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SEVERE ACCIDENT SEQUENCE ASSESSMENT FOR BOILING WATER REACTORS — PROGRAM OVERVIEW*

(Submitted for presentation at the Eighth NRC-RSR Water Reactor Safety Information Meeting, October 27–31, 1980, Gaithersburg, Maryland)

M. H. Fontana

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
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Introduction

The Severe Accident Sequence Assessment (SASA) Program was started at the Oak Ridge National Laboratory (ORNL) in June 1980. This report documents the initial planning, specification of objectives, potential uses of the results, plan of attack, and preliminary results.

ORNL was assigned the Brown's Ferry Unit 1 Plant with the station blackout being the initial sequence set to be addressed. This set includes (1) loss of offsite and onsite AC power with no coolant injection, and (2) loss of offsite and onsite AC power with high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) as long as DC power supply lasts. This report includes representative preliminary results for the former case.

Overall Goal and Key Objectives

The overall goal of the SASA Program is to significantly increase the actual and perceived safety of light water reactors (LWRs) by fulfilling the following key objectives:

(1) Identifying dominating reactor accidents that could involve severe core damage and/or pose a threat to fission product isolation from the environment;

(2) Determining, using best effort analysis, the behavior of reactors during the course of these accidents;

(3) Identifying potential corrective actions, assessing the effects of such actions, and recommending those actions that appear to be justified;

(4) Establishing feasibility and criteria for fundamental improvements in plant design and operation; and

(5) Making information available for implementation by interacting with NRC, utilities, and NSSS suppliers.

Although the level of nuclear reactor safety is deemed to be acceptable by a large fraction of the technical community, this conclusion is not unanimous. Also, the public has been drawn into the controversy through referendi and its perception of safety may prove to be the overriding factor affecting the future viability of nuclear power. Therefore, the overall goal includes the improvement of this perception as well as actual improvement in safety.

The absolute level of safety is very difficult to quantify; often it is presented as the sum of products of the probability of occurrence of each known event and its computed consequences. When the probabilities are very low and the consequences very high, the resulting products are highly questionable for use as quantitative indicators. However, the marginal or incremental improvement resulting from a design change, change in operation or an improvement in quantifying the consequence calculation can be more readily perceived and defended.

In the SASA Program, dominating reactor accidents will be identified. Complete probabilistic fault tree analyses of initiating events would not be done; rather, the many possible fault trees would be collapsed into a small number of representative events that could lead to significant core damage and risk to the environment.

The behavior of reactors under the prescribed accident conditions would be estimated using best estimate, as opposed to conservative, models. However, we would try to avoid computation in excruciating detail of each and every aspect of the reactor response. Emphasis would be on use of approximate models in an overall systems overview code; specific approximate models would be checked against in depth calculations using existing more complex codes, where appropriate. The models would be upgraded as increasing levels of comprehension were attained.

The above analysis would identify the progression of conditions, or states of the reactor and the time required to attain progressively worse states. Various modes of arresting the deterioration of the plant conditions would be identified along with the possible side effects of such ameliorative action. After sufficient review, including informal assessments by the manufacturer, the architect-engineer and the utility, actions that appeared to be justified would be recommended for formal comments and implementation.

After the aforementioned analysis of existing plants was well under way, fundamental improvements in plant design and operation could be conceptualized and assessed. Among these would be methods of assuring decay heat removal from containment and from the core in the absence of electrical power, inherently safe shutdown systems, key parameter display systems, on-line diagnostic systems, and on-line systems that predict the consequences of contemplated operator action. Feasibility of these ideas would be assessed; complete development lies outside the scope and resource of this program.

Finally, information would be made available to NRC, utilities, AE's and NSSS vendors by means of reports, briefing, research information letters and rulemaking hearings.

Constituency

The SASA Program results should be useful to the Nuclear Regulatory Commission Division of Reactor Safety Research (NRC-RSR), NRC Office of Regulation (REG), utilities and Nuclear Steam Supply Systems (NSSS) vendors.

NRC-RSR should find the output useful for developing the information base, developed outside of the regulatory process and its attendant day to day pressures, for understanding severe accidents; for guiding future R&D, keyed into long term needs identified by the overview; and for guiding advanced designs by assessing feasibility and establishing criteria for primary concepts that can be further developed under the auspices of other agencies if necessary.

NRC-REG should find the information useful for rulemaking hearings, such as degraded core analysis requirements; and for guiding NRC emergency response centers, utilities, and governments in evaluating options for managing severe accidents. The latter use was starkly illustrated during the TMI accident where the option of public evacuation was being considered without a firm basis of knowledge of what the reactor was likely to do as time progressed.

Utilities should find the program useful for aiding development of emergency operating procedures, which generally are now rudimentary for severe accidents. The results should also aid utilities in specifying more accident-resistant plants, where justified, and for training operators to recognize and respond to a wider spectrum of accidents and off-design conditions than is now the case.

Finally, vendors and architect-engineers (A-E's) should find the results useful for evolving plant improvements, guiding plant backfits and for guiding advanced designs.

Specific Objectives

The specific objectives through which the stated goals and specific objectives would be attained are:

(1) Identify and assess accident initiators with respect to their probability of occurrence and their potential for causing significant damage, and identify key sequences for further in-depth analysis.

(2) Given the important sequences, analyze the sequences with respect to (a) phenomena, including driving forces and fission product behavior; (b) timing of key events; (c) plant dynamic responses during these event sequences; (d) systems interactions throughout the accident sequences; (e) equipment performance, such as partial operation and rundown time of batteries, etc.; and (f) operator performance, including estimates of realistic expectations.

(3) Identify corrective action keyed to time windows established by the foregoing sequence analysis, identify requirements for implementation, and assess side effects. Such corrective action would include (a) equipment repairs; (b) operator action; (c) use of offsite special purpose equipment, such as backup electrical generators, flooding systems, chemical cleanup systems, etc.; and (d) public evacuation, based on at least a reasonable knowledge of the possible sequence of events to be encountered.

(4) Identify safe stable states and how they may be attained. By safe stable states, we mean conditions where the progressive deterioration of events can be arrested, preferably those conditions would be those that are relatively independent of various paths by which they may be reached. Perhaps, such stable states would be (a) core debris in situ, above the core support structure; (b) core debris on the lower head, with further penetration inhibited by internal and external flooding; (c) core debris in the bottom of the drywell, with provisions for permanent cooling; and so on.

(5) Identify inherent retention phenomena. Accident analyses in the past have usually assumed conservative behavior of fission product release and transport. Measurements at Three Mile Island (TMI), Unit 2, (reported in Volume II of the Reports of the Technical Assessment Task Force to the President's Commission on the Accident at Three Mile Island) indicate the partition coefficient of iodine as 10^{-6}; that is, the concentration of iodine in the gas phase in the reactor containment building potentially available for atmospheric dispersal was 10^{-6} of the concentration in the liquid phase. Subsequent chemical analyses showed the

existence of silver iodide in the sump water; the silver in the control rods (dispersed upon failure) and tied up iodine in an insoluble form. Searches should be made for such inherent retention phenomena because of the extremely large factors of reduction of potentially releasable radioactive materials that could be found.

(6) Establish feasibility of and criteria for improvements in (a) plant design, (b) instrumentation, (c) information displays and operation performance, and (d) emergency planning. Perhaps plants could be designed to be independent of electrical power to effect decay heat removal or they could be designed to have slow response, thereby allowing greater time for corrective action that may be needed eventually. Instrumentation should be able to track accident conditions as well as normal operating conditions and should be displayed in such a way that those responsible for managing the accident have information as accurate as possible about the condition of the plant at all times. TMI-2 was very instructive in this area. The in-core thermocouples (which were there for startup and fuel management purposes) were essential for estimating the condition of the core, but were displayed in such a way that they merely read off-scale on the readouts immediately available to the operator. Also, the radiation levels in the containment building went off scale, making it very difficult to determine the conditions within.

Emergency planning obviously needs to be improved. As mentioned previously, this program would provide estimates of the potential plant conditions as a function of time. Although we do not envision developing emergency evacuation plans, such information would provide a foundation upon which such contingency plans could be formulated.

(7) Identify research and development needs. Too much of the safety research and development in the past has been in response to almost random "what if" types of questions, usually arising from the heat of licensing procedures. Among the results has been a disproportionate effort on the large-break loss-of-coolant-accident (LOCA) and insufficient effort on practically everything else. Although this has been known to most workers in the field for many years, the TMI accident made this universally apparent. In this program we hope to first think through accident sequences of major importance, identify areas where more information is required, then identify R&D, keyed into the needs exhibited by the accident sequence analysis.

(8) Interact with NRC, utilities, architect engineers, and vendors. Obviously, the program would be of little value if it did not influence subsequent events. As mentioned previously, a constituency for this program exists, although it may not be universally recognized. Most interactions would be with NRC-RSR in the form of topical reports, progress reports and program reviews. Contact with NRC-REG would be by the aforementioned reports, research information letters, and meetings which would be set up, presumably, by NRC-RSR. Contact with TVA has been assured by an agreement at key high levels, by identification of persons responsible for coordination, exchange of information, and participation in regular meetings. The TVA interaction arises because of the assignment of Brown's Ferry Unit-1 as the first reference plant to be assessed by the ORNL Program. Meetings with the General Electric Company are now being set up. We emphasize that these interactions exist in the interest of obtaining accurate information and will not affect the independence of the program in any way. Interactions with other utilities, A-E and vendors have not been defined, other than by distribution of reports and by participation in meetings such as the NRC-RSR Light Water Reactor Safety Research Information Meetings.

The Accident Sequence Time-Line Chart

As an initial attempt at severe accident sequence assessment, ORNL has started to analyze the TVA Brown's Ferry Unit 1 Boiling Water Reactor. The first sequence attempted is the loss of offsite and onsite AC power.

ORNL has devised a Sequence Progression Time-Line Chart to assist in tracking the accident progression in time, identifying important events, identifying important phenomena, maintaining awareness of concurrent events, and identifying potential corrective action.

The basic elements of these charts are shown in Figures 1 and 2. Figure 1 shows a way of displaying events that progress in time, those factors that affect the event and vice versa. Time increases from left to right. The linkage to a prior event that triggers equipment action is shown labeled as (1). The indicator of whether or not the equipment operates as intended is shown by a diamond (2); the exit line from the diamond is shown; generally if the equipment operates as desired, the exit is upward, if not, downward. The factors that affect inoperability of the piece of equipment in question are shown as a fault tree (3); since these events progress throughout the accident the fault tree should show interactive effects of phenomena encountered during the accident in addition to normal fault tree analysis.

The effect of the maloperation results in a parameter change or damage of some sort that is shown as a damage function which increases with time (4); generally increases in the undesirable direction is plotted downward. If an instrument exists for sensing the parameters being plotted, the point at which it delivers an alarm or other important indication is shown by the half-shaded ball (5). The point at which irreversible damage occurs is shown as triangle (6). The indicator for corrective action is shown as a ball (7). The events that must occur to effect such corrective action are shown in the string (8) showing recognition time, operator action time, equipment response time and system response time. Also the actions that must occur to implement the corrective actions are shown as an event fault tree (9). The exits to continuation of the time-line chart would take route (10) if the damage is arrested before irreversible damage occurs and route (11) if not.

Phenomenological effects that are time independent or so fast as to be essentially instantaneous in the time scale of the chart are shown as in Figure 2. The example of a steam explosion in the primary vessel is used for illustration. The entry from the previous events is shown by the triangle connector (12). The severity of the event (13) is plotted to the right. For example, the pressure pulse due to a steam explosion when hot core debris falls into water can have a range of magnitude (depending on too many factors to enumerate here), therefore a method of showing the possible range of magnitudes is necessary. The sideways triangle (14) indicates the magnitude of the pulse at which the vessel (or associated appurtenances) would fail. If the pressure pulse is more than this failure threshold the exit line to the event sequence progressing from vessel failure is shown as in (15). If the pressure pulse is less than that required to fail the vessel, the exit line to the next phenomena that must be addressed is shown as (16). In this case, the subsequent phenomena that must be dealt with is the liquid slug that could be accelerated by the steam explosion and cause failure by impacting the top head. The magnitude of the force impacting the top head is shown as (17); this depends on many things such as void fraction, steam-liquid flow coherency; and energy absorbing characteristics of the upper internal structures. Again the magnitude at which vessel failure would occur is shown by the sideways triangle (18). The exit lines to subsequent progression paths with and without vessel failure are shown as previously discussed.

The chart developed for the station blackout without coolant injection for Brown's Ferry is shown in Figure 3. Numerical data was obtained from reference (1). This is our first attempt at using this method. It does not include all the possible paths, nor does it include items (3), (8), or (9) from Figure 1. For pertinent features of Brown's Ferry-type BWRs, the reader is referred to the figures in Appendix B.

Given an understanding of the basic elements illustrated in Figures 1 and 2, the chart is self-explanatory. The event starts at time t = 0 with loss of all AC power which causes full load rejection and fast closure of the turbine control valves. The time required to close the turbine stop valves is shown in the first "phenomena" plot. Concurrently, at t = 0 the recirculation pumps and condensor circulating water pumps trip off. The reactor pressure is plotted, with increasing (undesirable) pressure being plotted downward from the baseline. The opening and closing of the safety relief valves (SR/V) are shown. The first dotted line indicates the signal fed to the diamond which indicates subsequent operation by a downward vertical line. Notice that SR/V relief is to the suppression pool, the temperature of which is plotted starting at the time of initial SR/V opening.

Notice that, in addition to reactor pressure, the following time dependent phenomena are plotted; reactor power, main steam isolation valve (MSIV) opening, core water level, feedwater turbine flow, suppression pool temperature, suppression pool pressure, core heatup, hydrogen generation, a relatively undefined core damage function which flags various extents of zirconium oxidation and core melting, drywell pressure, core penetration through control rod structures, core heatup of bottom head, total vessel heatup, penetration of drywell concrete by corium (core material), hydrogen generation from reaction of corium and water, CO_2 generation from decomposition of limestone aggregate in the concrete, and steam generation of water in contact with corium.

Instantaneous events shown include the extent of core collapse into the bottom head, upward reaction forces as a result of bottom head failure (at pressure), steam explosion energy release, initial amount of concrete erosion from the jet blast that may arise from failure of the bottom head at pressure, release of gases and fission products to the external building through the wetwell, attenuation of fission products in wetwell water, dispersal of wetwell water, and direct release of fission product to the external building from drywell failure.

Equipment operability factors are self explanatory. However, phenomena questions may require explanation. The note at about t = 77 minutes asks the question concerning whether the core, upon melting, dribbles into the water in the lower head, or accumulates on the lower grid plate with subsequent collapse into the bottom head. For the calculation illustrated by Figure 3, the former was assured; subsequent heat up rates, boiling time, etc., depend on this assumption. This does not mean that we think that this case is any more likely than the other. At about t = 95 minutes, the phenomenological question is asked: "Does H_2 burn?" Since the Brown's Ferry drywell/wetwell system is inerted, the "no" branch is followed. At about t = 22 hours the question is asked as to which fails first; the wetwell or the drywell. The sequence stops shortly thereafter with transfer to appropriate fission product transport charts, which are now being developed.

It is apparent that the illustration in Figure 3 follows physical events and driving forces, such as pressure, temperature, and fluid flow. Concurrent with this chart should flow an adjacent chart, showing the disposition of fission products as time progresses. Also, it is apparent that only one chain is followed in Figure 3. Generally, at branch points, the path that appeared most likely, in our judgment, was followed. In principle, all potentially important paths should be followed. Obviously, to do this in depth would require an enormous amount of work.

The question probably in the readers mind is why not follow the standard event tree procedure as done in Reactor Safety Study[4] and illustrated in Figure 4. Although very useful to order event sequences, this type of tree has several problems. Foremost, timing is not shown. Also, once a piece of equipment is shown to either operate or not, it is difficult to show the subsequent repair and restart. Finally one has to be clever in arranging the events so that the proper order is maintained consistently. Since a major objective of our program is to show how much time elapses between major events and to indicate time periods available for potential corrective action, these factors would be serious shortcomings for our purposes. We think it is simpler to show the events as they would occur, as on a strip chart recorder.

Standard fault trees which relate thousands of potential initiating events will not be used. We will begin with a small chain of events that clearly would lead to core damage should they progress unperturbed. However, as mentioned previously, small fault trees would eventually be used to show how needed equipment may not be available, and "invert" fault trees would be used to show what must occur for corrective action to take place.

Potential Uses for Brown's Ferry Study

It is intended that the Brown's Ferry SASA Study can be used to (1) evaluate consequences for a blackout; (2) improve operating procedures, operator training, and licensing; (3) determine instrumentation and control requirements; (4) determine DC and AC reliability requirements; (5) develop improved plans for mitigative actions, that is, onsite emergency plans; and (6) develop improved offsite emergency plans.

By taking an overview of the whole problem prior to doing in-depth analysis of specific areas, we feel that we are more likely to identify areas of importance and place proper priority on items for further study. Also the time-line chart would serve as a reference of what the plant condition is as a function of time. It would be very useful to prepare a parallel chart that shows instrumentation readings available to the operator and then compare his comprehension of events with the actual case represented on the reference time-line chart. This exercise would be very helpful for operator training and for identifying instrumentation and control requirements. Perhaps the time-line chart can be computerized to play in real time as a simulator for training operators for emergency operations. Also, the chart could show trigger points for implementation of emergency action ranging from onsite repairs to public evacuation.

As mentioned previously, specific work areas are easily identified by constructing a time-line chart. A list of work areas covering phenomenology only was developed from the time chart and is shown in Appendix A. Subsequent lists identifying diesel reliability requirements, DC power reliability requirements, specific drastic corrective action such as vessel external flooding, effect of operator maloperation, and side effects of potential corrective action could be developed.

Conclusions

A set of overview studies of potentially severe reactor accidents is required to place these events in perspective, to perform best estimates rather than ultra-conservative bounding calculations, to identify areas requiring further work, to identify potential corrective action and design improvements, and to prioritize future work. These studies should not be tied to specific licensing issues at this time, although their influence would be eventually felt in the licensing process. It is estimated that scoping studies, over a time period of about one year, would establish the feasibility of the approach and would indicate if further work is warranted.

REFERENCES

1. D. Yue, *Severe Accident Sequence Analysis Brown's Ferry: Complete Station Blackout (No AC Power; No Injection) — Mark I Containment Response*, ORNL/TM (to be issued).

2. Reactor Safety Study, U. S. Nuclear Regulatory Commission, WASH 1400 (NUREG 75/014) October 1975.

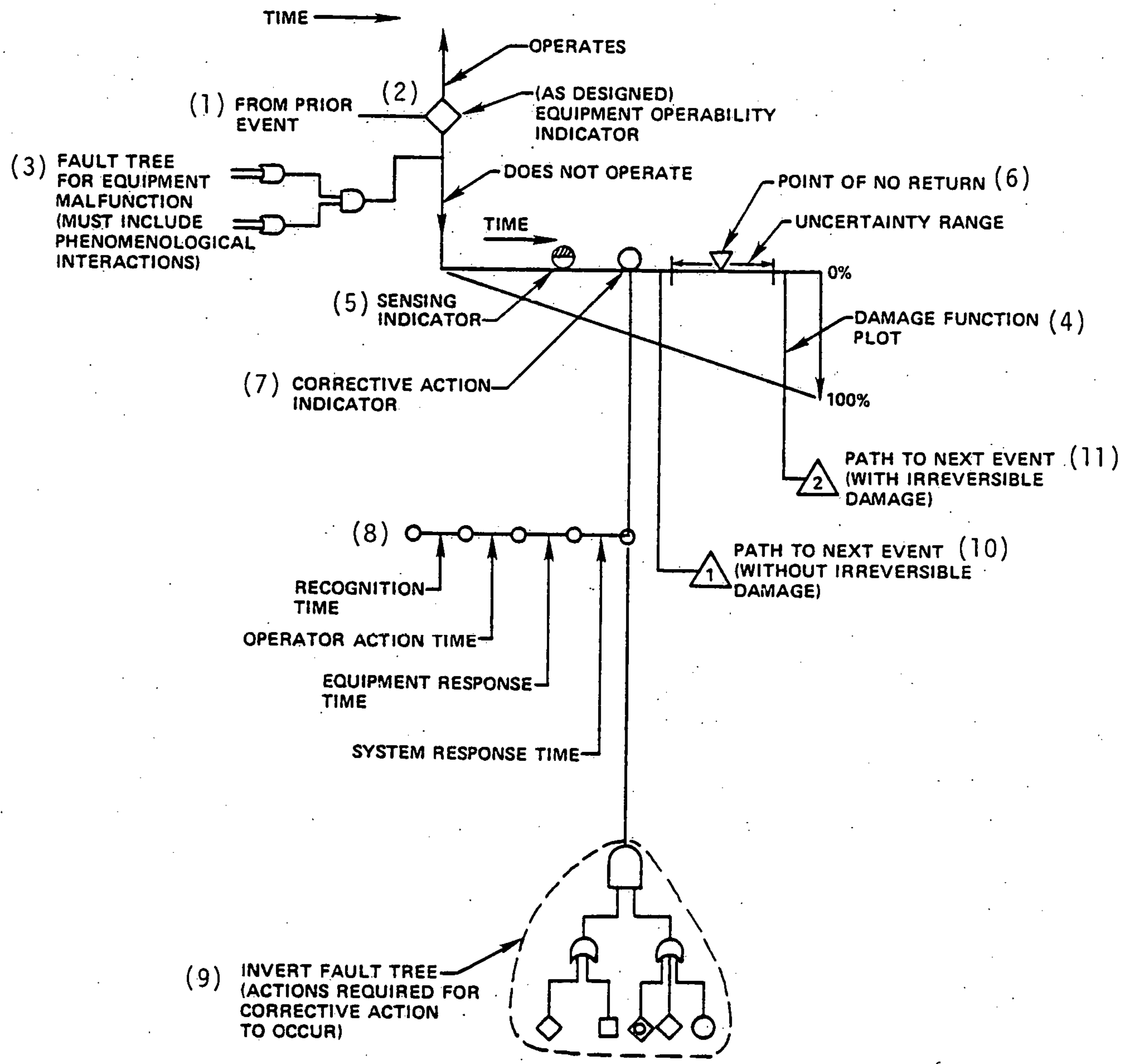


Figure 1. Basic Sequence Progression Elements

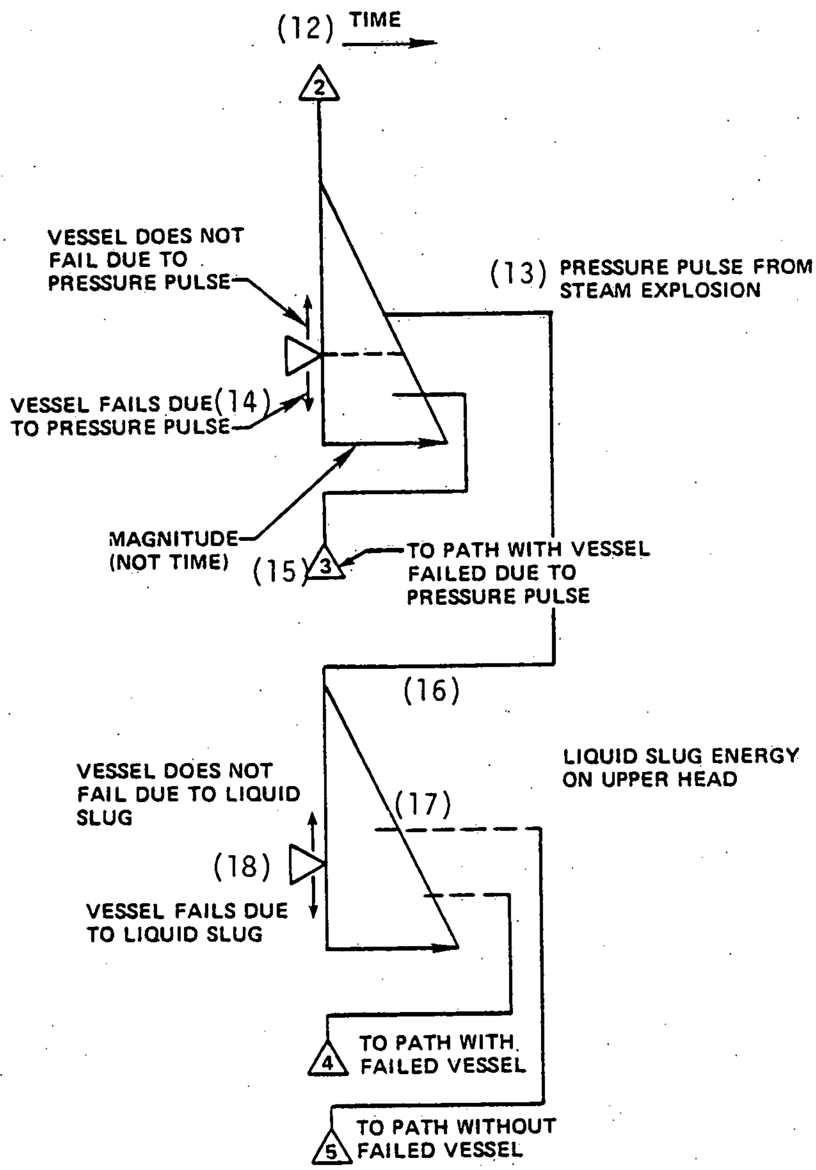


Figure 2. Time-Independent or Instantaneous Phenomenological Effects (Example - Steam Explosion)

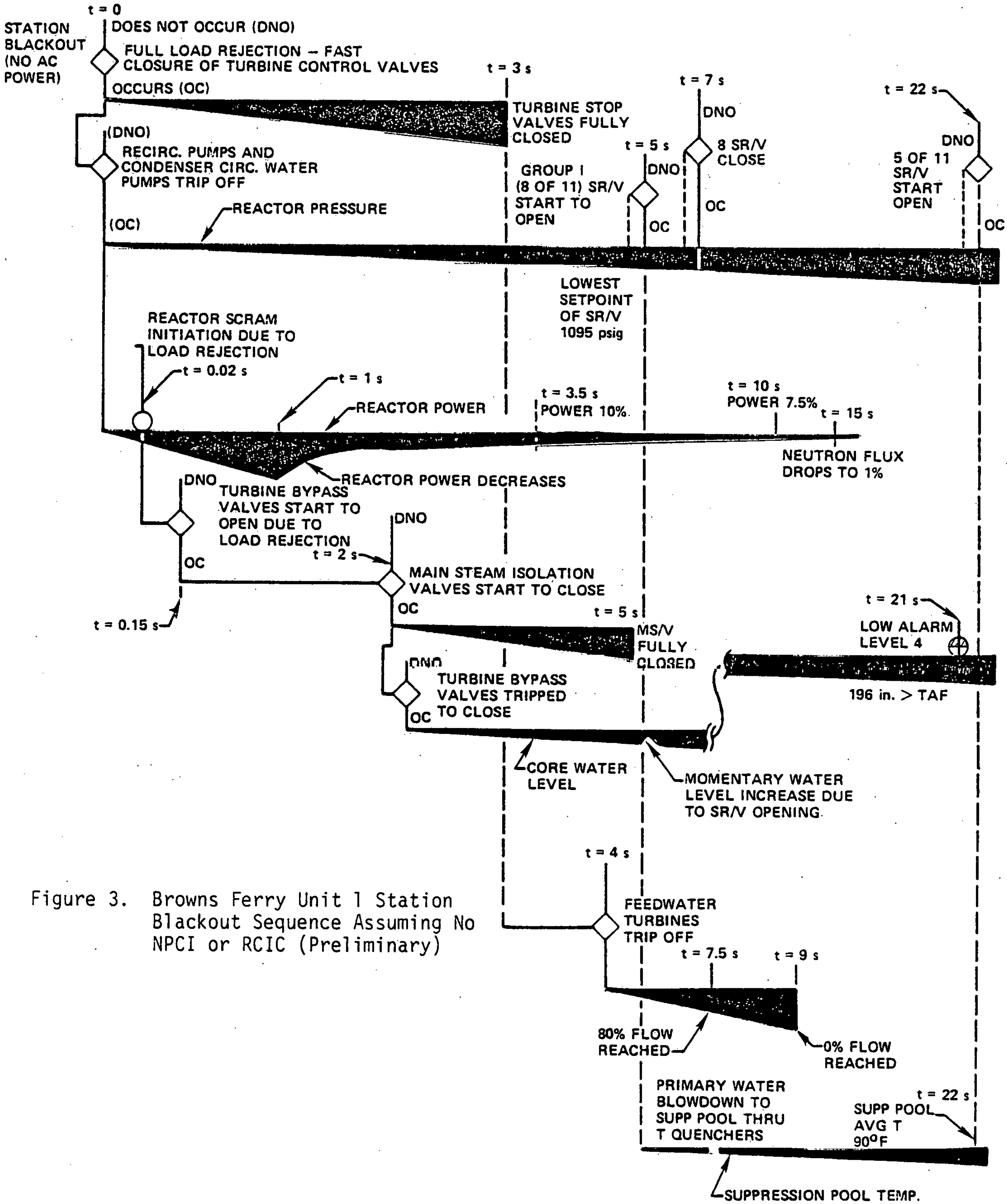


Figure 3. Browns Ferry Unit 1 Station Blackout Sequence Assuming No NPCI or RCIC (Preliminary)

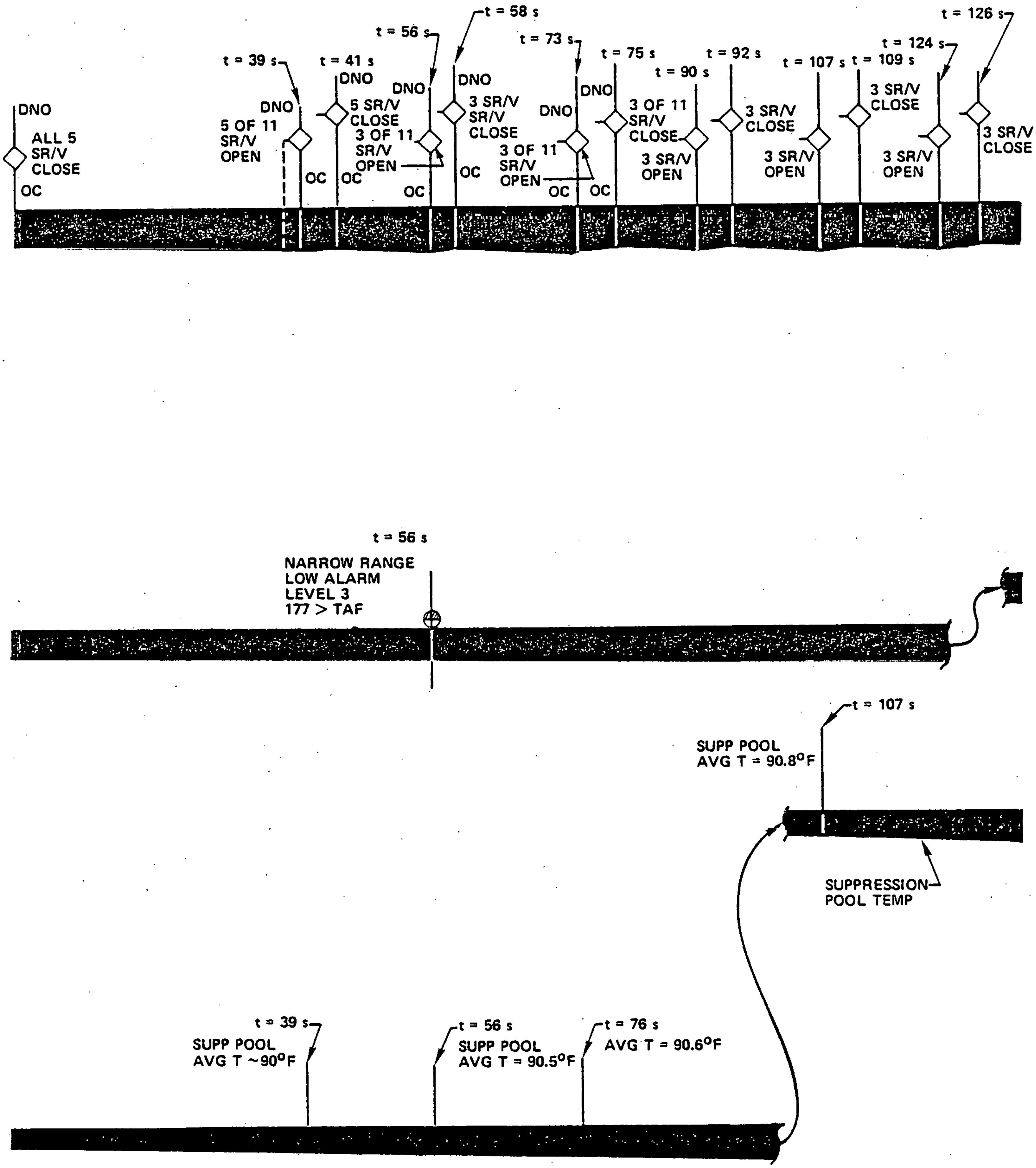
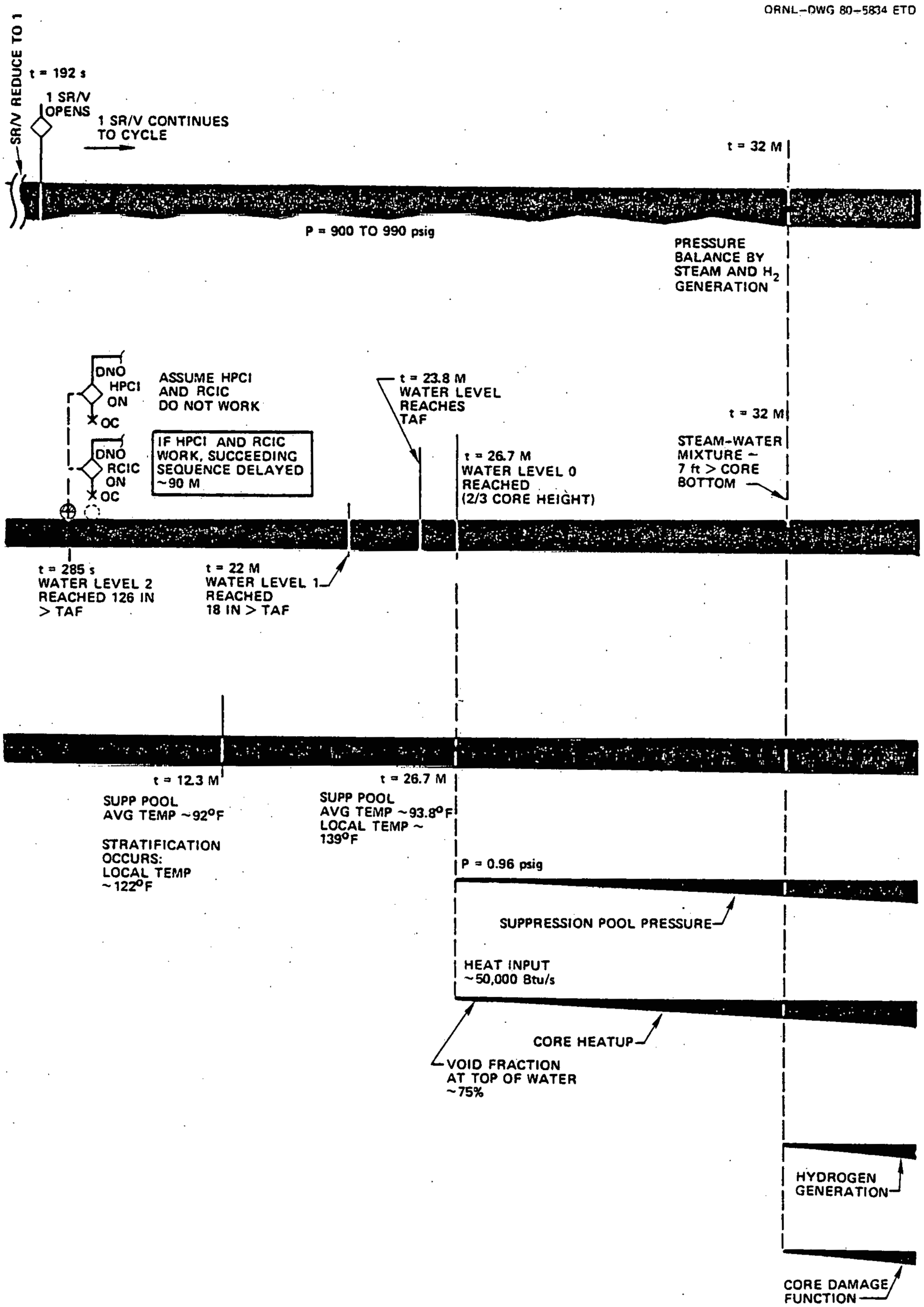


Figure 3. Continued


ORNL-DWG 80-5834 ETD
SR/V REDUCE TO 1
t = 192 s
1 SR/V OPENS
1 SR/V CONTINUES TO CYCLE
t = 32 M
P = 900 TO 990 psig
PRESSURE BALANCE BY STEAM AND H_2 GENERATION
DNO
HPCI ON
OC
ASSUME HPCI AND RCIC DO NOT WORK
DNO
RCIC ON
OC
IF HPCI AND RCIC WORK, SUCCEEDING SEQUENCE DELAYED ~90 M
t = 23.8 M WATER LEVEL REACHES TAF
t = 26.7 M WATER LEVEL 0 REACHED (2/3 CORE HEIGHT)
t = 32 M
STEAM-WATER MIXTURE ~ 7 ft > CORE BOTTOM
t = 285 s WATER LEVEL 2 REACHED 126 IN > TAF
t = 22 M WATER LEVEL 1 REACHED 18 IN > TAF
t = 12.3 M
SUPP POOL AVG TEMP ~92°F
STRATIFICATION OCCURS: LOCAL TEMP ~122°F
t = 26.7 M
SUPP POOL AVG TEMP ~93.8°F LOCAL TEMP ~ 139°F
P = 0.96 psig
SUPPRESSION POOL PRESSURE
HEAT INPUT ~50,000 Btu/s
CORE HEATUP
VOID FRACTION AT TOP OF WATER ~75%
HYDROGEN GENERATION
CORE DAMAGE FUNCTION

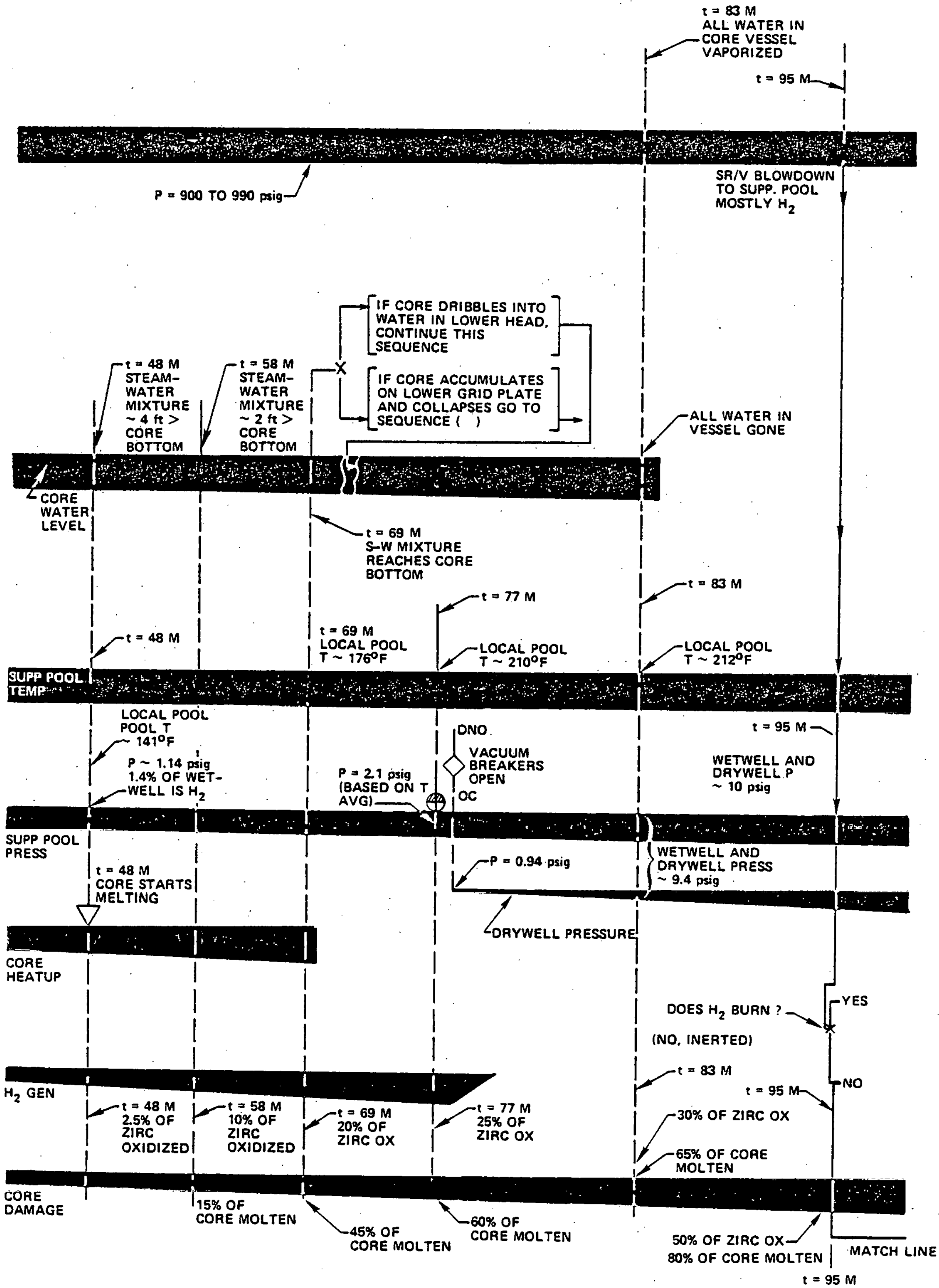


Figure 3. Continued

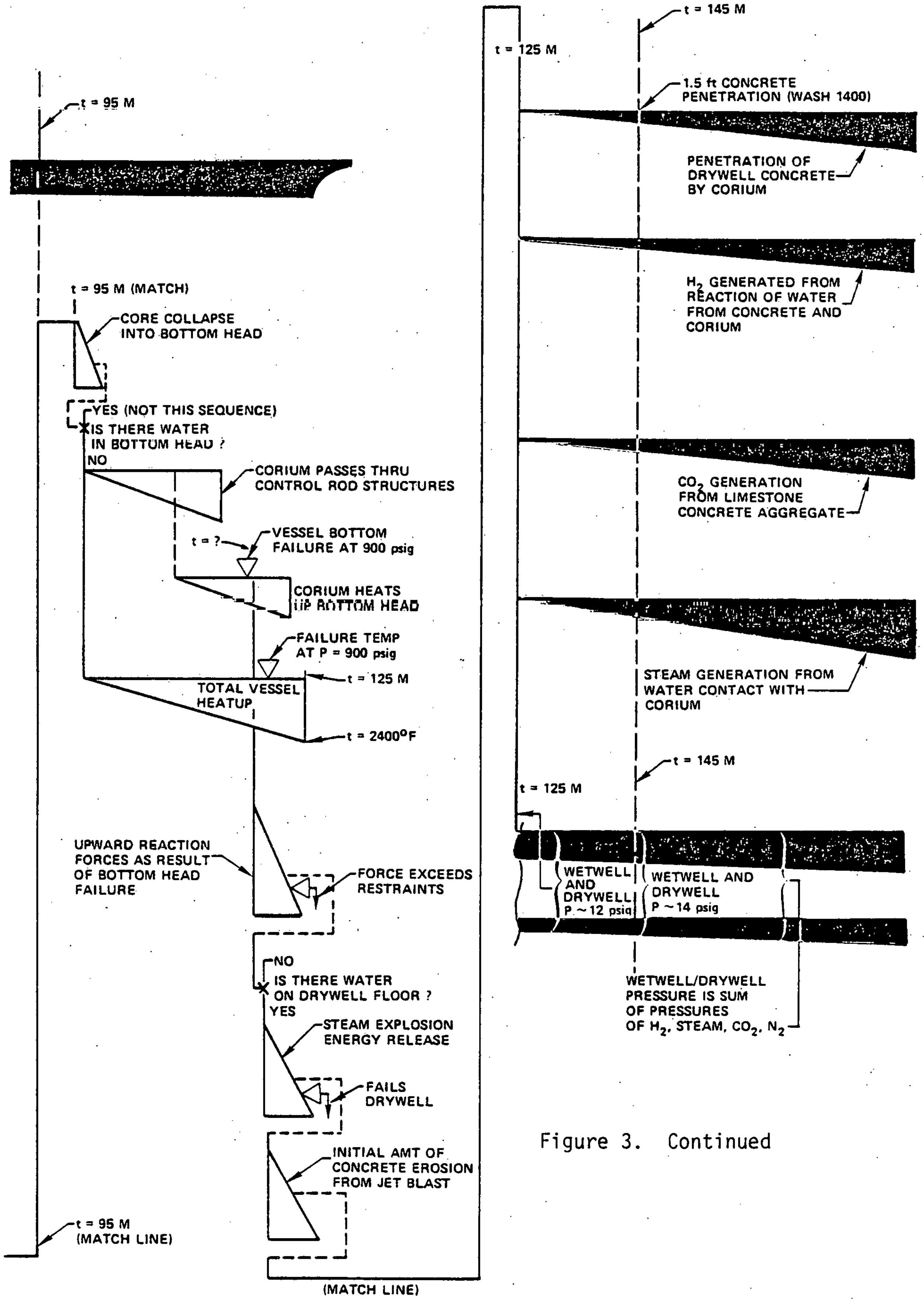


Figure 3. Continued

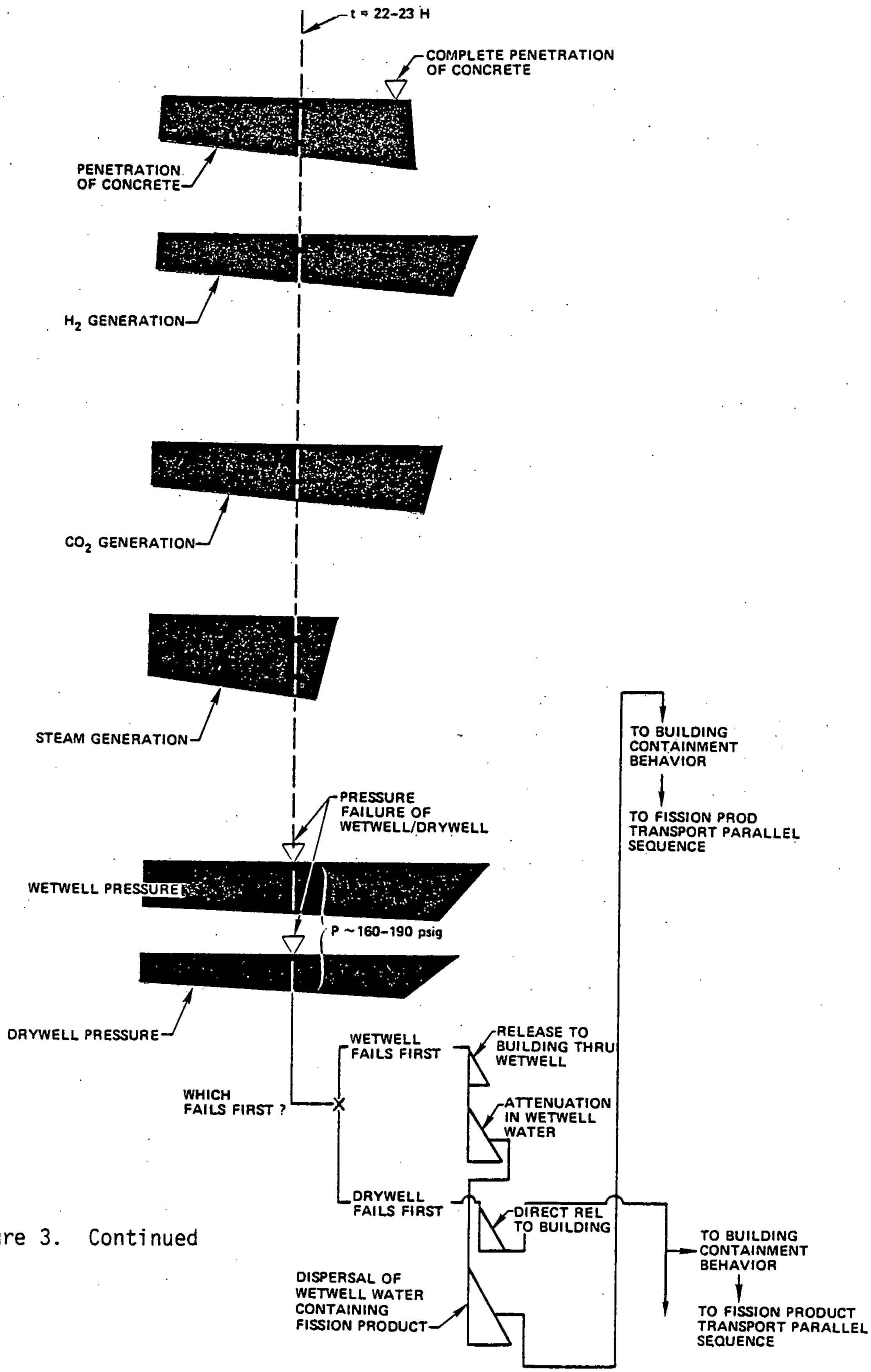
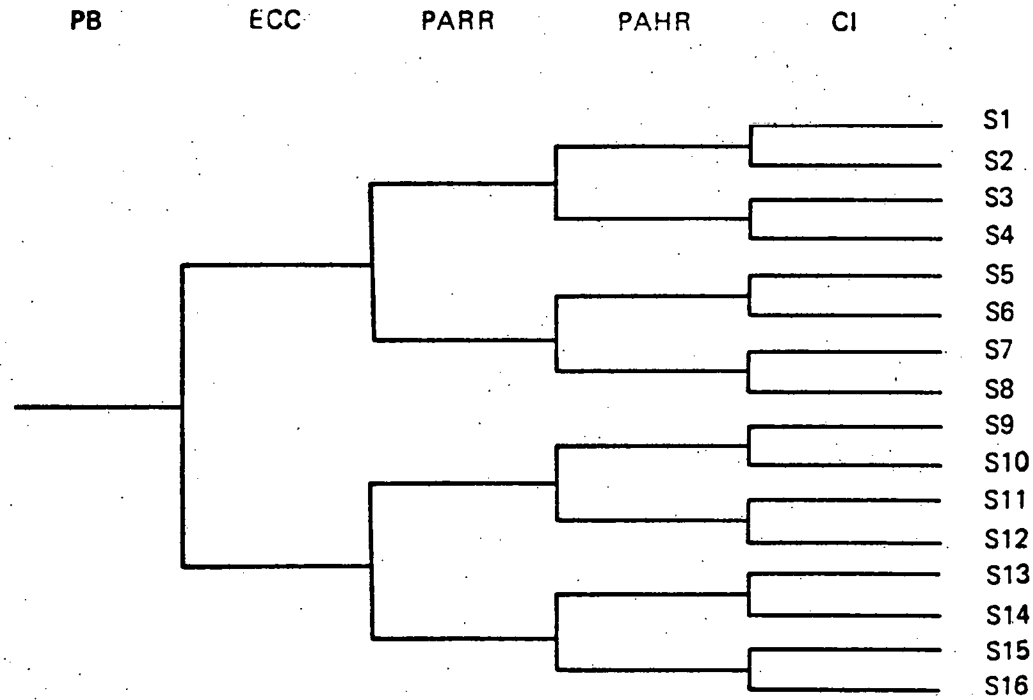


Figure 3. Continued



PB = Pressure Boundary Break
ECC = Emergency Core Cooling
PARR = Post Accident Heat Removal
CI = Containment Isolation

Figure 4. Illustrative Event Tree for LOCA Functions

Appendix A

BWR SPECIFIC WORK AREAS IDENTIFIED BY ACCIDENT SEQUENCE TIME-LINE ASSESSMENT (PHENOMENOLOGY)

Mass Loss, Depressurization from Relief Valve Opening as Function of

- Number of Valves
- Pressure
- Water Level
- Core Heat Generation Rate
- Stuck Valves

Water Level Swell as Relief Valves Open

Void Distribution in Core

- During Swell
- During Resettle
- Effect of Fuel Cooling
- Effect on Zirc Oxidation

Steam Flow Above Water Level

- Effect on Fuel Cooling
- Effect on Zirc Oxidation

Suppression Pool Temperature Stratification

- Effect on Torus Pressure
- Effect on Fission Product Trapping

BWR Specific Work Areas-2

Core Relocation Behavior (Dribble-Through vs Buildup on Crust)

Effect on Steaming Rate and Amount of Water Left in Bottom Head at Time of Core Support Structure Collapse

Core Damage and Relocation Incoherency

Effect on Core Support Structure Failure Time and Mode

Effect of Control Rod Guide Structure on Mode of Core Collapse into Bottom Head

Mode of Vessel Failure

Effect of Internal Pressure

Side Rip vs Coherent Failure of Bottom Head

Reaction Forces from Failure of Pressurized Vessel

Direction

Magnitude

Available Restraints

Steam Explosion in Vessel

Effect of Pressure

Amount of Water Left in Head

BWR Specific Work Areas-3

Damage Potential of Steam Explosion

Direct Shock Wave

Effect of Boiling Incoherency in Liquid Slug

Effect of Two-Phase Flow in Liquid Slug

Effect of Upper Internal Structures on Cushioning of Liquid Slug

Steam Explosion in Drywell

Effect of Limited Water Availability

Failure Mode of Drywell under Steam Explosion Forces

Shock Wave

Liquid Slugs

Erosion of Concrete under Conditions of Pressurized Melt-Through

Concrete Penetration by Quiescent Core Debris

Hydrogen Generation by Core Debris — Water Reactions

BWR Specific Work Areas-4

Water Generation Rate from Concrete Decomposition

CO_2 Generation Rate from Limestone Aggregate Decomposition

Steam Explosions, Steam Generation, and Debris Dispersal from Addition of Water to Hot Core Debris

Wetwell and Drywell Failure Modes

- Shock Loads
- Pressurization
- Melt-Through

Failure Modes of Outer Building

Fission Product Transport Paths

Fission Product Behavior

- Release from Fuel
- Chemical/Physical States
- Solution in Water
- Plateout/Fallout
- Effect of Spray and Filter Systems

Appendix B

DESCRIPTIVE DRAWINGS OF TYPICAL BROWN'S FERRY - TYPE BOILING WATER REACTORS

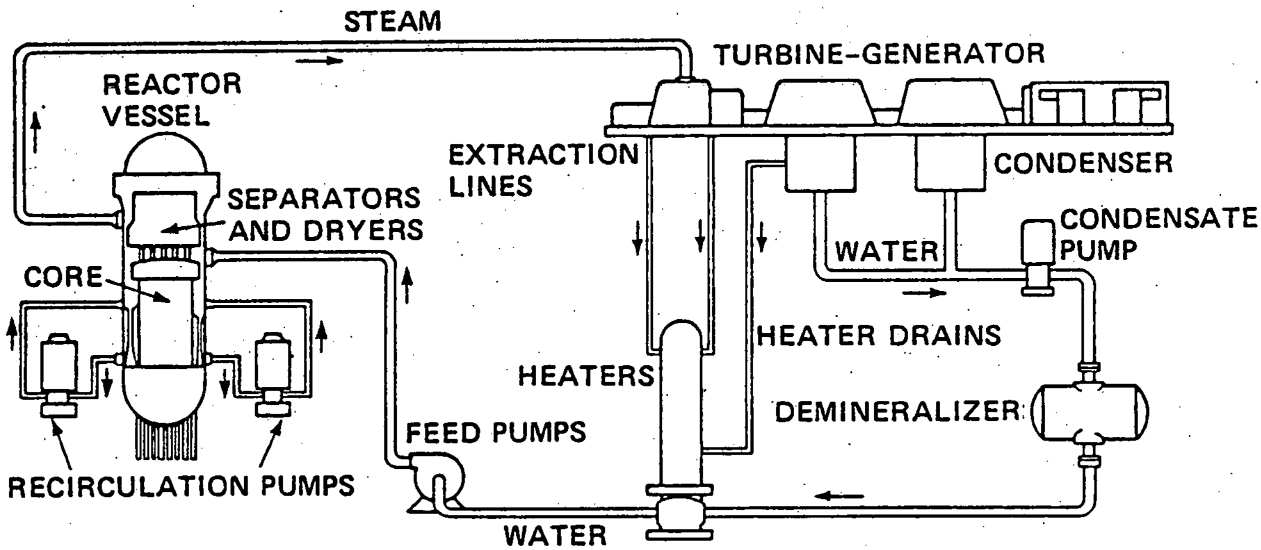


Figure B1. BWR Flow Schematic

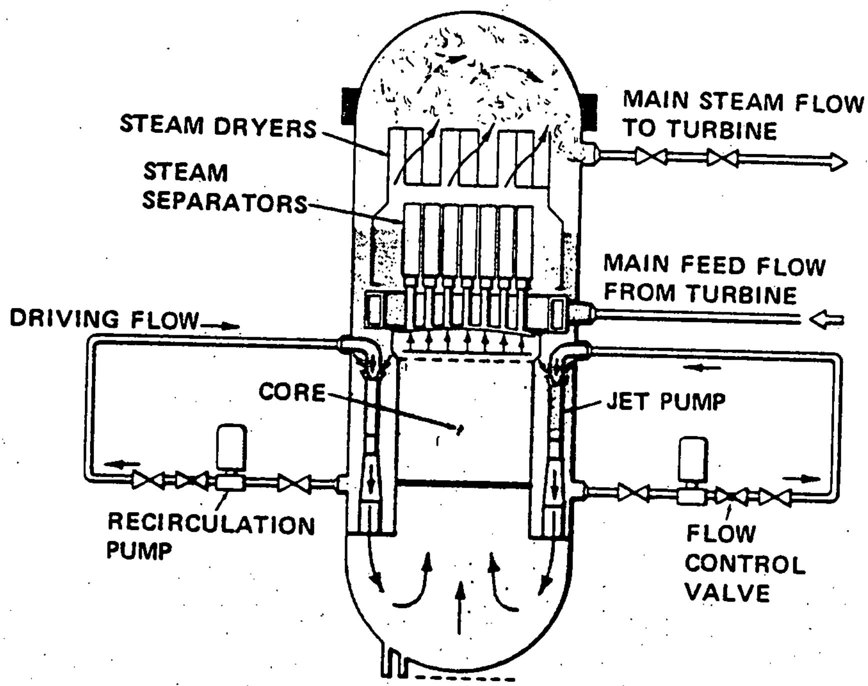


Figure B2. BWR Core Cooling System

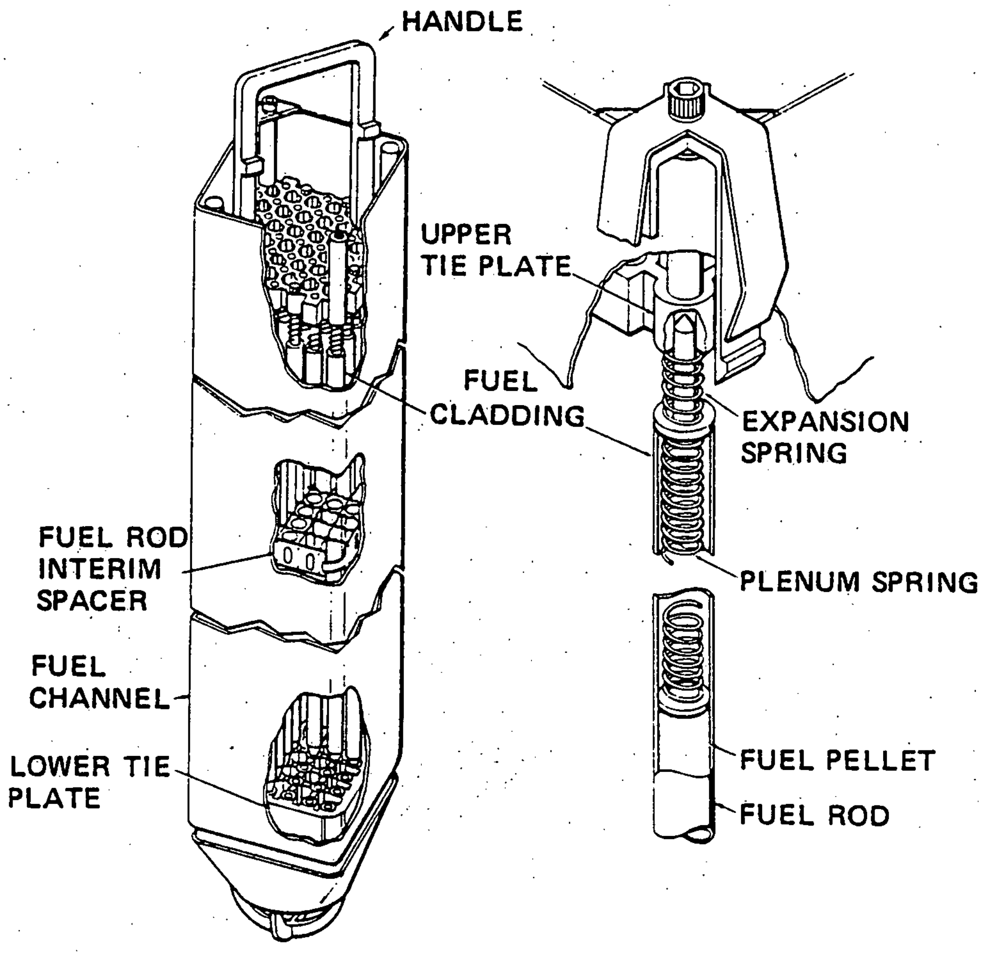


Figure B3. BWR Fuel Assembly

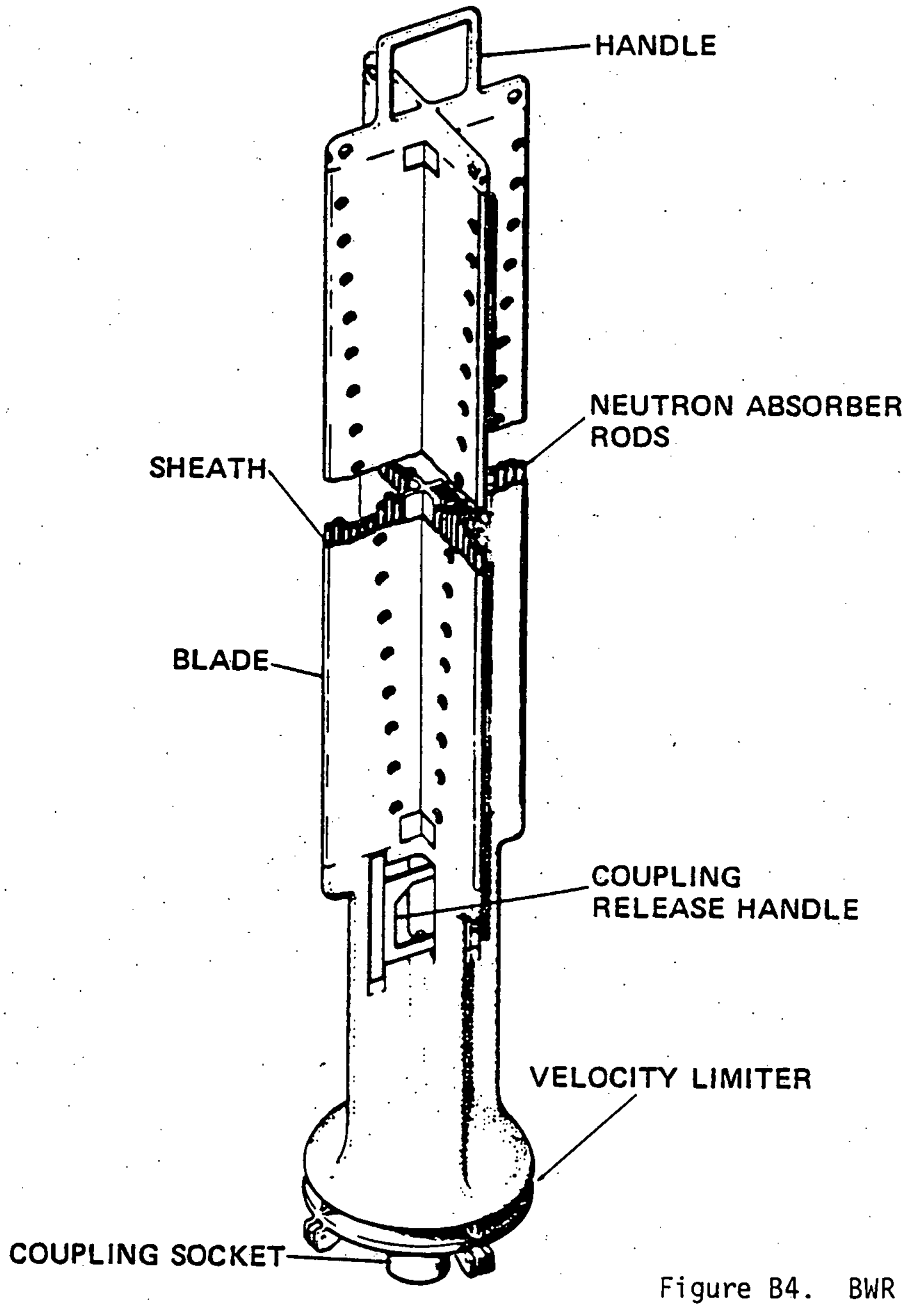


Figure B4. BWR Control Rod

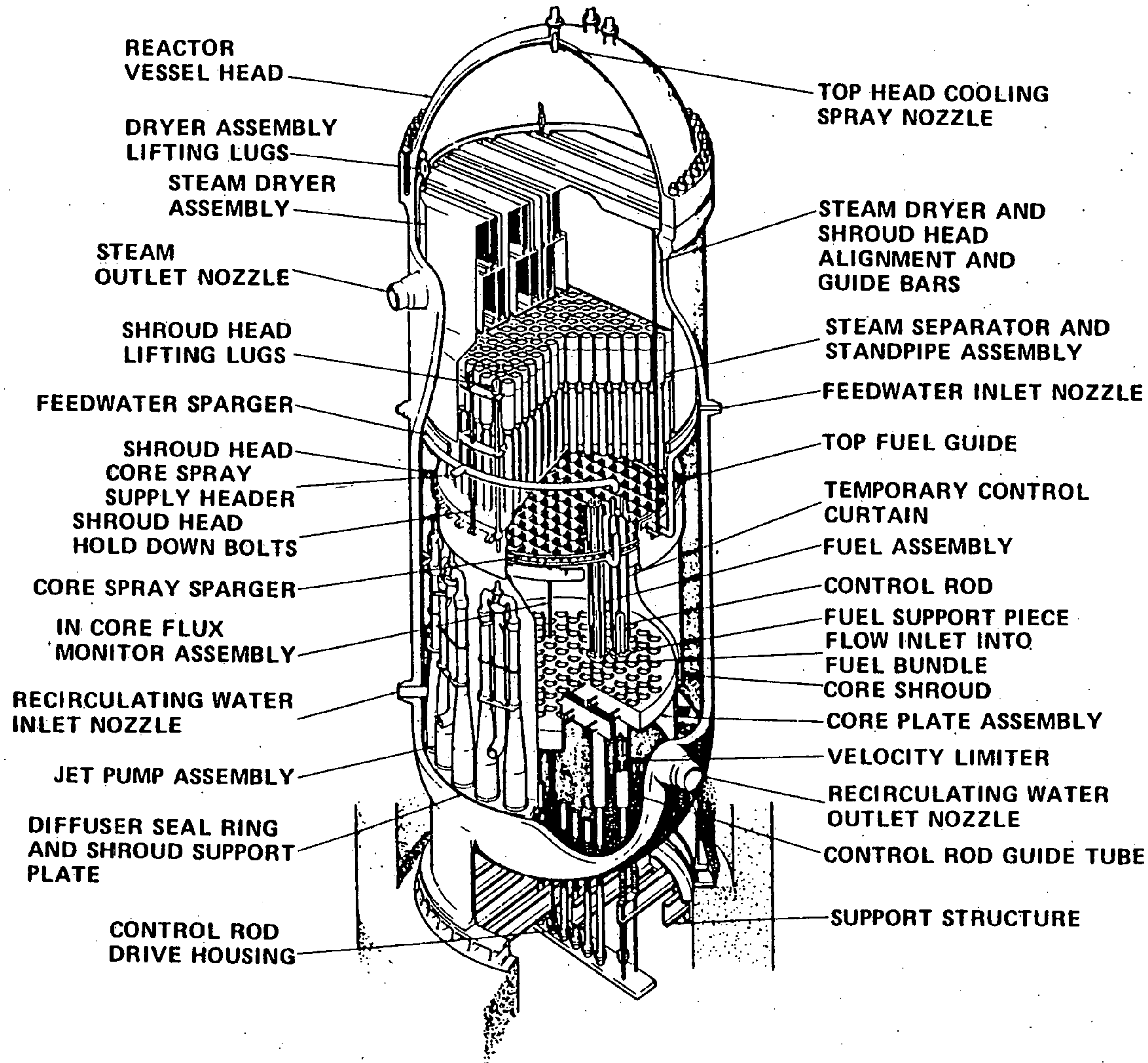


Figure B5. BWR Primary Vessel Internals

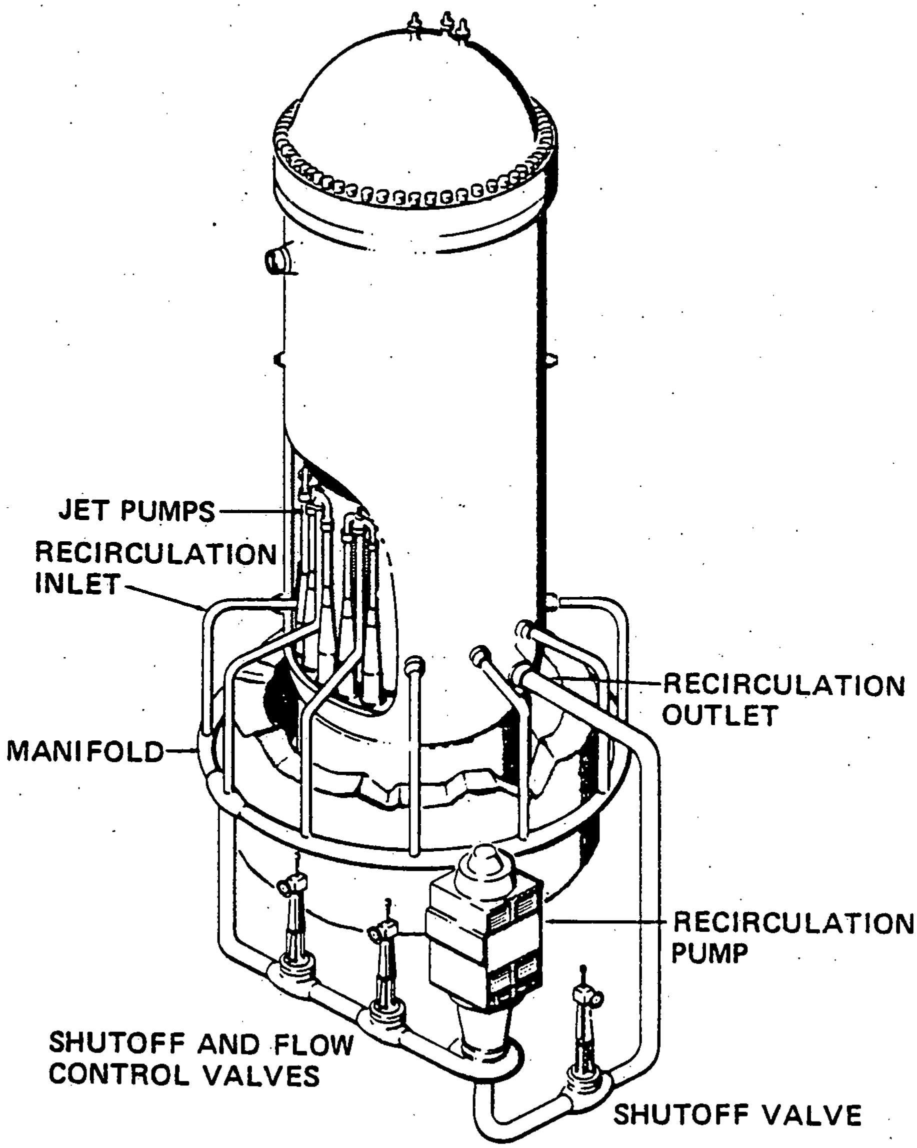


Figure B6. BWR Primary Vessel and Cooling System

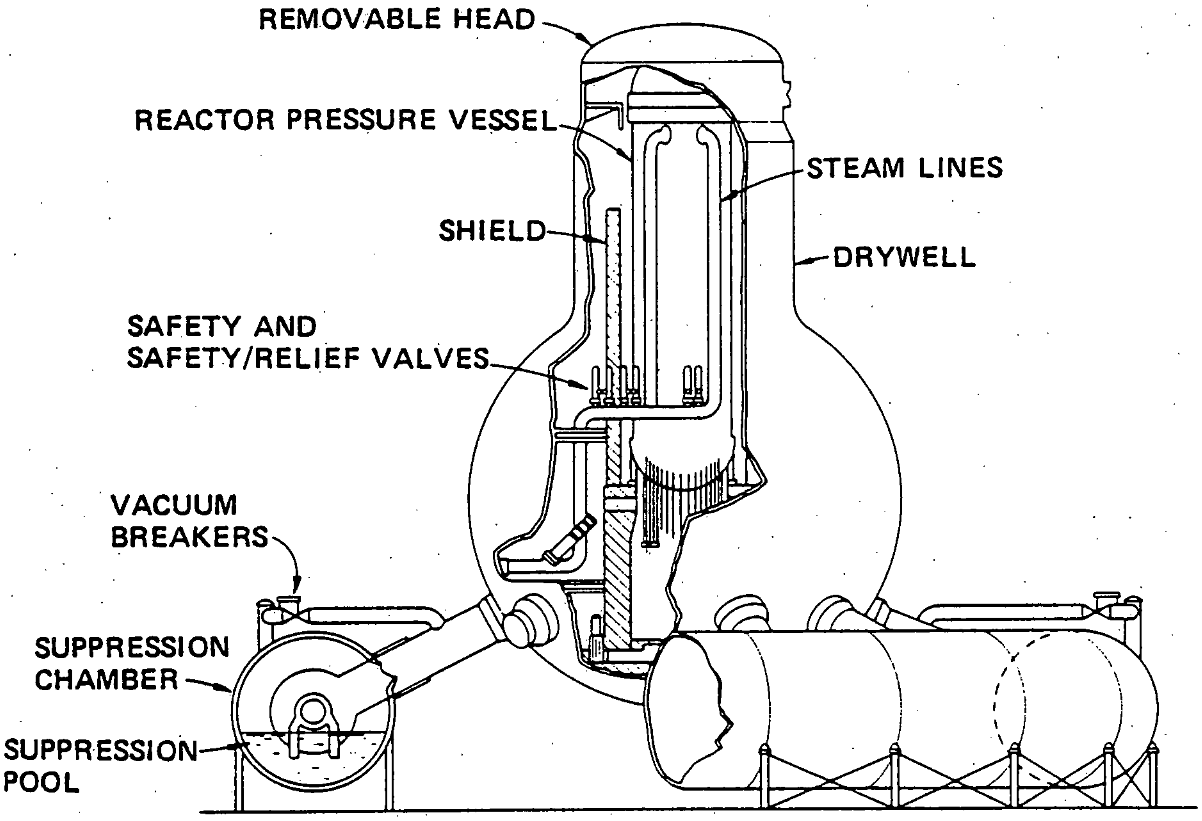


Figure B7. BWR Containment System (Brown's Ferry)

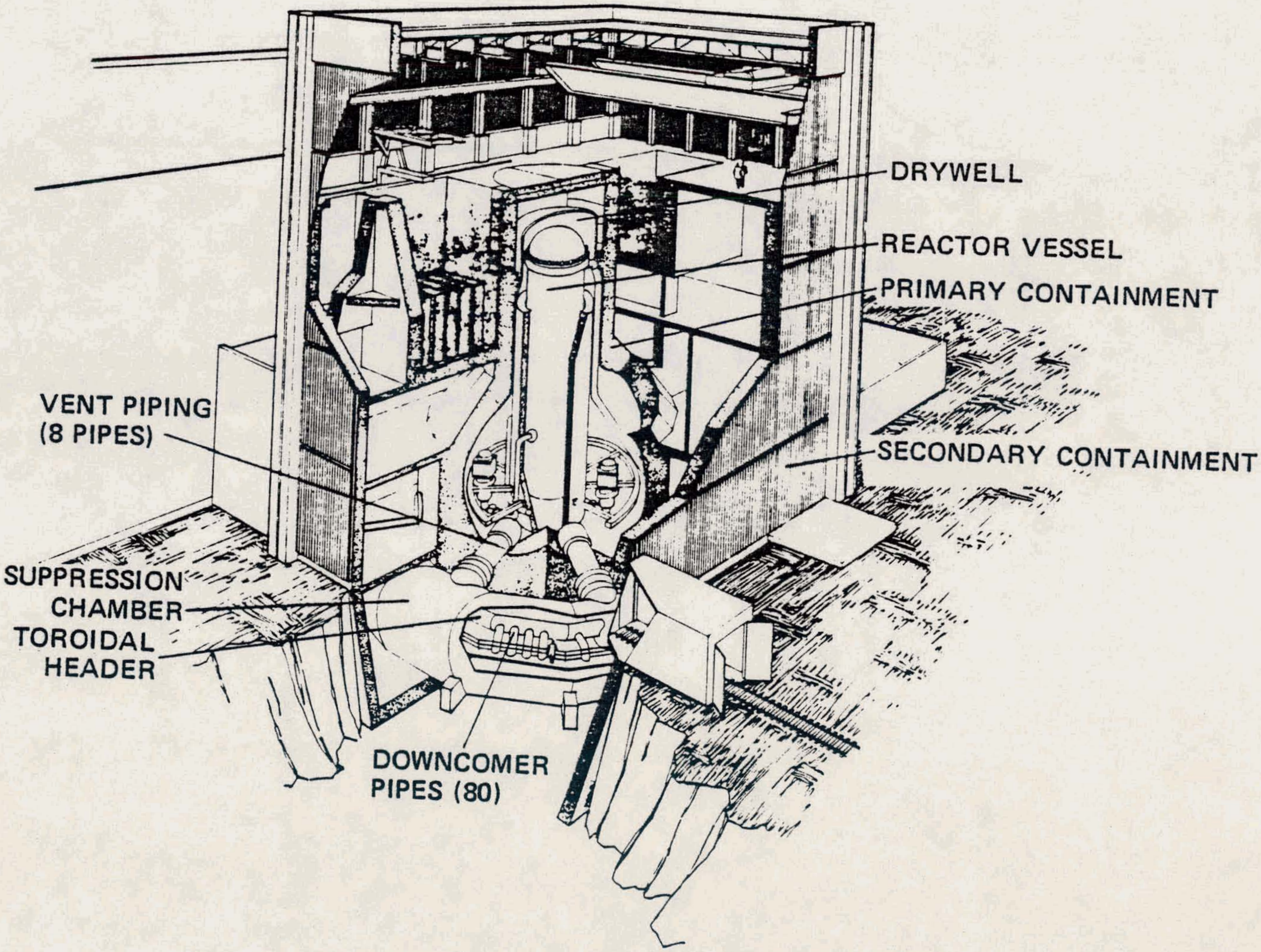


Figure B8. BWR Reactor Building Showing Primary Containment System Enclosed