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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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DOCUMENTATION OF THE TREATMENT OF THE LEVEL 1/LEVEL 2 INTERFACE IN PSAs, WITH EMPHASIS ON ACCIDENT MANAGEMENT ACTIONS

62823

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- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
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The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of all OECD Member countries except New Zealant and Poland. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of the NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

- *encouraging harmonization of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;*
- *assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;*
- *developing exchanges of scientific and technical information particularly through participation in common services;*
- *setting up international research and development programmes and joint undertakings.*

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA) is an international committee made up of senior scientists and engineers. It was set up in 1973 to develop, and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety among the OECD Member countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of the programme of work. It also reviews the state of knowledge on selected topics on nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus on technical issues of common interest. It promotes the co-ordination of work in different Member countries including the establishment of co-operative research projects and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences and specialist meetings.

The greater part of the CSNI's current programme is concerned with the technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the nuclear fuel cycle, conducts periodic surveys of the reactor safety research programmes and operates an international mechanism for exchanging reports on safety related nuclear power plant accidents.

In implementing its programme, the CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also cooperates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

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Requests for additional copies of this report should be addressed to:

Nuclear Safety Division
OECD Nuclear Energy Agency
Le Seine St-Germain
12 blvd. des Iles
92130 Issy-les-Moulineaux
France

ABSTRACT

By defining the initial and boundary conditions for the level-2 analysis, plant damage states provide the interface between level-1 and level-2. An important element of plant damage states are accident management actions considered in the level 1 analyses, which have the potential to significantly influence the further progression of the accident as well as the feasibility, effectiveness and success probabilities of severe accident management measures in the level 2 domain. In this report the treatment of such actions in 11 published PSAs is described.

FOREWORD

The NEA Committee on the Safety of Nuclear Installations (CSNI) believes that an essential factor in achieving their mandate is the continuing exchange and analysis of technical information. To facilitate this exchange CSNI has established various working group. To deal with technology and methods for identifying contributors to risk and assessing their importance, the Committee established Principal Working Group No. 5 - Risk Assessment in 1982. In 1987, "the Committee supported a suggestions that PWG5's activity should for the moment be primarily focused on PSA Level 1 methods, uses and assessments ...... (i.e., to consideration of PSA Level 2 issues where appropriate".

Over the last 10 years the scope of PSA programmes increased progressively to where today, in many countries, a Level 2 PSA is considered the normal standard. Accordingly, with the advent of increasing use of PSAs, a proposal was made at the 1993 PWG5 Annual meeting for future work in the area of Level 2 PSA. The main objective of the proposed task was to perform a state-of-the-art review of the methods available for performing level 2 PSAs and severe accident/source term uncertainty analyses for use in the regulatory process and the evaluation/implementation of severe accident management strategies. This proposal was accepted by PWG5 and forwarded to the CSNI. The new task was endorsed by CSNI during its annual meeting in 1993.

The overall scope of the task included review current Level 2-PSA methodologies and practices and to investigate how Level 2-PSA can support severe accident management programmes, i.e. the development, implementation, training and optimisation of accident management strategies and measures. The final product is contained in CSNI Report OCDE/GD/(97)198 published in late 1997. For the most part, the presented material reflects the state-of-the-art in 1996.

The information contained within this report reflects (along with three other reports) supplemental material which was prepared in conjunction with the main report. This specific report looks interfaces between Level 1 and Level 2 PSAs with special emphasis on accident management actions.

Much appreciation and thanks go to ther task group members listed below, who provided valuable time and considerable knowledge into this report. Special acknowledgement is given to Dr. Wolfgang Werner, who as an expert consultant provided much of the in-depth technical analysis provided throughout the report as well as many man-hours in editing and compiling the final report.

Task Group Member contributing to the report were:

Cojazzi, G.	Italy	Kersting, E.	Germany	Muramatsu.K.	Japan
Cozzoli, E.	Switzerland	Kim, T.W.	Korea	Murphy, J.A.	United States
Cunningham, M.	United States	Lantaron, J.	Spain	Otero, M.	Spain
Evrard, J.M.	France	Lee, C.J.	Korea	Seebregts, A.	Netherlands
Grant, A.	Canada	Liwang, B.	Sweden	Shepherd, C.	United Kingdom
Herttrich, P.M.	Germany	Meyer, P.	Switzerland	Versteeg, M.F.	Netherlands
Hirose, H.	Japan			Werner, W.	Germany

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1. INTRODUCTION

Level-2 PSAs examine the responses of the containment and of its engineered safety systems to the loads attending core damage accidents. Results of level-2 PSAs are:

- containments failure states and their frequencies of occurrence,
- the releases to the environment of radioactive substances attending containment failure states.

The input to the analysis is produced in a preceding level-1 PSA. It consists of the set of failure states, including frequencies of occurrence, of the reactor system and of active containment systems that are relevant to the containment analysis. The elements of this set are called plant damage states (PDS). They provide the interface between level-1 and level-2 analyses by defining the initial and boundary conditions for the level-2 analysis. Criterion for the definition of the plant damage states is the similarity of the failure states with regard to the:

- further progression of the accident in the reactor system,
- functionality of the active containment systems, and
- response of the containment structure to the attendant loads.

In the level 2 analysis it is necessary to take into account the influence of accident management actions that are considered in the level 1 analyses. Such actions can significantly influence the characteristics of plant damage states, the further progression of the accident as well as the feasibility, effectiveness and success probabilities of severe accident management measures in the level 2 domain. Therefore, the inclusion of failed preventive (level 1) accident management in the development of the plant damage states is an important aspect of the definition of the level 1/level 2 interface. An important aspect of the back end of the level 2 analyses, is whether the characteristics used for the grouping of the end states of the accident progression analysis permit to trace the influence of accident management actions all the way to the release categories.

The following plants are included in this report:

- PWRs Surry, Sequoyah, Maine Yankee, Beznau, Ringhals2/3/4.
- BWRs: Peach Bottom, Grand Gulf, Browns Ferry, Perry, Mühleberg, Swedish BWRs (level-2 PSAs not yet available)

The described issues are examined on basis of the PSAs and related literature listed in the reference section.

2. IDENTIFICATION OF RECOVERY AND ACCIDENT MANAGEMENT ACTIONS IN THE LEVEL-1 DOMAIN THAT ARE CRITICAL TO ACCIDENT PROGRESSION AND TO SEVERE ACCIDENT MANAGEMENT IN THE LEVEL-2 DOMAIN

Success or failure of a number of recovery and accident management actions in the level 1 domain is relevant for the accident progression analysis in the level 2 domain. It is therefore important that the information on such actions is available for the level 2 analysis by proper inclusion in the plant damage states. Below is a list of actions that were found to be important in the examined level 2 analyses. For each action, the HRA method used in the respective studies is indicated.

2.1 Pressurised Water Reactors

- Maintaining the availability of the service water system for containment spray recirculation during loss of offsite power events at Surry. HRA method: simplified THERP [21]. Surry has a gravity feed service water system which relies on the head difference between the intake and discharge canals. The intake canal is resupplied with water by the recirculating service water pumps. These are unavailable during loss of offsite power.

  In the event that a condenser fails to isolate, the outflow through the condenser is greater than the makeup capability of the diesel driven emergency service water pump, potentially leading to canal drainage before the restoration of offsite power. To maintain service water availability, the condenser(s) can be isolated by manually closing the isolation valves(s).

- Restoration of the bus powering the containment residual heat removal train pumps, following loss of offsite power events at Surry. HRA method: simplified THERP [21]. This bus must be manually reconnected after load shed during loss of offsite power events.

- Initiation of bleed/feed (all plants). HRA method: simplified THERP [21] in NUREG-1150, Dutch and Swedish studies, modified SLIM [22] in the Beznau and Maine Yankee studies.

  At all plants included in this study procedures and hardware are in place for conducting primary side bleed/feed. Failures to initiate bleed/feed are relevant in the level 2 domain because of their impact on the possibility to mitigate high pressure core melt sequences. By the application of the depressurization procedure it is intended to prevent DCH phenomena that could threaten the containment integrity.

- Initiation of secondary side bleed (to reach conditions for secondary side feed) at Maine Yankee. HRA method: modified SLIM [22]

- Diagnosis of steam generator tube rupture and identification of the ruptured steam generator (All plants). HRA method: simplified THERP [21] in NUREG-1150, Swedish studies, modified SLIM [22] in the Beznau and Maine Yankee studies

- Initiation of primary side bleed/feed in the event of steam generator tube rupture. In the event of core damage involving steam generator tube rupture with unisolated steam generator, the release to the environment of the volatile fission products, including noble gases, can be significantly reduced if primary side bleed/feed is applied: through the split-up of the mass flow between the broken steam generator tube (few cm^2) and the open pressurizer valves (from 20 cm^2 upward), most of the fission products released from the core are directed to the containment (Surry, Zion, Ringhals 2/3/4, Borssele, Beznau). HRA method: simplified THERP [21] in NUREG-1150, Swedish studies, modified SLIM [22] in the Beznau study

- Filling of ruptured steam generator with water. The releases from a ruptured tube in an unisolated steam generator can be very significantly reduced by scrubbing of gases in a column of water in the defective steam generator. The scrubbing effect strongly depends on the height of the water column above the break (Beznau, Ringhals 2/3/4, under study for Biblis-B). HRA method: simplified THERP [21] in the Swedish studies, modified SLIM [22] in the Beznau study

- Recovery from CCF of the containment spray heat exchanger valves by locally opening or repairing the valves at Surry. HRA method: simplified THERP [21]

- Diesel generator recovery (Surry, Sequoyah, other US plants). HRA method: simplified THERP [21]

- Establish alternate ESF pump room cooling by using portable coolers and fans (Sequoyah). HRA method: simplified THERP [21]

- Locally open the SWS motor operated valves to containment spray system heat exchangers (Sequoyah). HRA method: simplified THERP [21]

- External water injection to the containment.

- Additional water injection to the containment can be used for prevention of core damage and for mitigation of the consequences of core damage.

- Prevention: Backup water sources for:

    - low pressure injection/recirculation (Swedish PWRs, using fire trucks (CWIS)), HRA method: simplified THERP [21].

    - containment spray (Beznau and Swedish PWRs, with injection from fire trucks), HRA method: simplified THERP [21] in the Swedish studies, modified SLIM [22] in the Beznau study.

    - cooling of containment fan coolers (Beznau, using river water and mobile pumps). HRA method: modified SLIM [22].

- Mitigation: Water supply for flooding of the containment when core damage and possible RPV failure is imminent. (Swedish PWRs, Beznau). HRA method: simplified THERP [21] in the Swedish studies, modified SLIM [22] in the Beznau study. By having a deep water pool underneath the reactor vessel, the extent of basemat attack by molten core debris can be reduced or basemat attack may even be prevented, thus reducing or eliminating the production of combustible gases, as well as the likelihood of basemat penetration.

2.2 Boiling Water Reactors

- Diesel generator recovery (Peach Bottom, Grand Gulf, Browns Ferry, Perry, Mühleberg). HRA method: simplified THERP [21] in NUREG-1150 and Perry studies, modified SLIM [22] in the Mühleberg and Browns Ferry studies

- Offsite power recovery (Peach Bottom, Grand Gulf, Browns Ferry, Perry, Mühleberg). HRA method: simplified THERP [21] in NUREG-1150 and Perry studies, modified SLIM [22] in the Mühleberg and Browns Ferry studies

- Manual start of standby liquid control system in ATWS situation (all examined BWRs). HRA method: simplified THERP [21] in NUREG-1150 and Perry studies, modified SLIM [22] in the Mühleberg and Browns Ferry studies

- Manual alignment and actuation of backup injection, for example, fire water system, control rod drive system, high pressure service water system, river water, condensate system with feedwater pump bypass, after failure of the engineered injection systems, (all examined BWRs). HRA method: simplified THERP [21] in NUREG-1150 and Perry studies, modified SLIM [22] in the Mühleberg and Browns Ferry studies

- Manual depressurisation of RPV to reach conditions for low pressure injection after failure of high pressure injection and automatic depressurisation. (all examined BWRs). HRA method: simplified THERP [21] in NUREG-1150, and Perry studies, modified SLIM [22] in the Mühleberg and Browns Ferry studies

- Containment venting for RHR. In some plants the same venting system is used for residual heat removal and for overpressure protection of the containment (Peach Bottom, Grand Gulf, Browns Ferry, Perry, Swedish BWRs, all German BWRs except Würgassen, Dodewaard). HRA method: simplified THERP [21] in NUREG-1150and Perry studies, modified SLIM [22] in the Mühleberg and Browns Ferry studies

- Manual actuation of late RPV injection for pedestal cavity flooding (Perry). HRA method: simplified THERP

- Manual alignment and actuation of the external water injection system for flooding of the pedestal area. (Swedish BWRs, Mühleberg). HRA method: modified SLIM

- Manual actuation of flooding of the pedestal cavity from the wetwell (Swedish BWRs).

3. PLANT DAMAGE STATE DEFINITION AND ANALYSIS

The principal step of a level 1 analysis identifies the dominant event sequences that lead to core damage. The final stage of the event tree analysis process maps the dominant core damage sequences into plant damage states (PDSs).

The plant damage state analysis involves the identification of detailed PDS categories using multi-state indicators. The resultant number of plant damage states is usually large and difficult to manage in the containment event tree quantification process. Therefore, the plant damage states are grouped into more manageable plant damage state groups.

In the NUREG-1150 analyses, the ERI/HSK analysis for Beznau, the Ringhals 2 analysis, and in the Perry IPE, the frequencies of the multi-state indicators are passed on to the containment analysis in which they are used to calculate the split fractions in the containment event tree quantification process. In other studies, condensed information that is representative of the plant damage state groups, is passed on to the containment event tree quantification process.

3.1 Systems Event Trees/Plant Damage State Analysis

In the event trees developed for the systems analysis stage, only those events and systems failures are examined that are needed to determine whether or not the accident sequences would lead to core damage. This includes failures of containment systems that can put the plant in a core vulnerable state in which core damage can be caused indirectly as a consequence of the containment failures. For these examinations, it is not necessary to know which containment system (or which combination of containment systems) failed - only that some form of failure occurred. Thus, the status of the containment systems is simplified by using only a single top event to track all forms of containment heat removal failure. Additional event tree headings are added to further distinguish containment spray and containment heat removal systems failures.

In the plant damage state analysis in NUREG-1150, the Ringhals 2 analysis, and in the Perry IPE, event trees are developed that generate the information needed to assess the degree to which the containment systems remain operable as a means of preserving containment integrity and preventing or reducing the amount of radionuclide release following core damage. The plant damage state event trees include such headings as:

- containment spray injection,
- containment heat removal by the containment spray recirculation system
- high/low pressure recirculation system

(for PWRs)

- containment heat removal with RHR spray loop,
- containment heat removal with RHR suppression pool cooling,
- containment heat removal with venting,
- late RPV depressurisation,
- late RPV injection for pedestal cavity flooding

(for BWRs)

In the ERI/HSK analysis for Beznau, the information needed in the containment analysis is provided by a fault tree linking code that links the fault tree data for safety system with relevance for containment response to the general plant damage state cutset files. This includes information related to containment spray injection, containment heat removal by the containment spray recirculation system, the high/low pressure recirculation system, external water supply to the containment sump.

3.2 Definitions of the Plant Damage State Indicators for PWR Plants

3.2.1 NUREG-1150 Analysis for Surry and Sequoyah Plants

Seven indicators are used to identify a plant damage state. They address the following issues:

- Status of RCS at onset of core damage.
- Status of ECCS.
- Status of containment heat removal capability.
- Status of AC power.
- RWST injection capability.
- Steam generator heat removal capability.
- Status of RCP seal cooling.

Containment isolation failures are considered negligible for the PWR plants examined in the NUREG-1150 studies; therefore, they are not included among the plant damage state indicators (in contrast to other PSAs).

Each of the seven indicators is discussed below.

– Status of RCS at Onset of Core Damage.

This indicator provides information on the pressure of the reactor coolant system, and its integrity at the time of vessel failure.as the expected RCS pressure is related to RCS integrity. Eight categories of the RCS integrity status are identified and related to the initiating events, as shown below.

T no break (transient).
A large LOCA (6” to 29”).
S1 medium LOCA (2” to 6”).
S2 small LOCA (1/2” to 2”).
S3 very small LOCA (< 1/2”).
G steam generator tube rupture with steam generator integrity.
H steam generator tube rupture without steam generator integrity.
V interfacing LOCA.

The first character in the PDS designator is commonly referred to as the initiating event, however, the way it is used in the containment event tree (CET) analysis is to indicate the integrity of the RCS at the onset of core damage. Hence, the first character in the PDS designator may differ from the sequence initiating event. For example, if the initiating event is a transient with a RCP seal failure occurring before the onset of core damage, then the CET would treat this case as a small break in classifying the status of the RCS.

– Status of ECCS.

Indication of the past and present status of high and low pressure injection or recirculation cooling. Five categories are identified relative to the ECCS, as shown below.

A operated in injection only.
B operated in injection, not operating in recirculation.
R not operating, but recoverable.
N not operating and not recoverable.
L HPI failed, but LPI operable if pressure is reduced.

– Status of Containment Heat Removal Capability.

Indication of whether or not containment heat removal is available. For plant damage state definition, this is defined to be the availability of at least one containment spray train (at Surry) or at least one containment spray or LHR train (at Sequoyah) in the recirculation mode, with service water being supplied to the heat exchanger. The alternate means of containment heat removal (via AFW) included in the systems analysis stage event tree would not be available after vessel failure. Four categories are used for this indicator, as shown below. For this indicator it is not always possible to identify a unique state from the sequence outcome. Split fractions were developed to partition containment failure states into plant damage states.

Y operating or operable if/when needed.
R not operating, but recoverable.
N never operated, not recoverable.

S sprays operable, but no CHR (no SW to HXs) .

– Status of AC Power.

Indication of whether or not the AC power needed for safety systems is available. Two status categories are identified for this indicator.

Y available.
R not available, but recoverable.
N not available, not recoverable.

– RWST Injection Capability.

Indication of whether or not the reactor cavity is full of water. In order to assure that the cavity is full of water, the RWST must be fully injected into the containment. No partial credit is taken for RWST injection. Three categories are identified:

Y fully injected into containment.
R not fully injected, but could be injected with power recovery.
N not fully injected, cannot be injected in future.

– Steam Generator Heat Removal Capability.

Indication of the status of the AFW system and its ability to provide steam generator heat removal. Six status categories were used for this indicator.

X at least one AFWS operating, SGs not depressurized.
Y at least one AFWS operating, SGs depressurized.
C steam driven pump operated until battery depletion, electric driven pump recoverable with power recovery - SGs not depressurized.
D steam driven pump operated until battery depletion, electric driven pump recoverable with power recovery - SGs depressurized.
S steam driven pump failed at beginning, electric driven pump recoverable with power recovery.
N no AFWS operating, no AFWS recoverable.

– Status of RCP Seal Cooling.

Indication of the availability of cooling to the RCS pump seals, which provides a direct measure of the ability to preserve the reactor coolant pressure boundary at the reactor coolant pump seals. Three status categories were used for this indicator.

Y operating.
R not operating, but recoverable.
N not operating and not recoverable.

With the number of attributes possible for each of the seven PDS indicators, there are potentially 25,920 different plant damage states.

All core damage sequences greater than l E-7/yr. are assigned to the appropriate plant damage state, and. all PDSs with frequencies greater than l E-7/yr. are retained for containment event tree analysis. If any PDS between l E-9 and l E-7 represents a substantially more severe containment state than any of the PDSs above l E-7/yr., it is also retained for further analysis.

3.2.2 ERI/HSK Analysis for the Beznau Plant

A total of five indicators are used to identify a plant damage state. The five indicators address the following issues:

- Status of RCS at onset of core damage.
- Status of ECCS, including RWST injection capability.
- Status of containment isolation (excluding containment bypass scenarios).
- Status of containment heat removal capability.
- Status of containment integrity (containment bypass scenarios).

Each of these indicators is discussed below.

- Status of RCS at Onset of Core Damage.

This indicator provides information on the pressure of the reactor coolant system, and its integrity at the time of vessel failure, as the expected RCS pressure is related to RCS integrity. Three categories of the RCS integrity status are identified and related to the initiating events, as shown below.

T no break (transient).
A large or medium LOCA (> 2”).
S small LOCA (< 2”).

The first character in the PDS designator is commonly referred to as the initiating event, however, the way it is used in the containment event tree (CET) analysis is to indicate the integrity of the RCS at the onset of core damage. Hence, the first character in the PDS designator may differ from the sequence initiating event. For example, if the initiating event is a transient with a RCP seal failure occurring before the onset of core damage, then the CET would treat this case as a small break in classifying the status of the RCS. In this context, LOCA events also include steam generator tube rupture and interfacing system LOCA events.

– Status of ECCS.

Indication of the past and present status of high and low pressure injection or recirculation cooling, including status of RWST injection. Three categories are identified relative to the ECCS, as shown below.

E Failure of the ECCS in the injection mode (Beznau has only high pressure injection), no RWST injection (early core damage).
E' RWST injection following core damage at or after RPV failure (early core damage).
L Failure of ECCS in the recirculation mode (late core damage).

– Status of Containment Isolation.

Indication of containment isolation, excluding containment bypass scenarios.

I Containment isolated.
U Containment unisolated.

– Status of Containment Heat Removal Capability.

Indication of whether or not engineered containment safety systems (containment spray and fan coolers) are available.

C Containment spray available for both injection and recirculation.
C' Containment spray available for injection only.
C" Containment spray available for recirculation only.
F Containment fan coolers are available.

Unavailability of the system functions is designated by absence of the respective letters.

– Status of Containment Integrity.

V1 Interfacing systems LOCA with isolated containment.
V2 Unisolated steam generator tube rupture.
V3 Interfacing systems LOCA with unisolated containment.
Unisolated reactor coolant pump seal return or letdown line LOCA.

With the number of attributes possible for each of the five PDS indicators, there are potentially 288 different plant damage states.

The 26 PDSs with an upper 95% quantile greater than l E-8/yr. are retained for containment event tree analysis.

3.2.3 Maine Yankee IPE Analysis

A total of five indicators are used to identify a plant damage state. The five indicators address the following issues:

- RCS pressure at onset of core damage.
- Status of RWST injection capability.
- Status of containment isolation (excluding containment bypass scenarios).
- Status of containment heat removal capability.
- Status of containment integrity (containment bypass scenarios).

Each of these indicators is discussed below.

- RCS Pressure at Onset of Core Damage.

Three pressure ranges are considered:

$p < 200$ psia,
200 psia $< p <$ 1800 psia,
$p > 1800$ psia.

In the low pressure range, conditions are assumed to be created that allow core debris to enter the cavity without forcible ejection, possibly resulting in increased concrete attack.

In the medium pressure range, conditions are assumed to be created that

- suppress in-vessel steam explosion,
- lead to forcible ejection of molten fuel at a limited rate,
- can lead to DCH phenomena,
- lead to rapid ejection of hydrogen accumulated inside the reactor vessel.

In the high pressure range, hot leg rupture is assumed to be likely.

- Status of RWST Injection Capability.

Indication of the past and present status of RWST injection. Two categories are identified, as shown below:

RWST injection before the molten core attacks the reactor vessel lower head.
RWST injection after the molten core attacks the reactor vessel lower head.

– Status of Containment Isolation.

Indication of containment isolation, excluding containment bypass scenarios:

Containment isolated.
Containment unisolated.

– Status of Containment Heat and Fission Product Removal Capability.

Indication of whether or not engineered containment safety systems (containment spray in the injection and recirculation mode, with or without heat removal) are available.

Containment spray available for both injection and recirculation, with heat removal.
Containment spray available for injection and recirculation, without heat removal.
Containment spray available for recirculation only, with heat removal.
Containment spray unavailable.

– Status of Containment Integrity.

Small containment bypass.
Large containment bypass.

With the number of attributes possible for each of the five PDS indicators, there are 70 physically possible different plant damage states.

These are condensed to 12 plant damage state groups (called key plant damage states). It is stated that the key plant damage states are conservative representations of the 70 possible plant damage states. Input to the containment event tree analysis is one sequence for each key plant damage state. The selected sequence is considered to be representative of the dominant sequences included in the corresponding key damage state.

3.3 Plant Damage State Analysis for Pressurised Water Reactors

3.3.1 NUREG-1150 Analysis for Surry, Sequoyah and Zion

In the Surry analysis, there are 37, and in the Sequoyah analysis there are 34 individual core damage sequences, with point estimate frequency above l E-7/yr. after recovery actions. Each of these are assigned to plant damage states. Figures 3.3.1-1 - 3.3.1-4 show examples of this assignment process. Depicted are the plant damage state event trees (which are expansions of the core damage event trees that incorporate containment functions) for the loss of feedwater and small LOCA events at the Surry and Sequoyah plants.

3.3.2 ERI/HSK Analysis for Beznau

The assignment of plant damage states is based on the following assumptions:

- Containment pressure will exceed 2.31 bar (i e the containment spray automatic initiation set point) immediately following all large LOCAs.

- If the containment is unisolated containment pressure will not reach 2.31 bar even following vessel breach for any events (i.e. transients small LOCAs or large LOCAs).

- Failure of the safety injection signal is included in the containment spray fault tree to account for failure of the high containment pressure signal for automatic containment spray actuation For those damage states with containment sprays unavailable a recovery probability can be applied to account for the fraction of the PDS frequency for which sprays can be manually.

The PDS assignments include the electric power dependencies for the containment sprays.

All excessive LOCAs are grouped with large LOCAs with late core damage.

The link between the fault tree data for the analysis of core damage sequences and plant damage state cut set files is provided by a fault tree linking code.

3.3.3 Maine Yankee IPE Analysis

All core damage sequences with frequency > 1 E-8 are assigned to the appropriate plant damage states, according to the attributes of the plant damage state indicators.

3.4 Regrouping of Plant Damage States for Pressurised Water Reactors

3.4.1 NUREG-1150 Analysis for Surry, Sequoyah and Zion

The individual plant damage states are assigned to seven plant damage state groups in order to facilitate the quantification of the containment event tree. The grouping of the PDSs is shown in Tables 3.4.1-1 and 3.4.1-2 for Surry and Sequoyah, respectively.

Table3.4.1-1. Plant Damage State Groups for Surry

Group Number and Name	Plant Damage States assigned to Group
1 Slow Station Blackout	TRRR-RDY S_3RRR-RDR S_2RRR-RDR TRRR-RDR S_2RRR-RCR S_3RRR-RCR
2 LOCAs	S_1IYY-YYN S_1NYY-YYN AIYY-YYN S_1LYY-YYN ALYY-YYY S_3LYY-YYN S_2LYY-YYN ANYY-YYN
3 Fast Station Blackout	TRRR-RSR
4 Event V	V
5 Transients	TBYY-YNY TLYY-YNY
6 ATWS	S_3NYY-YXN TLYY-YXY GLYY-YXY
7 Steam Generator Tube Rupture	HINY-NXY GLYY-YXY GLYY-YNY HINY-YXY

Table 3.4.1-2 Plant Damage State Groups for Sequoyah	
Group Number and Name	Plant Damage States assigned to Group
1 Slow Station Blackout	TRRR-RDR S_3RRR-RDR S_2RRR-RDR S_2RRR-RCR S_3RRR-RCR
2 Fast Station Blackout	TRRR-RSR S_2RRR-RSR
3 LOCAs	S_1INY-YYN S_1LYY-YYN AIYY-YYN AINY-YYN S_1IYY-YYN ALYY-YYY S_3LYY-YYN S_3IYY-YYN S_3INY-YYN S_2LYY-YYN S_2INY-YYN ANYY-YYN
4 Interfacing LOCA	V
5 Transients	TBYY-YNY
6 ATWS	S_3NYY-YXN TLYY-YXY GLYY-YXY
7 Steam Generator Tube Rupture	HINY-NXY GLYY-YNY

Below, the grouping is discussed for Surry with some additional remarks on other US plants.

PDS Group 1 consists of six Slow Blackout PDSs. In these accidents, offsite power is lost and the diesel generators fail to start or run. The steam-turbine-driven AFW pump is available until the batteries are depleted, thus failing power for instruments and controls. Battery depletion is estimated to take about 4 hours. For some sequences in group 1, the RCP seals may fail or the PORVs may stick open. Thus, the six PDSs in this group have the RCS in different conditions when core damage begins.

In two of the PDSs in this group, the RCS is intact at the time of core uncovering. Another two of the PDSs have S3-size breaks (failures of the reactor coolant pump seals), and the final two PDSs in this group have S1-size breaks (stuck-open PORVs).

The difference between the two "T" PDSs in Group l is whether there is cooling to the RCP seals. The difference between the two "S3" PDSs is whether the secondary system is depressurized before the core uncovers and while the AFW is operating.

Relevant recovery and accident management actions to be included in the PDS group are 2.1.1, 2.1.2, 2.1.3 and 2.1.9.

PDS Group 3 consists solely of TRRR-RSR - Fast Blackout. In this accident, auxiliary feedwater fails to start and run. The dominant failures occur early in the sequence, thus the "fast" blackout nomenclature. Core damage is likely before failure of the RCP seals.

Relevant recovery and accident management actions to be included in the PDS group are 2.1.1, 2.1.2, 2.1.3 and 2.1.9.

PDS Group 2 consists of seven LCD PDSs. Four of the PDSs have an A-size break, and two of the PDSs have an S1-size break. There is one PDS with an S2-size break and one PDS with an S3-size break. Four of the PDSs in this group have the LPIS operating. In PDS ALYY-YYY, the accumulators have failed and the LPIS is operating successfully (all trains). For an A break, the success criteria require both accumulator injection and LPIS operation. Thus, even though the RCS pressure is low and the LPIS is injecting water successfully, core damage is assumed. In PDS SlLYY-YYY, HPIS has failed and the LPIS is operating successfully (all trains). For an S1 break, the success criteria require HPI early in the accident and LPIS operation later. In this PDS also, the RCS pressure is low and the LPIS is injecting water successfully, but core damage is assumed since the success criteria are not been met. In PDS S2LYY-YYY and S3LYY-YYY, the break does not depressurise the RCS enough to allow LPI. Thus the accident will progress to vessel failure at a pressure too high to allow LPI unless a large temperature induced break occurs or the primary system is deliberately depressurized.

PDS Group 4 consists solely of Event V. This is a large break in low pressure piping following the failure of the two check valves that isolate the low pressure piping from the RCS. The break is outside containment in the auxiliary building, so the break both fails the RCS pressure boundary and bypasses the containment.

PDS Group 5 consists of two PDSs that have failure of both AFW and bleed and feed. In PDS TBYY-YNY, both LPI and HPI are available and the PORVs are not opened. In PDS TLYY-YNY, only LPI is available. All AFW is failed and bleed and feed is not possible because the HPIS is failed. As all sources of feedwater are lost, it is not possible to depressurize the RCS. Some plants have procedures for emergency feed of the steam generators using fire water system. These efforts are given little chance of success for two reasons.

Many of the failures contributing to the sequence are operator errors, thus compounding the probability of subsequent errors.

The timing of the sequence leaves very little time to establish fire water to the SG, given previous failed attempts to restore feedwater and establish feed and bleed.

Consideration of these two factors resulted in not al lowing credit for further recovery from these sequences. Similarly, no credit was given for depressurizing the RCS by operator action after the onset of core damage in PDS Group 5.

Since there is RCP seal cooling, and SGTRs are not very likely, the only effective means of depressurizing the RCS are the PORVs/SRVs sticking open or the failure of the hot leg/surge line. If the RCS pressure decreases to the low range, the LPIS will inject.
Relevant recovery and accident management action to be included in the PDS group is 2.1.3.

PDS Group 6 contains the three ATWS PDSs. They differ in the status of the RCS at the time the core uncovers, whether the ECCS worked in the injection phase, and in whether cooling for the RCP seals is operating or failed. In this group also, the LPIS is available in some of the PDSs, and will inject of the RCS reaches low pressure.

PDS Group 7 consists of three PDSs that are initiated by SGTRs and which do not have scram failures. HINY-NXY is an SGTR with stuck-open SRVs in the secondary system. GIYY-YNY has the RCS PORVs open, since the operators are attempting to keep the core cooled by feed and bleed. It might have been denoted (G-S2) IYY-RNYY. HINY-NXY has no RWST water being injected into the containment. In the other two PDSs, while some of the water lost from the RCS is released from the containment through the ruptured steam generator, much more water is directed to the containment through the PORVs. Relevant recovery and accident management actions to be included in the PDS group are 2.1.5 and 2.1.6.

3.4.2 ERI/HSK Analysis for Beznau

The 26 plant damage states retained for the containment event tree analysis are further condensed to 11 plant damage state groups according to a binning process that preserves

- the initiating event attributes,
- the containment isolation status,
- the ECCS status.

Thus, the only information that is collapsed in forming the PDS-groups is the containment heat removal system operability attribute. However, during the quantification process for the containment event trees, the information on the containment heat removal systems operability is carried through the analysis.

The mapping of the plant damage states to the plant damage state groups is shown in table 3.4.2-1.

Table 3.4.2-1 Relationship between PDS and PDS Groups

PDS Group	Plant Damage States	Mean Frequency/yr.
1	ALICF, ALIC,ALIC’F	1 E-6
2	AEUC’F, AEUF, ALUCF, ALUC”F, ALUC	6.6 E-8
3	SLICF, SLIC, SLIC’F	6.1 E-7
4	SE’ICF, SE’IC’F, SE’IC’	6.8 E-7
5	TLICF, TLIC’F, TLIC’, TLI	2 E-7
6	TE’ICF, TE’IC’F, TE’IC’	1.9 E-6
7	SEIF	1.5 E-8
8	TEI	5.9 E-6
9	TEU, TEUF, SEUF	7.8 E-7
10	V2	4.3 E-7
11	V4	4.6 E-8

PDS Group 1 consists of three large break LOCA plant damage states with late ECCS failure, isolated containment, and some or all containment heat removal systems available.

PDS Group 2 consists of five large LOCA plant damage states with early/late ECCS failures unisolated containment and some or all containment heat removal systems available.

PDS Group 3 consists of three small LOCA plant damage states with late ECCS failure, isolated containment, and some or all containment heat removal systems available.

PDS Group 4 consists of three small LOCA plant damage states with early ECCS failure, isolated containment, and some or all containment heat removal systems available.
Relevant recovery and accident management action to be included in the PDS group is 2.1.3.

PDS Group 5 consists of four transient plant damage states with late ECCS failure, isolated containment, and some or all containment heat removal systems available.

PDS Group 6 consists of three transient plant damage states with early ECCS failure, isolated containment, and some or all containment heat removal systems available.

PDS Group 7 consists of one small LOCA plant damage state with early ECCS failure, isolated containment, containment spray system unavailable, fan coolers available.
Relevant recovery and accident management action to be included in the PDS group is 2.1.3.

PDS Group 8 consists of one transient plant damage state with early ECCS failure, isolated containment, containment heat removal by containment spray and fan coolers unavailable.

PDS Group 9 consists of two transient and one small LOCA plant damage state with early ECCS failure , unisolated containment, no containment spray available, various failure states of fan coolers. Relevant recovery and accident management action to be included in the PDS group is 2.1.3.

PDS Group 10 consists of one steam generator tube rupture plant damage state with unisolated containment. Relevant recovery and accident management actions to be included in the PDS group are 2.1.5 and 2.1.6.

PDS Group 11 consists of one reactor coolant pump seal LOCA damage state.

3.4.3 Maine Yankee IPE Analysis

The 70 physically possible plant damage states are condensed to 12 plant damage state groups (called key plant damage states). It is stated that the key plant damage states are conservative representations of the 70 possible plant damage states. Input to the containment event tree analysis is one sequence for each key plant damage state. The selected sequence is considered to be representative of the dominant sequences included in the corresponding key damage state. The 12 key plant damage states are characterised below.

Key Plant Damage State 6D represents sequences that are at high pressure (> 1800 psia) at the time of core damage; without any RWST injection either before or after vessel melt-through. Containment is isolated and not bypassed. Containment heat and fission product removal are not available. Relevant recovery and accident management action to be included in the PDS group would be 2.1.3.

Key Plant Damage State 5A represents sequences that are at intermediate pressure (200 psia to 1800 psia) at the time of core damage. The RWST is injected prior to vessel melt-through. Containment is isolated and heat and fission product removal capability are available. Relevant recovery and accident management action to be included in the PDS group would be 2.1.3.

Key Plant Damage State 8A represents sequences that are at high pressure (>1800 psia) at the time of core damage. The RWST is injected prior to vessel melt-through. Containment heat and fission product removal are available. Relevant recovery and accident management action to be included in the PDS group would be 2.1.3.

Key Plant Damage State 4A represents sequences at intermediate RCS pressure (200 to 1800 psia) at the time of core damage. The RWST is injected after vessel meltthrough. Containment heat and fission product removal are available. Relevant recovery and accident management action to be included in the PDS group would be 2.1.3.

Key Plant Damage State 5D represents sequences at intermediate RCS pressure (200 to 1800 psia) at the time of core damage. The RWST is injected after vessel meltthrough. Containment heat and fission product removal capability are not available. Relevant recovery and accident management action to be included in the PDS group would be 2.1.3.

Key Plant Damage State 2A represents sequences that are at low pressure (<200 psia) at the time of core damage. The RWST is injected prior to vessel melt-through. The containment is isolated and not bypassed. Containment heat and fission product removal are available.

Key Plant Damage State 2D represents sequences that are at low pressure (<200 psia) at the time of core damage. The RWST is injected prior to vessel melt-through. The containment is isolated and not bypassed. **C**ontainment heat and fission product removal are not available.

Key Plant Damage State 7A represents sequences that are high pressure (> 1800 psia) at the time of core damage. The RWST is injected after vessel melt-through. The containment is isolated and not bypassed. Containment heat and fission product removal are available. Relevant recovery and accident management action to be included in the PDS group would be 2.1.3.

Key Plant Damage State 3L represents sequences at intermediate RCS pressure (200 to 1800 psia). The containment is bypassed (less than 3” equivalent inside diameter). The RWST is not injected to containment so containment heat and fission product removal are not available. Relevant recovery and accident management action to be included in the PDS group would be 2.1.3.

Key Plant Damage State 4C represents sequences at intermediate RCS pressure (200 to 1800 psia) at the time of core damage. The RWST is injected after vessel meltthrough. Containment heat removal is not available. Relevant recovery and accident management action to be included in the PDS group would be 2.1.3.

Key Plant Damage State 6J represents sequences at high RCS pressure (> 1800 psia). The RWST is not injected into the containment so heat and fission product are not available. The containment is not isolated; the isolation failure is large (greater than 3" equivalent inside diameter). Relevant recovery and accident management action to be included in the PDS group would be 2.1.3.

Key Plant Damage State lN represents sequence at low RCS pressure with a large containment bypass. The RWST is not injected to containment, there is no containment heat or fission product removal.

3.5 Definitions of the Plant Damage State Indicators for BWR Plants

3.5.1 NUREG-1150 Analysis for Peach Bottom

Sixteen indicators are used to identify a plant damage state. They address the following issues:

- Initiating event.
- Status of external electrical power supply.
- Availability of AC-power.
- Availability of DC power.
- Status of the safety valves in the reactor coolant system.
- Status of high pressure injection.
- Status of mechanical rod drive system.
- Pressure in the reactor vessel.
- Status of low pressure injection.
- Status of heat removal from the reactor vessel.
- Status of condensate system.
- Status of high pressure service water system.
- Status of containment spray system.
- Status of the containment venting system.
- Containment leakage.
- Location of eventual containment leakage.

Each of the sixteen indicators is discussed below.

- Initiating Event:

A	Large LOCA,
S1	Medium LOCA,
S2/3	Small/very small LOCA,
T	Transient,
TC	Transient without scram (ATWS),
IORV	Inadvertent open relief valve.

– Status of External Electrical Power Supply:

Seismic induced LOSP (not relevant for internal events analyses).
Internal event or random LOSP.
No LOSP.

– Availability of AC Power:

Internal event or random LOSP and loss of all diesel generators.
At least one diesel generator available.

– Availability of DC Power, given Station Blackout:

All DC power is failed.
At least one DC train is available.

– Status of the Safety Valves in the Reactor Coolant System:

At least one SRV sticks open.
No stuck open SRV.

– Status of High Pressure Injection:

Both HPCI and RCIC are initially failed.
Either HPCI or RCIC is initially working.

– Status of the Mechanical Control Rod Drive System:

CRD is failed.
CRD actuation failure.
CRD operable.

– Pressure in the Reactor Vessel:

High - ADS has failed.
High - operator does not depressurise after failure of ADS.
Low - Depressurisation by ADS or manual, or by LOCA, or transient with stuck open SRV.

– Status of Low Pressure Injection:

Both LPCI and LPCS have failed and can not be recovered.
Both LPCI and LPCS are currently not available but can be recovered.
One pump is running, but no injection due to high pressure in the reactor vessel.
Either LPCI or LPCS is working.

– Status of Heat Removal from the Core:

All RHR modes are failed.
All RHR modes are currently unavailable, but can be recovered.
One RHR mode is available.

- Status of Condensate System:

Condensate system is failed.
Condensate system is recoverable.
Condensate system is available but not injecting.
Condensate system is working (not possible given core damage states).

- Status of High Pressure Service Water System:

HPSW is failed.
HPSW is recoverable.
HPSW is available, manual line-up and actuation required.
HPSW is working (not possible given core damage states).

- Status of the Containment Spray System:

CSS<is failed.
CSS is recoverable.
CSS is available, manual line-up and actuation required.
CSS is working.

- Status of the Containment Venting System:

Containment is not vented.
Drywell is vented.
Drywell is vented in ATWS, but pressure is still high.
Wetwell is vented in ATWS, but pressure is still high.
Wetwell is vented.

- Containment Leakage:

No leakage exceeding technical specifications.
Leak occurs after accident.
Rupture occurs after accident.
Leak or isolation failure occurs before accident.
Rupture or large isolation failure occurs before accident.

- Location of Containment Leakage:

Containment intact.
Drywell leakage.
Drywell head leakage.
Wetwell leakage.

3.5.2 NUREG-1150 Analysis for Grand Gulf

Thirteen indicators are used to identify a plant damage state. They address the following issues:

- Initiating event.
- Status of external electrical power supply.
- Availability of AC-power.
- Availability of DC power.
- Status of injection to the core.
- Pressure in the reactor vessel.
- Status of heat removal from the reactor vessel.
- Status of suppression pool makeup.
- Status of the gas treatment system.
- Status of the containment venting system.
- Containment leakage.
- Status of hydrogen igniters.
- Timing of core damage.

The attributes of the damage state indicators are similar to those in the Peach Bottom analysis and are therefore not repeated here.

3.5.3 ERI/HSK Analysis for Mühleberg

Six indicators are used to identify a plant damage. The six indicators address the following issues:

- Initiating event.
- Status of the ADS system.
- Status of containment prior to vessel breach.
- Status of suppression pool.
- Status of pedestal area at vessel breach.
- Status of reactor building.

Each of the six indicators is discussed below.

- Initiating Event.

This indicator provides information on the pressure of the reactor coolant system, and its integrity at the time of vessel failure, as the expected RCS pressure is related to initiator. Three categories of the RCS integrity status are identified and related to the initiating events, as shown below:

T Transient.
TC Transient without scram.
A large LOCA.

Other initiating events do not show among the dominant plant damage states; therefore, they are not included in the attributes of this indicator.

- Status of the ADS System.

H Depressurisation unsuccessful.
L Depressurisation successful.

- Status of Containment prior to Vessel Breach.

I Containment intact/isolated.
F Containment isolation failed, containment unisolated.

- Status of Suppression Pool.

A Suppression pool active.
B Suppression pool bypassed.

- Status of Pedestal Area at Vessel Breach.

D Pedestal area dry.
W Pedestal area wet/flooded.

- Status of Reactor Building.

Y Reactor building isolated.
N Reactor building not isolated.

With the number of attributes possible for the six PDS indicators, there are potentially 96 different plant damage states. The 6 plant damage states with an upper 95 % quantile greater than 1 E-7/yr. are retained for containment event tree analysis.

3.5.4 Browns Ferry IPE Analysis

Four indicators are used to identify a plant damage state. The indicators address the following issues:

- Pressure in the reactor vessel at time of vessel breach ???(This information appears to be unavailable in the level 1/level 2 interface).
- Availability of water on the drywell floor at time of vessel failure???(This information appears to be unavailable in the level 1/level 2 interface).
- Status of containment integrity.
- Availability of water to cool core debris.
- Status of suppression pool cooling.
- Status of containment venting.
- Status of secondary containment isolation and integrity.
- Status of standby gas treatment system.

The attributes of the indicators are similar to those used in the NUREG-1150 analysis for Peach Bottom. They are not discussed further.

With the number of attributes possible for the six PDS indicators, there are potentially 384 different plant damage states.

3.5.5 Perry IPE Analysis

Ten indicators are used to identify a plant damage state. The indicators address the following issues:

- Initiating event.
- Power recovery after core damage.
- Pressure in the reactor vessel at the time of core damage.
- Time of RPV injection failure.
- Status of suppression pool cooling.
- Status of containment spray.
- Status of containment venting.
- Status of hydrogen igniters.
- Status of the annulus exhaust gas treatment system.

– Availability of water in the pedestal area.

The attributes of the indicators are similar to those used in the NUREG-1150 analysis for Grand Gulf. They are not discussed further.

With the number of possible attributes, 75 physically possible plant damage states are obtained.

3.6 Plant Damage State Analysis for Boiling Water Reactors

3.6.1 NUREG-1150 Analysis for Peach Bottom

Each of the 18 dominant core damage sequences with frequencies > E-10 is assigned by cut set to one or more plant damage states. Each plant damage state is defined by a 16 character vector with attributes according to section 3.5.1. The assignment process results in 20 distinct plant damage states. Eleven of the core damage sequences are assigned to exactly one plant damage state, while the remaining seven are assigned to several plant damage states, according to several significant cut sets.

3.6.2 NUREG-1150 Analysis for Grand Gulf

Each of the 50 core damage sequences with frequencies > E-10 is assigned by cut set to one or more plant damage states. Each plant damage state is defined by a 13 character vector with attributes according to section 3.3.6. The assignment process results in 12 distinct plant damage states. One of the core damage sequences is assigned to exactly one plant damage state, all others are assigned to several plant damage states, as there are several significant cut sets. On the other hand, four plant damage states contain 3 distinct core damage sequences, the others exactly one core damage sequence.

3.6.3 HSK/ERI Analysis for Mühleberg

All core damage sequences with frequencies above 1 E-7, and sequences involving ATWS, interfacing system LOCAs and steam line breaks outside containment are assigned to the appropriate plant damage

3.6.4 Browns Ferry IPE Analysis

All core damage sequences with frequencies above 1 E-8 are assigned to the appropriate plant damage states. This results in 20 distinct plant damage states with frequencies greater 1E-8.

3.6.5 Perry IPE Analysis

All core damage sequences with frequencies above 1 E-8 are assigned to the appropriate plant damage states. Figures 3.6.5-1 and 3.6.5-2 show examples of this assignment process. Depicted are the plant damage state event trees (which are expansions of the core damage event trees that incorporate containment functions) for the loss of offsite power and small LOCA events.

The assignment process results in 75 distinct plant damage states with frequencies greater 1E-8

3.7 Regrouping of Plant Damage States for Boiling Water Reactors

3.7.1 NUREG-1150 Analysis for Peach Bottom

The 20 individual plant damage states are assigned to nine plant damage state groups to facilitate the quantification of the containment event tree. The grouping is discussed below:

PDS Group 1 is composed of two accident sequences,

- a large LOCA initiator followed by immediate failure of the LPCS and LPCI systems (other high or low pressure systems can not mitigate this sequence in time or fail as a result of the initiator). The result is early core damage.

- a medium LOCA initiator followed by failure of all injection capability, resulting in early core damage. CRD is working in both sequences and all containment heat removal is working. Venting would work if needed, but is not be demanded before core damage.

Recovery and accident management action to be included in the PDS group is 2.2.6.

PDS Group 2 is composed of four sequences. It is similar to PDS group l. Different transient initiators with subsequent failure of SRVs result in the equivalent of an intermediate LOCA. The sequences then follow the same pattern as in PDS group l. Containment heat removal is working, but steam is directed through the SRVs to the suppression pool. This results in early core damage. Venting will not occur before core damage.

Recovery and accident management action to be included in the PDS group is 2.2.6.

PDS Group 3 is composed of two sequences. This group is similar to PDS group l. Transient initiators with subsequent failure of SRVs result in the equivalent of an intermediate LOCA. The sequences then follow the same pattern as in PDS group l. Containment heat removal is not working, but steam is directed through the SRVs to the suppression pool, CRD is not working in some cut sets, containment heat removal is not working and HPSW is failed by operator error or can not be used in time

Recovery and accident management action to be included in the PDS group is 2.2.4.

PDS group 4 is composed of two sequences:

- The first sequence is a station blackout, followed by one stuck open SRV. High pressure injection fails and early core damage results. Vessel pressure remains low; DC power has also failed.

- For the second sequence, there is no stuck open SRV, so the vessel is at high pressure. Venting is not possible unless AC is restored, AC systems are available with recovery of AC power.

Recovery and accident management actions to be included in the PDS group are 2.2.1, 2.2.5 and 2.2.6.

PDS Group 5 is composed of three sequences, involving a station blackout with or without one stuck open SRV and initially successful operation of HPCI or RCIC. Battery depletion may or may not occur before core damage. The vessel remains at low pressure if a SRV is stuck open, otherwise, it repressurizes on loss of DC. AC systems are available on recovery of AC power. Venting not possible until AC is restored.

Recovery and accident management actions to be included in the PDS group are 2.2.1, 2.2.2 and 2.2.6.

PDS Group 6 is composed of two sequences, an IORV or a loss of AC bus with failure to scram, SLC works, HPCI works initially, and the vessel is not manually depressurized, PCI fails on high suppression pool temperature. The containment is not vented before core damage, but venting is operable.

Recovery and accident management action to be included in the PDS group is 2.2.6.

PDS Group 7 is composed of one sequence, an IORV with failure to scram and SLC also failed, HPCI fails on high suppression pool temperature, the reactor is:

a) not manually depressurized, or

b) manually depressurized to use low pressure systems.

If a) then early core damage results and venting will not occur before core damage.

If b) then the containment will pressurise until venting, containment failure, or SRV reclosure on high containment pressure.

In all b) cases, the low pressure injection systems will fail due to low NPSH or harsh environments and core damage will result. Venting will be tried before core damage. The CRD system is working in all cases.

PDS Group 8 is composed of three sequences, loss of AC bus or PCS with failure to scram, and SLC also fails. HPCI fails on high suppression pool temperature, and the reactor is

a) not manually depressurized or

b) is manually depressurized to use the low pressure systems.

If a) then early core damage results and venting will not occur before core damage.

If b) then the containment will pressurise until either venting, containment failure, or SRV reclosure on high containment pressure. In all b) cases, the low pressure injection systems will fail due to low NPSH or harsh environments and core damage will result. Venting will be tried before core damage. The CRD system is working in all cases.

Recovery and accident management actions to be included in the PDS group are 2.2.1 and 2.2.6.

PDS Group 9 is composed of one sequence, a LOSP with failure to scram and SLC failure, HPCI fails on high suppression pool temperature and the reactor is:

a) not manually depressurized, or

b) is manually depressurized to use the low pressure systems.

If a) then early core damage results and venting will not occur before core damage.

If b) then the containment will pressurise until venting, containment failure, or SRV reclosure on high containment pressure.

In all b) cases, the low pressure injection systems fails due to low NPSH or harsh environments and core damage will result. Venting will be tried before core damage. The CRD system is working in all cases. Recovery and accident management action to be included in the PDS group is 2.2.6.

3.7.2 NUREG-1150 Analysis for Grand Gulf

The 12 distinct plant damage states resulting from core damage sequence assignment process are not further condensed. Instead, the analysis is simplified by restricting the attributes of the indicators to the attributes characterising the dominant events and states, as shown in table 3.7.2-1

Table 3.7.2-1 Plant Damage State Indicators for Dominant Events

Indicators for plant damage states 1 - 4	
B1	Station blackout transient has occurred. Offsite power is not recoverable because there is no emergency DC power.
B2	Station blackout transient has occurred. Offsite power is recoverable.
T2	Loss of PCS transient has occurred. Offsite or onsite power is available.
TC	ATWS has occurred. Offsite or onsite power is available.

Indicators for plant damage state 7	
Pl	The reactor vessel is at high pressure at the onset of core damage and depressurization is not possible.
P2	The reactor vessel is at high pressure at the onset of core damage because the operator failed to depressurize; depressurization is possible.
P3	The reactor vessel could be at high pressure at the damage.
P4	The reactor vessel is at low pressure.

Indicators for plant damage state 5	
Il	Injection to the reactor vessel is not available after the onset of core damage.
I2	Injection with the Firewater system is available before and after the onset of core damage.
I3	Injection with the condensate system is recoverable with the restoration of offsite power.
I4	Injection with the low pressure systems (LPCS and coolant injection) is recoverable with the restoration of offsite power.
I5	Injection with both the high and low pressure systems is recoverable with the restoration of offsite power.
I6	Injection with high pressure systems (RCIC and CRD) and low pressure systems (LPCS and coolant injection) is recoverable with the restoration of offsite power.

Indicators for plant damage state 6	
Hl	CS is not available at the onset of core damage, neither is it recoverable.
H2	At least one train of CS is recoverable with the restoration of offsite power.
H3	At least one train of CS is available at the onset of core damage.

Indicators for plant damage states 8 - 12	
Ml	Venting, standby gas treatment, containment isolation, hydrogen igniters) are not available at the onset of core damage.
M2	Venting, standby gas treatment, containment isolation, hydrogen igniters are recoverable with the restoration of offsite power.
M3	Venting, standby gas treatment, containment isolation, hydrogen igniters are available at the onset of core damage.

Indicators for timing of core damage	
ST	Core damage occurs in the short term (at -1 hour).
LT	Core damage occurs in the long term (>1 hour).

The 12 plant damage states are described below.

PDS 1 (B2-P3-I5-H2-M2-ST) involves station blackout scenarios where LOSP is recoverable. Coolant injection is lost early such that core damage occurs in the short term and with the vessel at high pressure because depressurization did not have an effect in the prevention of core damage. If offsite power is restored then the following functions are available: either high pressure injection or low pressure injection or both, heat removal via the sprays, and venting, standby gas treatment, containment isolation, hydrogen igniters.

This PDS also includes cut sets with either one or two stuck open relief valves. With the restoration of offsite power, the following coolant injection systems are recoverable: HPCS, condensate, LPCI and

LPCS. In some cases, HPCS and LPCS are recoverable, but only for around twelve hours, they are then lost on room heatup. The fire water system is available in every cut set. For those cut sets with two stuck open relief valves, the RCIC system is available.

Recovery and accident management action to be included in the PDS group is 2.2.6.

PDS 2 (B2-P3-I5-H1-M2-ST) involves station blackout scenarios where LOSP is recoverable Coolant injection is lost early such that core damage occurs in the short term and with the vessel at high pressure because depressurization did not have an effect in the prevention of core damage. If offsite power is restored then the following functions are available: either high pressure injection or low pressure injection or both, venting, standby gas treatment, containment isolation, hydrogen igniters. Heat removal via the sprays is not available with the recovery of offsite power.

This PDS also includes cut sets with either one or two stuck open relief valves. With the restoration of offsite power, the following coolant injection systems are recoverable: HPCS and condensate. In some cases, LPCS is recoverable, but only for around twelve hours, it is then lost on room heatup. The fire water system is available in every cut set. For those cut sets with two stuck open relief valves, the RCIC system is available.

Recovery and accident management actions to be included in the PDS group are 2.2.2, 2.2.4 and 2.2.6.

PDS 3 (B2-P3-I3-H1-M2-ST) involves station blackout scenarios where LOSP is recoverable. Coolant injection is lost early such that core damage occurs in the short term and with the vessel at high pressure because depressurization did not have an effect in the prevention of core damage.

If offsite power is restored then the following functions are available: low pressure injection only with condensate and venting, standby gas treatment, containment isolation, hydrogen igniters. Heat removal via the sprays is not available with the restoration of offsite power.

The PDS also includes cut sets with either one or two stuck open relief valves. With the restoration of offsite power, the following coolant injection system is recoverable: condensate. HPCS and LPCS, but only for around twelve hours, they are then lost on room heatup. The fire water system is available in every cut set. For those cut sets with two stuck
the RCIC is available.

Recovery and accident management actions to be included in the PDS group are 2.2.2, 2.2.4 and 2.2.6.

PDS 4 (B2-P4-I5-H2-M2-LT) involves station blackout scenarios where LOSP is recoverable. Coolant injection is lost late such that core damage occurs in the long term and with the vessel at low pressure. If offsite power is restored then the following functions are available: either high pressure injection or low pressure injection or both, heat removal via the sprays, and venting, standby gas treatment, containment isolation, hydrogen igniters.

With the restoration of offsite power, the following coolant injection systems are recoverable: HPCS, condensate, LPCI and LPCS. In some cases, HPCS and LPCS are recoverable, but only for around twelve hours, they are then lost on room heatup. The fire water system is available in every cut set.

Recovery and accident management actions to be included in the PDS group are 2.2.2, 2.2.4 and 2.2.6.

PDS 5 (B2-P4-I5-H1-M2-LT) involves station blackout scenarios where LOSP is recoverable. Coolant injection is lost late such that core damage occurs in the long term and with the vessel at low pressure. If offsite power is restored then the following functions are available: either high pressure injection or low pressure injection or both, and venting, standby gas treatment, containment isolation, hydrogen igniters. Heat removal via the sprays is not available with the restoration of offsite power.

There are some cut sets where RHR sprays are available with offsite power restoration, but these have negligible contributions and therefore were not separated out.

Recovery and accident management actions to be included in the PDS group are 2.2.2, 2.2.4 and 2.2.6.

PDS 6 (B2-P4-I2-H1-M2-LT) involves station blackout scenarios where LOSP is recoverable. Coolant injection is lost late such that core damage occurs in the long term and with the vessel at low pressure, fire water is recoverable. If offsite power is restored then venting, standby gas treatment, containment isolation, hydrogen igniters are available. Heat removal via the sprays is not available with the restoration of offsite power.

HPCS is available with the restoration of offsite power, but only for around twelve hours, it is then lost on room heatup.

Recovery and accident management actions to be included in the PDS group are 2.2.2, 2.2.4 and 2.2.6.

PDS 7 (B1-P1-I1-H1-M1-ST) involves station blackout (without any DC power) scenarios where LOSP is not recoverable. Coolant injection is lost early such that core damage occurs in the short term and with the vessel at high pressure and depressurization is not possible. Since offsite power is not recoverable, injection, heat removal, venting, standby gas treatment, containment isolation, hydrogen igniters are not available.

This PDS also includes cut sets with either one or two stuck open relief valves.

Recovery and accident management actions to be included in the PDS group are 2.2.2, 2.2.4 and 2.2.6.

PDS 8 (B1-P1-I1-H1-M1-LT) involves station blackout (without any DC power) scenarios where LOSP is not recoverable. Coolant injection is lost late such that core damage occurs in the long term and with the vessel at high pressure and depressurization is not possible. Since offsite power is not recoverable, injection, heat removal, venting, standby gas treatment, containment isolation, hydrogen igniters are not available.

Recovery and accident management action to be included in the PDS group is 2.2.2.

PDS 9 (TC-P2-I6-H3-M3-ST) involves ATWS transient scenarios. Coolant injection is lost early such that core damage occurs in the short term and with the vessel at high pressure because the operator failed to depressurize. High pressure injection with RCIC is available. Heat removal via the sprays is available and venting, standby gas treatment, containment isolation, hydrogen igniters are available.

Recovery and accident management actions to be included in the PDS group are 2.2.4 and 2.2.6.

PDS 10 (TC-P2-I4-H3-M3-ST) involves ATWS transient scenarios. Coolant injection is lost late such that core damage occurs in the long term and with the vessel at high pressure because the operator failed to depressurize. Low pressure injection is recoverable with reactor depressurization. Heat removal via the sprays is available and venting, standby gas treatment, containment isolation, hydrogen igniters are available.

Recovery and accident management actions to be included in the PDS group are 2.2.4, 2.2.5 and 2.2.6.

PDS 11 (T2-P2-I5-H3-M3-ST) involves transient scenarios where the PCS is lost. Coolant injection is lost early such that core damage occurs in the short term and with the vessel at high pressure because the operator failed to depressurize. Both high pressure and low pressure are recoverable since the failures involved operator failures. Heat removal via the sprays is available and venting, standby gas treatment, containment isolation, hydrogen igniters are available.

Recovery and accident management actions to be included in the PDS group are 2.2.4 and 2.2.6.

PDS 12 (T2-P2-I5-H3-M3-LT) involves transient scenarios where the PCS is lost. Coolant injection is lost late such that core damage occurs in the long term and with the vessel at high pressure because the operator failed to depressurize. Both high pressure and low pressure injection are recoverable since the failures involved operator failures. Heat removal via the sprays is available and venting, standby gas treatment, containment isolation, hydrogen igniters are available.

Recovery and accident management actions to be included in the PDS group are 2.2.4 and 2.2.6.

3.7.3 ERI/HSK Analysis for Mühleberg

The six plant damage states resulting from the core damage sequences to plant damage states assignment process are not further condensed. The six plant damage states are described below.

PDS 1, TCLFB DY -This represents ATWS sequences with core damage at low pressure, failed containment isolation, suppression pool bypass. The pedestal area is dry and the reactor building is isolated.

PDS 2, V. This PDS includes breaks outside containment. Since these accidents result in containment bypass, and therefore in containment failure at the time of core melt, this PDS does not explicitly require an event tree analysis.

PDS 3, TLIADN. This PDS includes the analysis for a damage state, which corresponds to aircraft impacts destroying both the primary and secondary containments.

PDS 4, TLIADY. This PDS consists of station blackouts with ADS operation. It is the most dominant PDS, with a frequency exceeding E-5 per reactor year, mostly due to seismically initiated station blackouts. For completeness, this PDS also groups the other low pressure transients, which have much smaller frequencies. Short term station blackouts are also included in this PDS.

Recovery and accident management action to be included in the PDS group would be 2.2.2.

PDS 5, THIADY. This PDS consists of **s**tation blackouts without ADS operation. It includes accidents involving common cause failure of all batteries. The reactor coolant system remains at high pressure during the core melt progression.

Recovery and accident management action to be included in the PDS group would 2.2.2.

PDS 6, ALIAWY. This PDS includes both large and small LOCAs. The contribution from small LOCAs to this damage state is less than 10%.

3.7.4 Browns Ferry IPE Analysis

The 20 plant damage states with frequencies above 1 E-8 are condensed to nine plant damage state groups (called key plant damage states in the Browns Ferry IPE. They are described below.

PDS Group 1 consists of LOSP sequences with HPCI/RCIC failure, low RCS pressure at time of onset of core damage, but repressurization before vessel breach, isolated containment, dry pedestal area, no containment heat removal.

Recovery and accident management actions to be included in the PDS group would be 2.2.2, 2.2.4 and 2.2.6.

PDS Group 2 consists of IOSRV sequences with loss of LPCI, low RCS pressure at time of vessel breach, isolated containment, wet pedestal area, containment heat removal.

PDS Group 3 consists of sequences with failure of all high pressure injection, high RCS pressure at time of vessel breach, isolated containment, wet pedestal area, containment heat removal.

PDS Group 4 consists of transients with failure of DC power, failing all injection due actuation faults, stuck open SRV, low RCS pressure at time of vessel breach, isolated containment, wet pedestal area, containment heat removal.

Recovery and accident management actions to be included in the PDS group would be 2.2.2 and 2.2.4.

PDS Group 5 consists of LOSP sequences with failure of all high pressure injection, high RCS pressure at time of vessel breach, isolated containment, dry pedestal area, no containment heat removal.

Recovery and accident management actions to be included in the PDS group would be 2.2.2 2.2.5 and 2.2.6.

PDS Group 6 consists of various sequences with long term failure of high pressure injection, high RCS pressure at time of vessel breach, late containment failure, dry pedestal area, failed containment heat removal.

Recovery and accident management actions to be included in the PDS group would be 2.2.5 and 2.2.6.

PDS Group 7 consists of ATWS sequences with failure of the operator to start SLC, high RCS pressure at time of vessel breach, early containment failure, wet pedestal area.

PDS Group 8 consists of sequences with failure of turbine trip and of MSIVs to close, and with unisolated breaks outside containment, low RCS pressure at time of vessel breach, bypassed containment, wet pedestal area.

PDS Group 9 consists of interfacing systems LOCAs, with high pressure at time of vessel breach, wet drywell, bypassed containment.

3.7.5 Perry IPE Analysis

The 75 plant damage states with frequencies above 1 E-8 are condensed to 15 plant damage state groups. The four most important groups are discussed below.

PDS Group 1, represents non-SBO sequences with the containment intact at core damage and successful containment heat removal with the RHR spray loop, late injection failure, and RPV depressurization. This group has 7 dominant sequences above 1% of the core damage frequency. The most dominant sequence is an ATWS (successfully shutdown with the standby liquid control) loss of power conversion system, failure of the motor feed pump, and failure to inhibit ADS. This group also includes four other (shutdown) ATWS with loss of PCS sequences.

PDS Group 2 represents non-SBO sequences with containment intact at core damage and successful containment heat removal with the venting system, late injection failure, and RPV depressurization. This group has 6 dominant sequences above 1% of the core damage frequency. The most dominant sequence is a flooding scenario with failure of all injection.

PDS Group 3 represents sequences other than critical. ATWS or LOOP & SBO with containment failed at core damage, unsuccessful late injection, and successful RPV depressurization.

Recovery and accident management actions to be included in the PDS group are 2.2.4 and 2.2.5.

PDS Group 4 represents SBO sequences with failure of all injection.

Recovery and accident management actions to be included in the PDS group are 2.2.2 and 2.2.4.

4. IMPACT OF THE PLANT DAMAGE STATE INDICATORS AND THEIR ATTRIBUTES ON THE TREATMENT OF SEVERE ACCIDENT MANAGEMENT IN THE CONTAINMENT EVENT TREES

Severe accident management actions can significantly influence accident progression in the reactor system and containment following a core damage accident. In many cases, such actions depend on the success or failure of recovery and preventive accident management actions which are object of the examinations in the level 1 analyses. For an adequate treatment of the severe accident management actions it is important that all relevant information is available to the level 2 analysis.

4.1 NUREG-1150 Analysis for Surry and Sequoyah

As explained in section 3.1 and depicted in Figures 3.3.1-1 - 3.3.1-4, all information on the level 1 event tree split fractions needed for the quantification of severe accident management actions in the containment event trees is passed on to the level 2 analysis. In the analysis for Surry, for example, such information is input to the quantification process for 31 of the 71 questions (branch points) of the containment event tree. For 11 questions, the input is directly provided by the attributes of the plant damage states, and for 20 others, the input contains information inherent in the plant damage state attributes.

4.2 ERI/HSK Analysis for Beznau

In the Beznau analysis, fault tree linking provides the coupling of the information on the level 1 event tree split fractions to the containment event tree nodal questions. For 11 of the 31 containment event tree questions, the input is directly provided by the attributes of the plant damage states, and for three others it contains information inherent in the plant damage state attributes.

4.3 Maine Yankee IPE Analysis

Information on the level 1 event tree split fractions can be partly lost in the condensation process that generates the plant damage state groups. Input to the containment event tree analysis is one sequence for each key plant damage state. The selected sequence is considered to be representative of the dominant sequences included in the corresponding key damage state. Thus, only information related to the selected sequence can be conveyed to the containment event tree analysis, whereas information related to the other sequences is deleted. This process can produce reasonable results, if a plant damage state group contains only similar sequences, or if a plant damage state group contains one dominant sequences, while the other sequences are insignificant. This the case for the highest ranking plant damage state group 4A, in which the dominant sequence accounts for more than 90% of the sequences in the group. However, other plant damage state groups, for example 5A, contain dissimilar sequences with dissimilar split fractions, but with similar frequencies. Thus, information that can be important to the quantification of branch point probabilities, including consideration of severe accident management, may be lost when one ”representative” sequence is selected.

4.4 NUREG-1150 Analysis for Peach Bottom and Grand Gulf, and IPE Analysis for Perry

As explained in section 3.1, all information on the level 1 event tree split fractions needed for the quantification of severe accident management actions in the containment event trees is passed on to the level 2 analysis. In the analysis for Peach Bottom, such information is input to the quantification process for 61 of the 145 questions (branch points) of the containment event tree. For 13 questions, the input is directly provided by the attributes of the plant damage states, and for 48 others, the input contains information inherent in the plant damage state attributes.

In the analysis for Grand Gulf, information on the level 1 event tree split fractions is input to the quantification process for 36 of the 125 questions (branch points) of the containment event tree. For 15 questions, the input is directly provided by the attributes of the plant damage states, and for 21 others, the input contains information inherent in the plant damage state attributes.

In the IPE analysis for Perry, information on the level 1 event tree split fractions is input to the quantification process for 21 of the 68 questions (branch points) of the containment event tree. For 11 questions, the input is directly provided by the attributes of the plant damage states, and for 10 others, the input contains information inherent in the plant damage state attributes

4.5 ERI/HSK Analysis for Mühleberg and IPE Analysis for Browns Ferry.

Information on the level 1 event tree split fractions can be partly lost in the condensation process that generates the plant damage state groups. Input to the containment event tree analysis is one sequence for each key plant damage state. The selected sequence is considered to be representative of the dominant sequences included in the corresponding key damage state. Thus, only information related to the selected sequence can be conveyed to the containment event tree analysis. whereas information related to the other sequences is deleted. This process can produce reasonable results, if a plant damage state group contains only similar sequences, or if a plant damage state group contains one dominant sequences, while the other sequences are insignificant. This the case for the high ranking plant damage state groups in both analyses. Thus, it appears that no information important to the quantification of branch point probabilities, including consideration of severe accident management, is lost with the schemes for selecting one ”representative” sequence.

4.6 Concluding Remark

Considering the readily available computational tools for performing level 2 analyses, the economy of manually defining the condensed plant damage groups and selecting representative sequences, as was done in the Maine Yankee and Browns Ferry IPEs, must be doubted. It appears to be more economical to adopt approaches as in the NUREG-1150 studies in the ERI/HSK analysis for Beznau, which pass on all relevant information on split fractions of the level 1 event trees to the containment event tree analysis. In addition, this approach avoids the risk of loosing important information in the condensation process.

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FIGURES

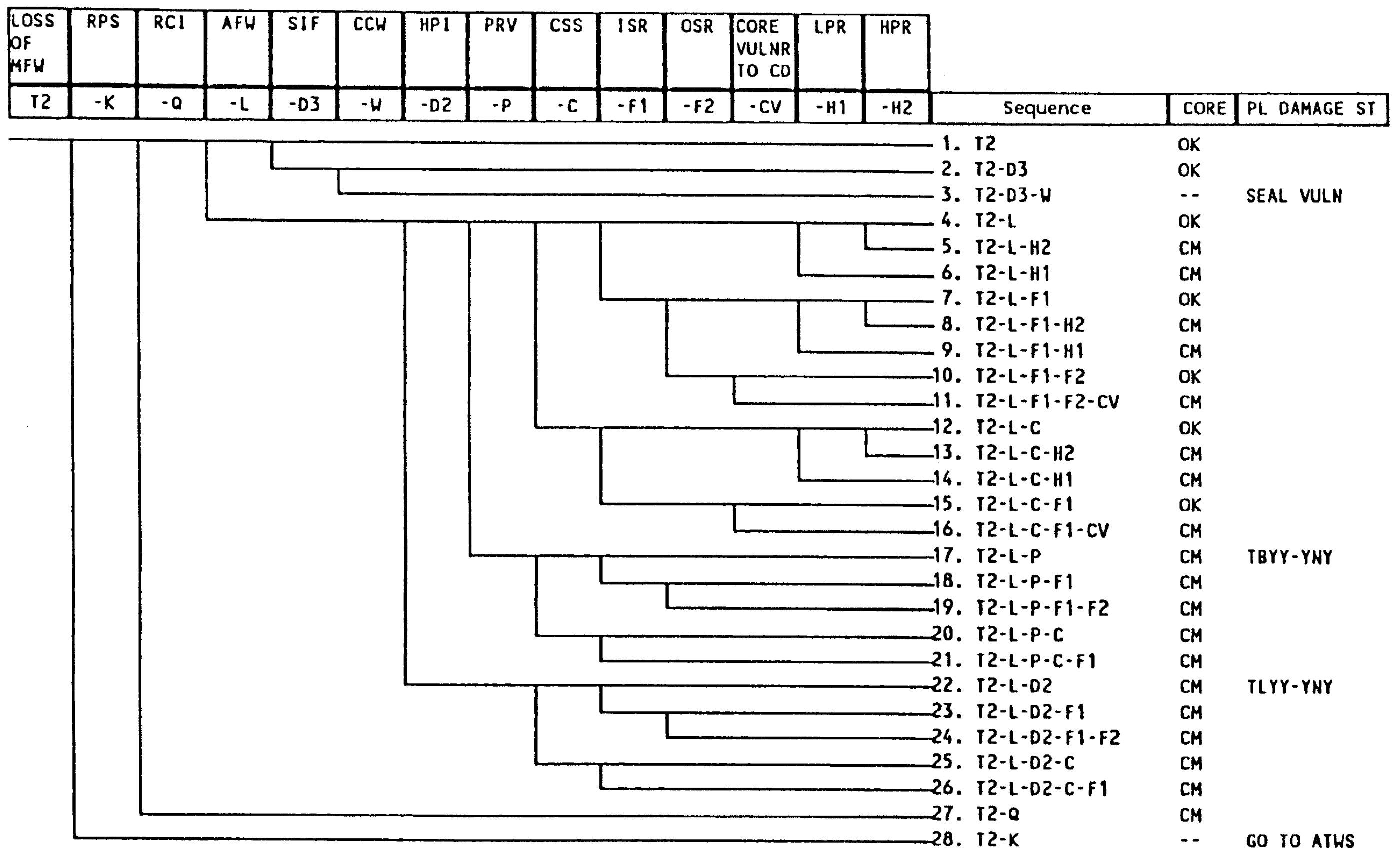

Figure 3.3.1-1 Plant Damage State Tree for Loss of Main Feedwater, Surry NUREG-1150

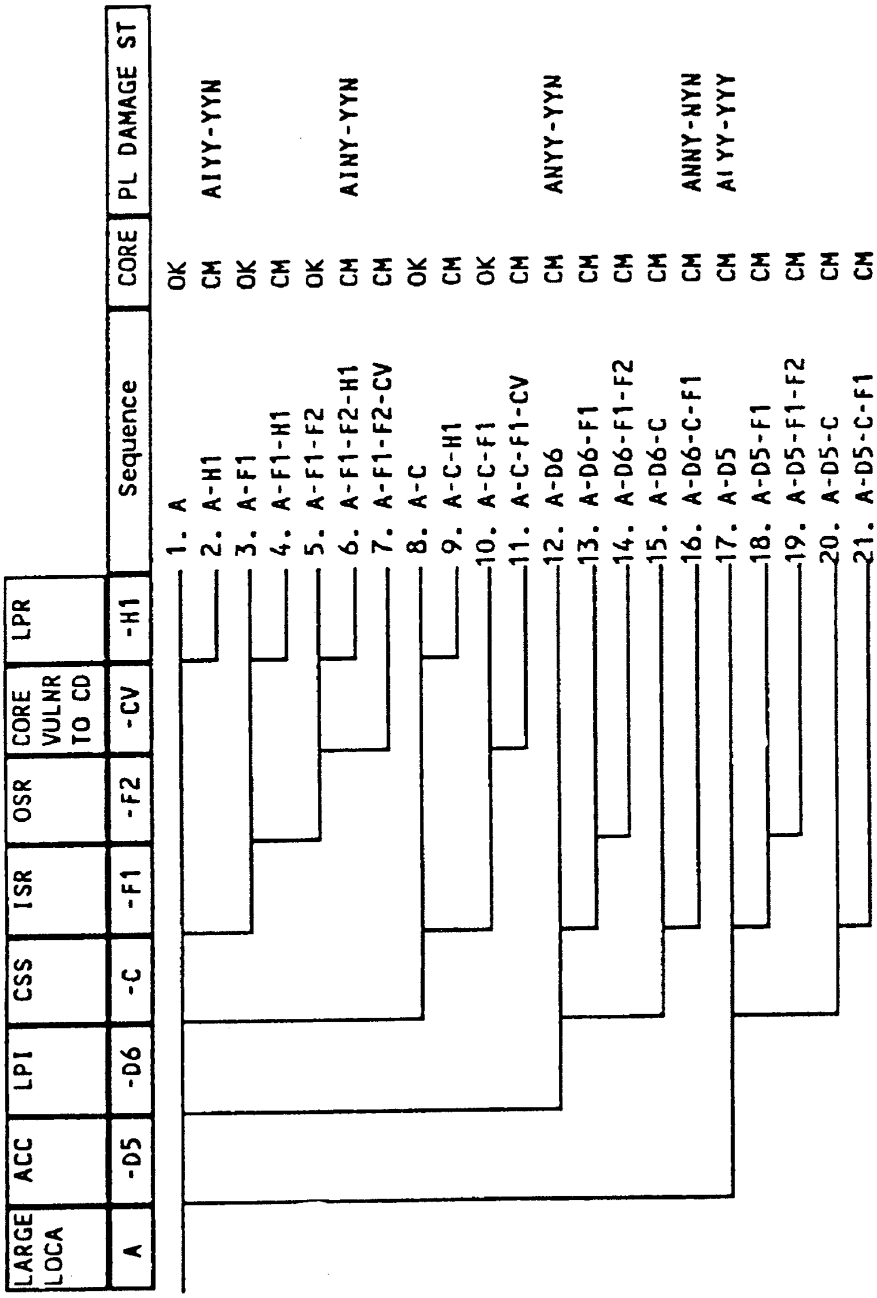

Figure 3.3.1-2 Plant Damage State Tree for Large LOCA, Surry NUREG-1150

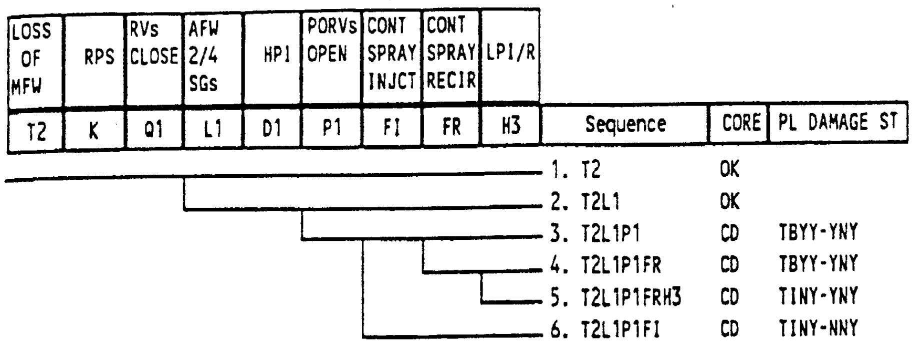

Figure 3.3.1 - 3. Plant Damage State Tree for Loss of Main Feedwater , Sequoyah

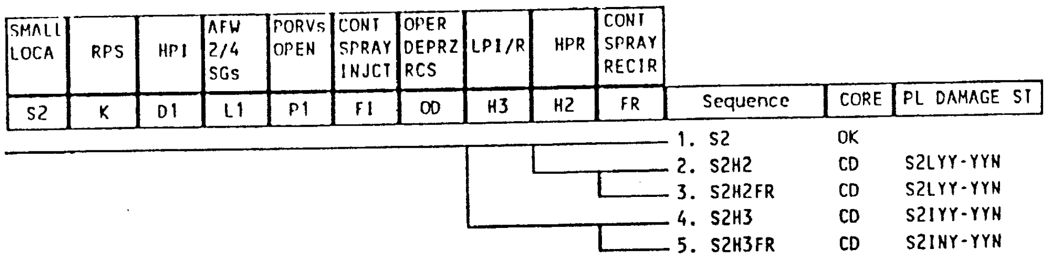

Figure 3.3.1 - 4. Plant Damage State Tree for Small LOCA , Sequoyah

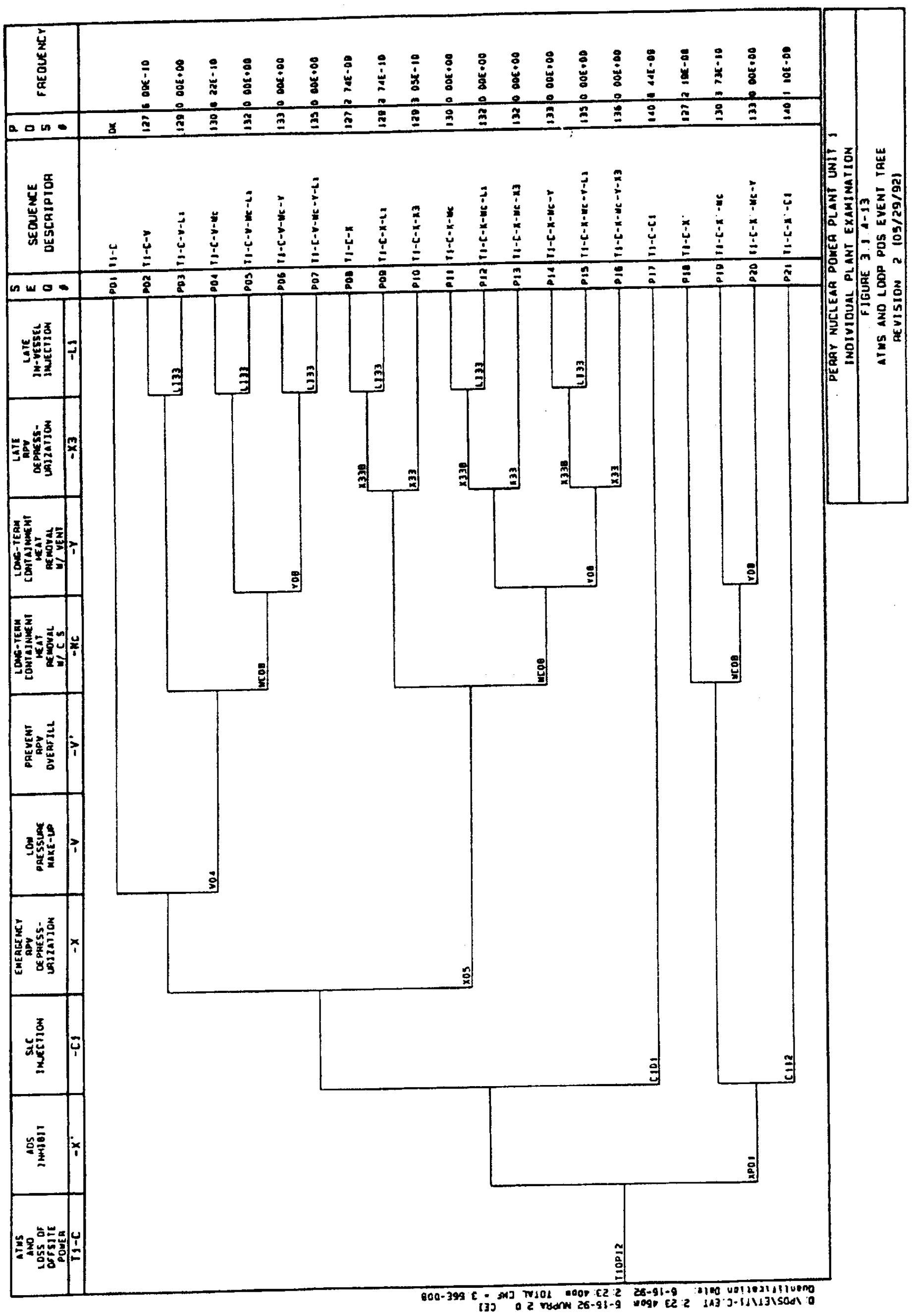

Figure 3.6.5-1 Plant Damage State Tree for Loss of Offsite Power (and ATWS), Perry IPE

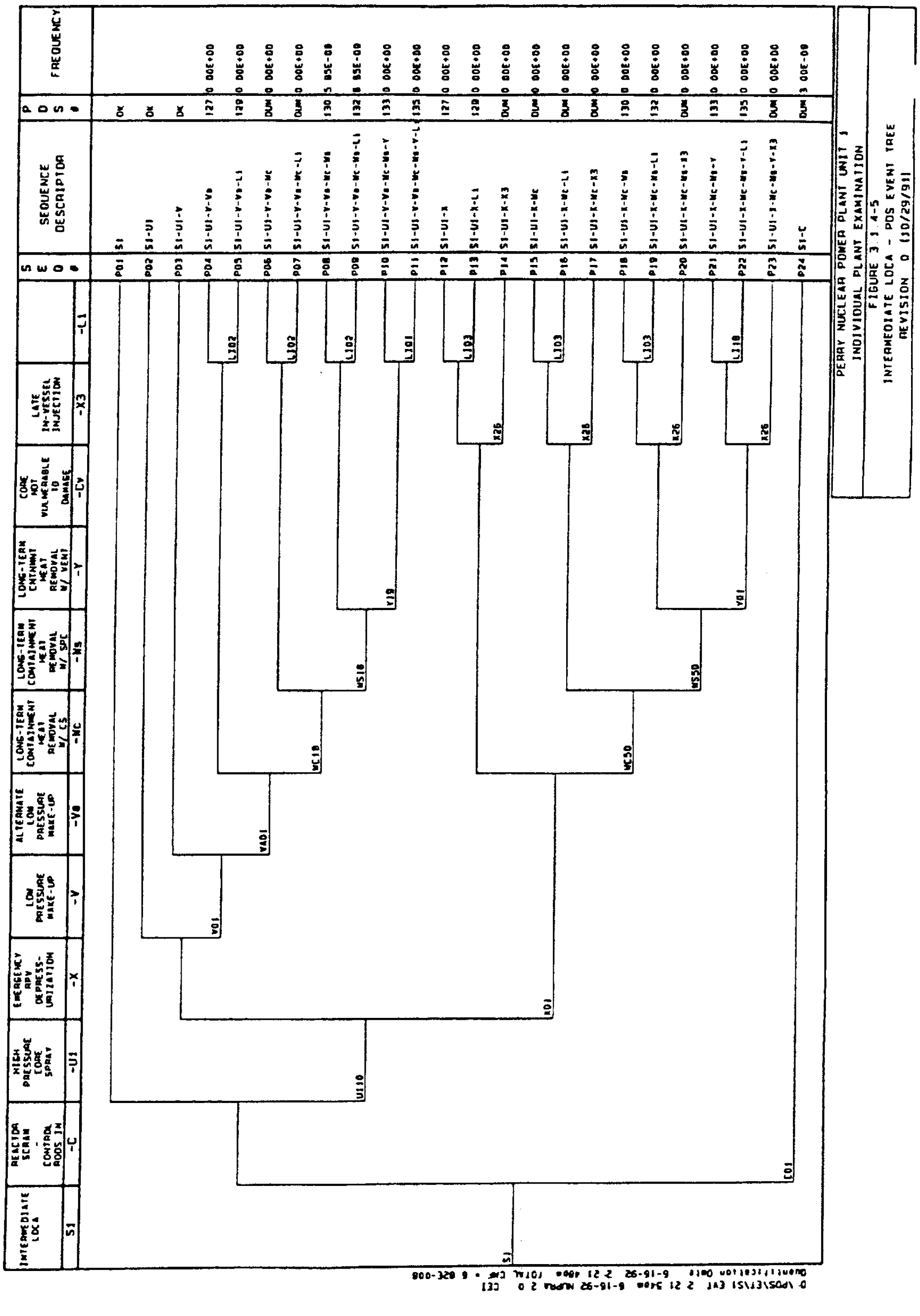

Figure 3.6.5-2 Plant Damage State Tree for Intermediate LOCA, Perry IPE