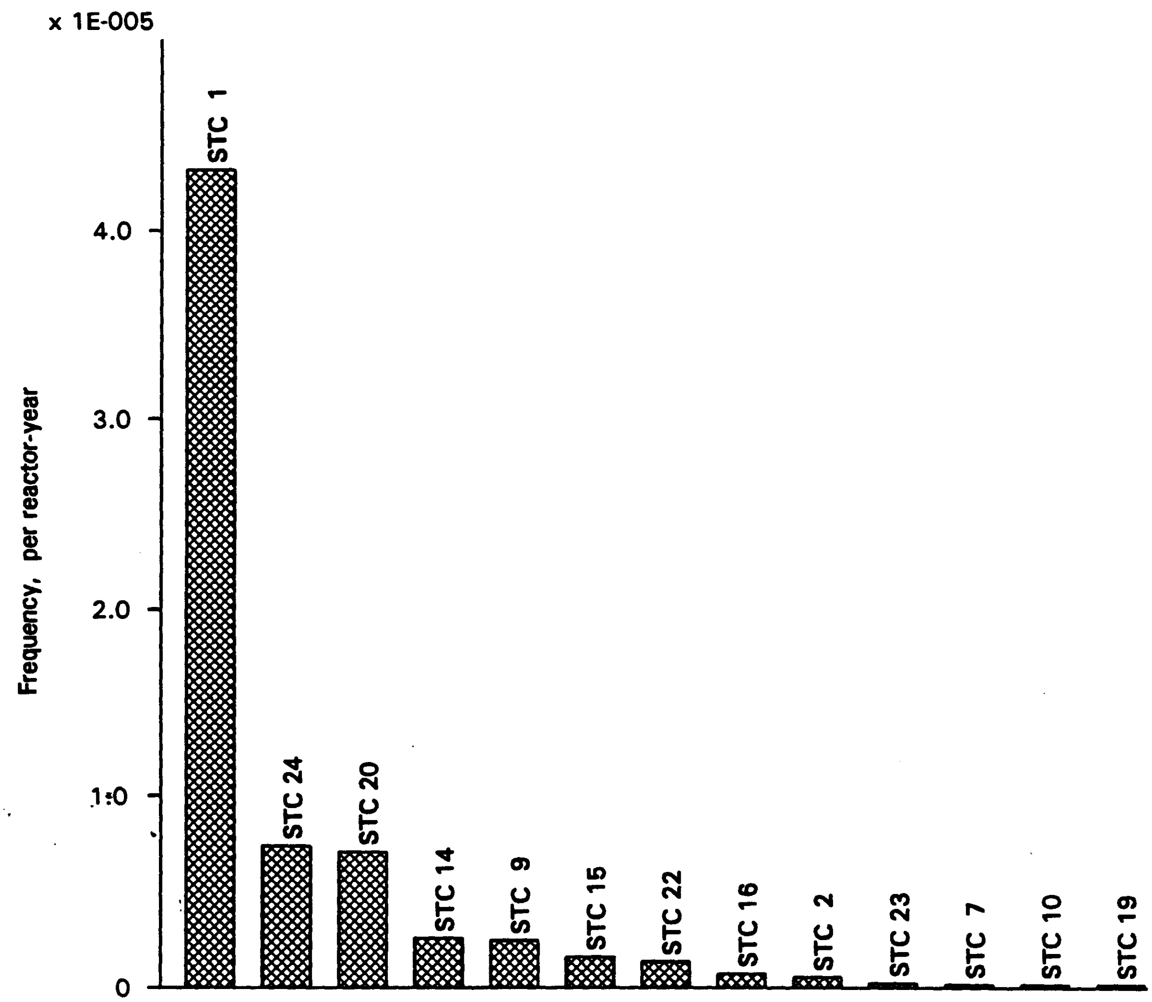


FIGURE 4.7.4-1 Source Term Category Frequencies

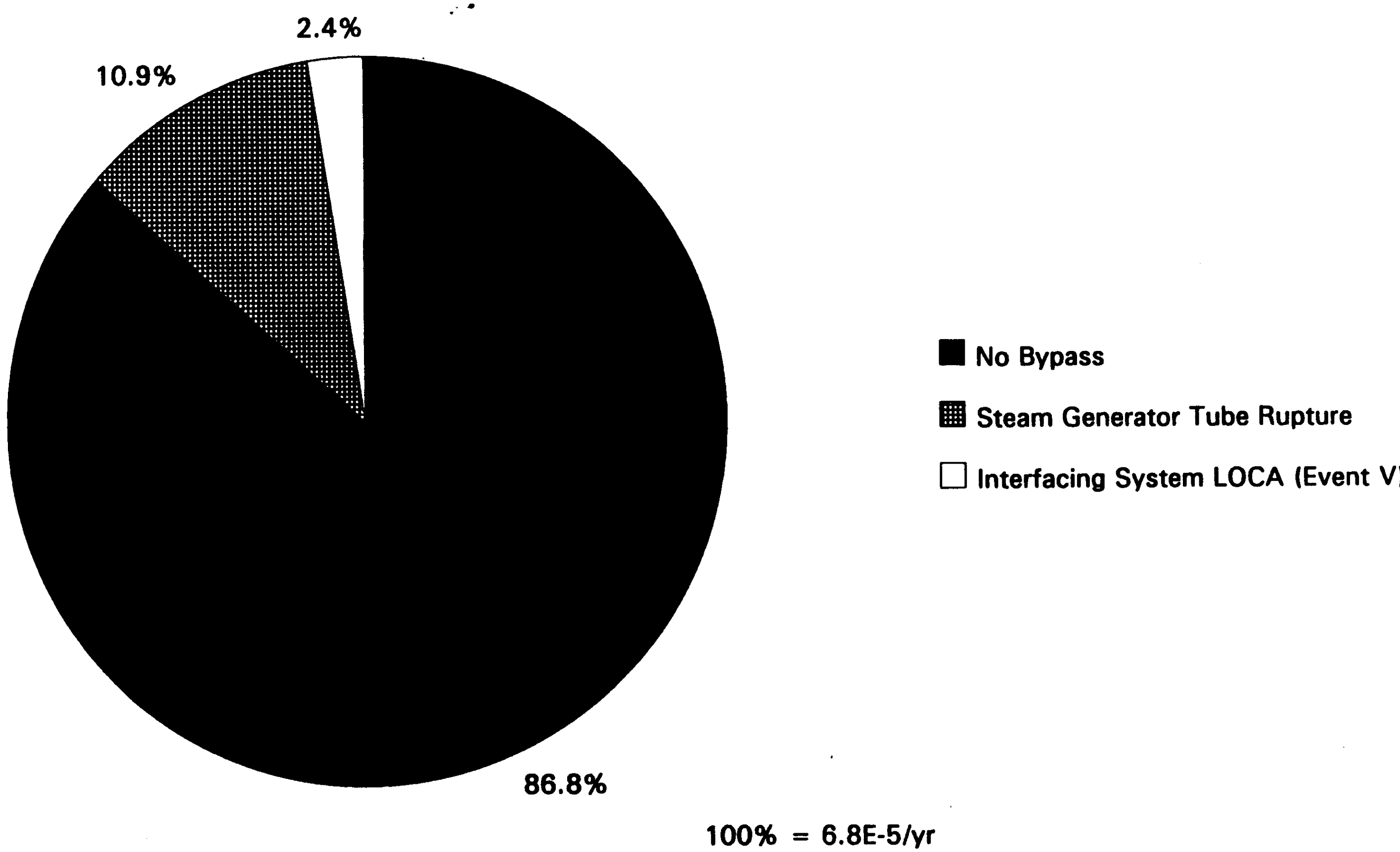


Figure 4.7.4-2
Containment Bypass Source Term Categories

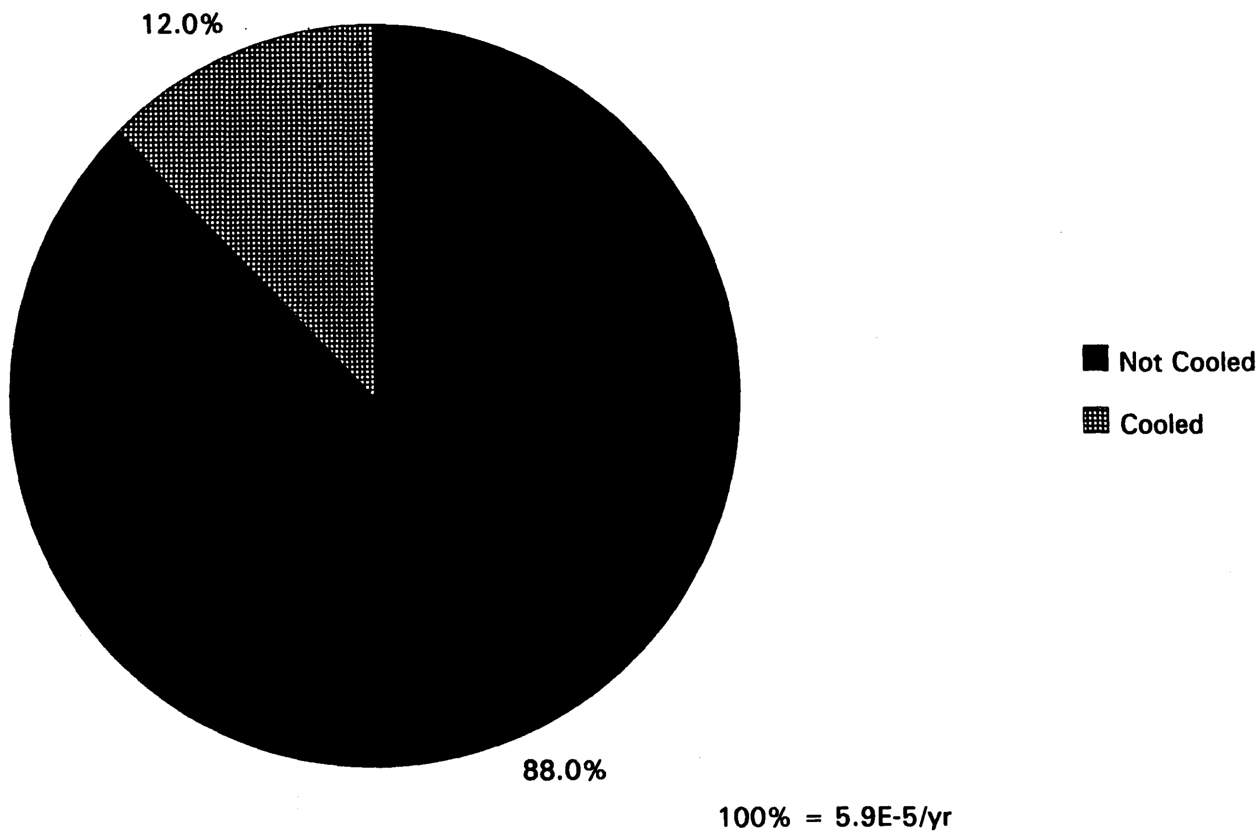


Figure 4.7.4-3
Fraction of Source Term Categories with No Reactor Vessel Failure

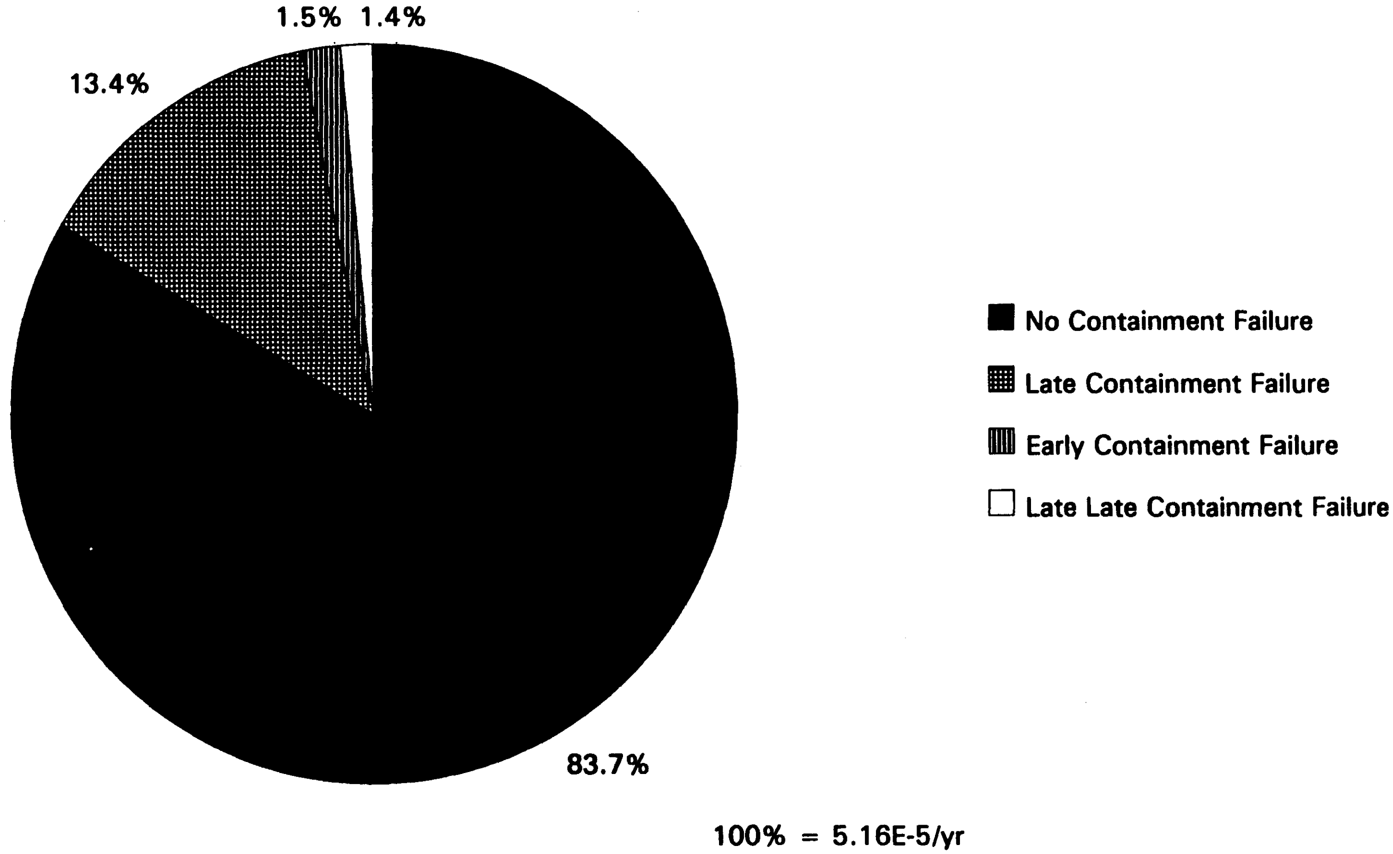


Figure 4.7.4-4
Time of Containment Failure

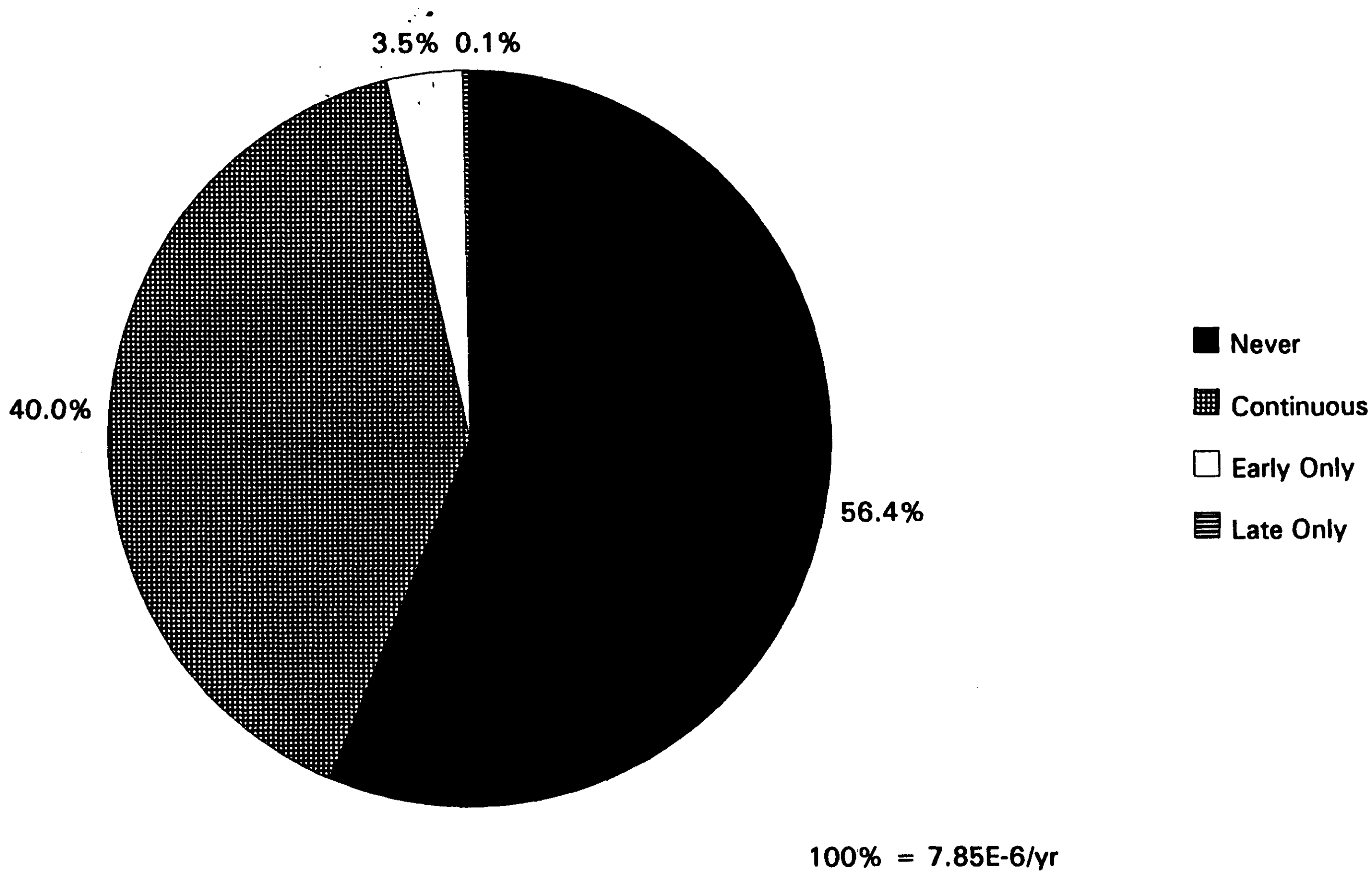


Figure 4.7.4-5
Containment Spray Operation

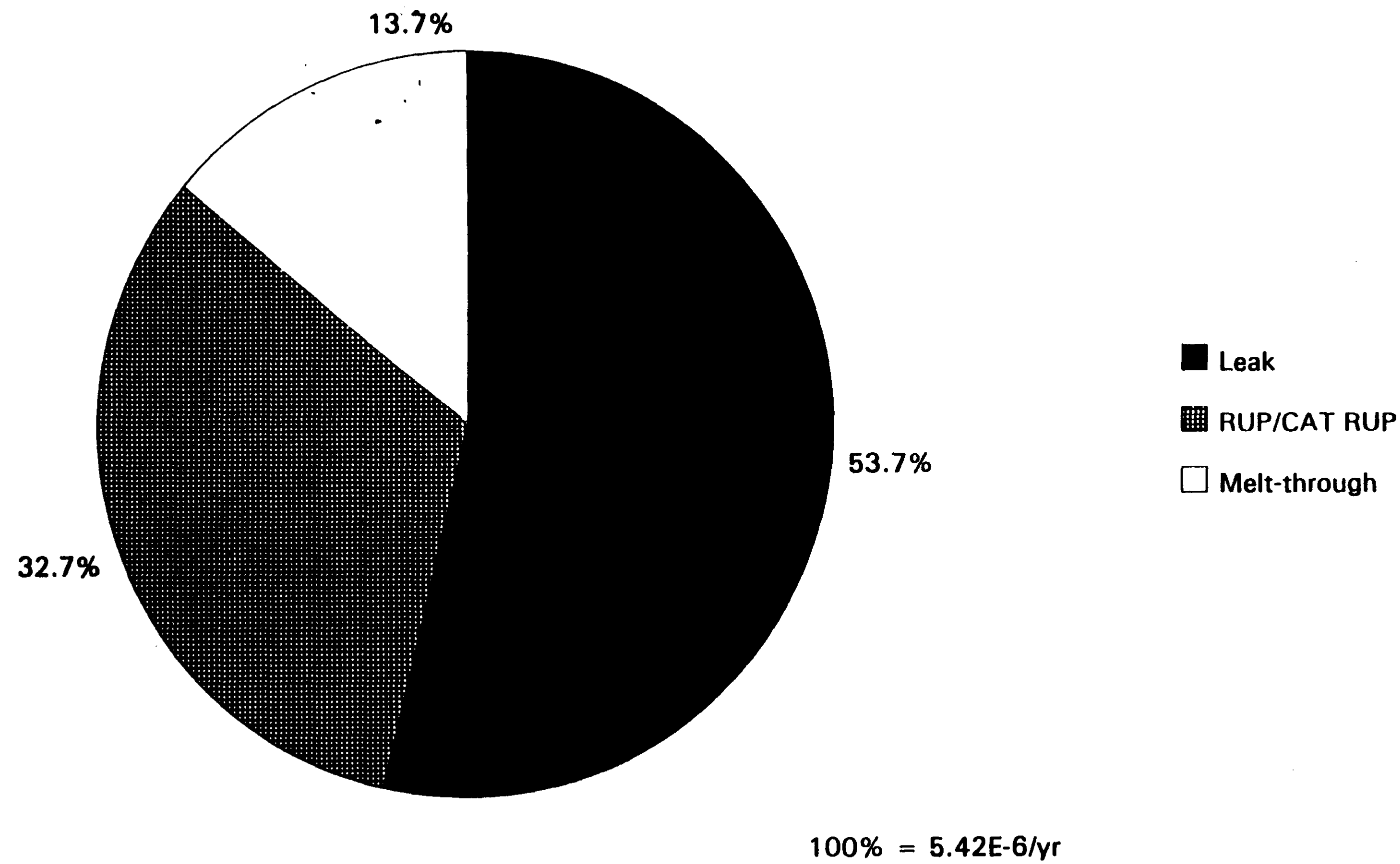


Figure 4.7.4-6
Mode of Containment Failure

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