BWR OWNERS' GROUP

K. S. Putnam, Chairman
Tel: (319) 851-7238
Fax: (319) 851-7364
Ken.putnam@nmcco.com

Nuclear Management Company • Duane Arnold Energy Center • 3277 DAEC Road • Palo, IA, 52324

NRC Project 691

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SUBJECT: **TRANSMITTAL FOR REVIEW AND APPROVAL OF LICENSING TOPICAL REPORT, NEDO 33148, "SEPARATION OF LOSS OF OFFSITE POWER FROM LARGE BREAK LOCA," UNDER 10 CFR 170**

Reference: Letter from Kenneth Putnam to Jesse L. Funches, REQUEST FOR FEE WAIVER FOR NRC STAFF REVIEW OF LICENSING TOPICAL REPORT, NEDO 33148, "SEPARATION OF LOSS OF OFFSITE POWER FROM LARGE BREAK LOCA," UNDER 10 CFR 170, DATES APRIL 6, 2004.

The subject report is being transmitted to you for your review and approval. This report is being provided as the basis for an exemption request for BWRs to the requirements to consider loss of offsite power coincident with a large break LOCA. This is one of the key industry and NRC efforts relative to risk informing 10 CFR 50 via Option 3. The NRC Staff and the BWROG have held several meetings during 2002 and 2003 on this subject and we are enthusiastic to have this opportunity to provide a first of a kind regulatory approach to Option 3.

The goal of this exemption request is to achieve a more risk-balanced plant design and thereby, improve overall plant safety, with the elimination of this unnecessary and burdensome regulatory requirement.

Reference 1 requested a fee waiver for your review of the report. Since a timely NRC Staff review is needed to support possible rulemaking to risk informing 10 CFR 50 via Option 3, it is requested that your review begin immediately while the fee waiver request is being deliberated.

The BWROG would appreciate meeting with the NRC staff at your earliest opportunity. Please contact either Rick Hill (GE) (408) 925-5388 or Tony Browning (NMC) (319) 851-7750 regarding these arrangements

Very truly yours,

Kenneth S. Putnam

Kenneth S. Putnam
BWR Owners' Group Chairman

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cc: BWROG Executive Committee
BWROG Primary Representatives
Risk Informed Part 50 Option 3 Committee
J. E. Conen, BWROG Vice Chairman
T. G. Hurst, GE
R. A. Hill, GE
J. Butler, NEI
B. Pham, NRC

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LICENSING TOPICAL REPORT

SEPARATION OF LOSS OF OFFSITE POWER FROM LARGE BREAK LOCA

R. M. Wachowiak
R. Engel

Approved by: R A Hill
R. A. Hill, Project Manager
General Electric Company

Prepared by GE Nuclear Energy for the BWR Owners' Group
Risk Informed Part 50 Option 3 Committee

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LIST OF ACRONYMS

Term	Definition
AC	Alternating Current
AOO	Anticipated Operational Occurrence
ARI	Alternate Rod Insertion
ATWS	Anticipated Transient without Scram
B&W	Babcock and Wilcox
BOP	Balance of Plant
BWR	Boiling Water Reactor
BWROG	BWR Owners' Group
CCDP	Conditional Core Damage Probability
CCR	Combined Change Request
CDF	Core Damage Frequency
CFR	Code Of Federal Regulations
CPR	Critical Power Ratio
CRD	Control Rod Drive
CSS	Containment Spray System
CST	Condensate Storage Tank
DBA	Design Basis Accident
DC	Direct Current
DG	Diesel Generator
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPG	Emergency Procedure Guidelines
EPU	Extended Power Uprate
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GE	General Electric Company

Term	Definition
HPCI	High Pressure Coolant Injection System
HPCS	High Pressure Core Spray
IORV	Inadvertent Open Relief Valve
IRIR	Integrated Risk Informed Regulation
ISLOCA	Interfacing Systems LOCA
LBLOCA	Large-Break Loss Of Coolant Accident
LBLOCA/LOOP	LBLOCA with a Concurrent Loss Of Offsite Power
LERF	Large Early Release Frequency
LHGR	Linear Heat Generation Rate
LOCA	Loss Of Coolant Accident
LOOP	Loss Of Offsite Power
LOSW	Loss of Service Water
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LTR	Licensing Topical Report
MBLOCA	Medium Break LOCA
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
NRC	Nuclear Regulatory Commission
OLTP	Original Licensed Thermal Power
PCS	Power Conversion System
PCT	Peak Cladding Temperature
PRA	Probabilistic Risk Assessment
PSS	Pressure Suppression System
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling System
RCS	Reactor Coolant System

Term	Definition
RG	Regulatory Guide
RHR	Residual Heat Removal
RITS	Risk Informed Technical Specification
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SBLOCA	Small Break LOCA
SBO	Station Blackout
SLC	Standby Liquid Control
SORV	Stuck Open Relief Valve
SP	Suppression Pool
SPC	Suppression Pool Cooling
SRM	(NRC) Staff Requirements Memorandum
SRV	Safety Relief Valve
TBCCW	Turbine Building Closed Cooling Water
TMI	Three Mile Island
UHS	Ultimate Heat Sink

Executive Summary

Beginning with the Reactor Safety Study (WASH-1400) in 1975, the Nuclear Regulatory Commission (NRC) has endeavored to incorporate risk insights into the regulatory process. Today, risk insights, using Probabilistic Risk Assessment (PRA) models of power plants, form the cornerstone of NRC's Reactor Oversight Process and govern the routine operation of power plants, via the Maintenance Rule (10 CFR 50.65). In SECY-98-300 (Reference 1), the Staff proposed a framework for changing the current regulatory requirements using similar risk-informed processes. This Licensing Topical Report (LTR) is the culmination of extensive dialogue between the NRC and the Boiling Water Reactor Owners' Group (BWROG) to apply that risk-informed process to the current regulatory requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix A – General Design Criterion 35. Specifically, this LTR justifies removing the current requirement to postulate a loss-of-offsite power (LOOP) concurrent with the design basis accident (DBA) large break loss-of-coolant accident (LOCA).

Pending rulemaking to accomplish this goal, the Staff has stated that individual licensees may submit requests for exemption, pursuant to 10 CFR 50.12, to these current requirements. The purpose of this LTR is to provide adequate technical and regulatory justification for the NRC to approve such an exemption to the regulations for a requesting BWR licensee, without having to repeat the required justification in a plant specific submittal. To accomplish this, the BWROG has:

- Identified the specific regulations for which the exemption is applicable and the basis for granting the exemption. It is understood that not all the regulations which contain the coincident LOOP requirement are covered by this exemption.
- Identified the specific design changes facilitated by the exemption. These changes are defined at a conceptual design level with some detailed design information provided where needed for clarity. It is anticipated that individual licensees will limit the scope of the exemption request to those changes specified in this report. However, an individual licensee may request additional changes on a plant specific basis as long as adequate justification is provided.
- Completed a risk analysis that demonstrates: 1) the probability of a large break LOCA in combination with a LOOP is consistent with the Regulatory Guide 1.174 (Reference 4) threshold for such regulatory changes (i.e., less than $1x10^{-6}$); and, 2) there is an overall risk balance for BWRs by implementing the changes. It is recognized that the analysis in this report is generic to various BWR product lines; however, key parameters are identified so a licensee can demonstrate conformance with this generic analysis.
- Completed a "defense-in-depth" evaluation, as well as a demonstration of the "safety margins" following implementation of the changes identified in this report, which show that there is reasonable certainty that no gross fuel failure will occur and that the reactor

coolant pressure boundary, as well as the containment barriers, remain intact, even if the assumed LOOP/LBLOCA were to occur. It is not expected that individual licensees would need to perform this evaluation on a plant specific basis in their exemption requests.

The goal of this exemption request is to achieve a more risk-balanced plant design and improve overall plant safety by the elimination of this unnecessary and burdensome regulatory requirement.

1.0 Introduction

In Commission Paper SECY-98-300, "Options for Risk-Informed Revisions to 10 CFR Part 50 – 'Domestic Licensing of Production and Utilization Facilities'," dated December 23, 1998 (Reference 1), the NRC Staff recommended that consideration be given to changing some of the individual requirements of 10 CFR Part 50 based upon the insights that had been gained from Probabilistic Risk Assessments (PRA) of nuclear power plants. This proposal was referred to as "Option 3" in SECY-98-300. The pilot application chosen for Option 3 was 10 CFR 50.46, the specific requirements for Emergency Core Cooling Systems (ECCS).

In its Staff Requirements Memorandum (SRM) dated March 31, 2003 (Reference 2), the NRC Commissioners directed the Staff to proceed with rulemaking, under the Option 3 framework, to risk-inform the ECCS functional reliability requirements in 10 CFR Part 50, Appendix A - General Design Criterion 35, by revising the current requirements and removing the consideration of a large break loss-of-coolant accident (LBLOCA) coincident with a loss-of-offsite power (LOOP).

The NRC staff was directed, as allowed by 10 CFR 50.12, to consider licensees request for exemptions to these requirements. The Staff's review of such exemption(s) will facilitate the rulemaking process, similar to the approach utilized in development of the pending rule for Special Treatments, 10 CFR 50.69, referred to as Option 2. The BWROG has met with the Staff on many occasions to discuss the technical approach that would form the basis for such an exemption. The result of these meetings was the development of this Licensing Topical Report (LTR).

This LTR provides the technical and regulatory basis for an exemption request and licensee guidance for implementation. It should be noted that this LTR makes use of other published reports, both NRC and Industry, for its justification of several important assumptions made in this study, e.g., LBLOCA probability, consequential/delayed LOOP, and "double sequencing" of electrical loads.

It is intended that this LTR can be referenced by any BWR licensee in a specific 10 CFR 50.12 exemption request without the need to perform extensive plant specific analyses, as outlined in the implementation guidelines provided in Section 9 of this report. This approach should help streamline the Staff's review of the individual submittals and provide the supporting framework for the proposed rulemaking outlined in Reference 2.

2.0 Technical Approach

2.1 Background

From Reference 3 (Section 3.1.2, "ECCS-Related Design Basis Changes"):

For applications that would seek to change the design basis for the ECCS (e.g., to remove an accident from the ECCS design basis analyses), the resulting change in CDF and LERF must also meet the criteria from the framework for risk-informing 10 CFR Part 50 and Regulatory Guide 1.174 as described in Section 3.1.1. The resulting changes in CDF and LERF are determined by assuming that the plant can no longer respond to this particular accident (i.e., the subject accident is assumed to lead directly to core damage). As an example, a large LOCA with a coincident LOOP could be removed from the design basis if the frequency of that combination of events from all possible contributors (random pipe breaks from all known mechanisms, seismic events, heavy load drops, etc.) were to be assumed to lead directly to core damage, but still meet the framework and RG 1.174 criteria. This could allow, for example, an increase in the diesel generator start time. The full extent of plant changes to be allowed, with respect to both design basis changes and design or operational changes, would need to be established by the staff.

Regulatory Guide 1.174, Section 2 (Reference 4) provides the key principles of risk-informed decision-making when making changes to the licensing basis:

- The proposed change meets current regulations unless it is explicitly related to a requested exemption or rule change.
- The proposed change is consistent with defense-in-depth philosophy.
- The proposed change maintains sufficient safety margins.
- When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
- The impact of the proposed change should be monitored using performance measurement strategies.

2.2 BWROG Approach

With the publication of Reference 3, as highlighted above, the BWROG began developing its approach for evaluating the feasibility of implementing this change using the RG 1.174 template. This approach was developed in stages over the last two years. Public meetings were held with the Staff to discuss the conceptual approach as it was developed and to define the key points for detailed evaluation in the proposed LTR. As noted earlier, the LTR serves a dual purpose. One is to assist licensees in requesting exemption from the existing regulatory requirements, pursuant to 10 CFR 50.12; and, the other is to offer insights to assist the Staff in their preparation of rulemaking to revise the existing regulations, as requested by Reference 2.

The cornerstone of the approach is the recognition that the LBLOCA/LOOP combination is not completely removed from the licensing basis with the proposed change in regulation, but is redefined from a design basis event to a severe accident, for which some mitigation capability must be assured. Also, it must be acknowledged that continued compliance with 10 CFR 50.46 (and all other relevant regulations) must be demonstrated after any plant changes to the revised design basis (LBLOCA with offsite power available, and Small Break LOCAs, both with and without LOOP).

The following discusses the overall philosophy and rationale for the approach taken in this study. The details of this approach follow in the subsequent sections of this LTR.

From the beginning, key questions were asked that would help focus the approach, such as:

Definition of a "Large" Break LOCA

The existing regulations cover a spectrum of break sizes from a very small leak up to the assumed double-ended guillotine break of the largest pipe in the reactor coolant pressure boundary. Underlying this spectrum of break sizes is a corresponding expected frequency of occurrence, with the frequency being inversely proportional to the break size - the larger break areas being significantly less likely to occur than small cracks or leaks. So, based upon likelihood of occurrence, some portion of the break spectrum can be relegated to severe accident space. While the specific value for the pipe size that demarcates a "large break" in the total break spectrum is being debated in other forums, it is not essential to this LTR. In our approach any combination of break size (frequency), combined with the probability of LOOP, of less than the RG 1.174 criterion for "small increase" in core damage frequency (CDF) of $1x10^{-6}$ per reactor year, is acceptable.

Application of the Risk Metrics in RG 1.174

Because the BWROG believes that the changes afforded by this re-definition of the design basis can lead to an improvement in overall reactor safety, it was decided in developing this strategy that the evaluation could be on an overall risk balance basis, i.e., that the incremental increase in core damage frequency from relaxing LBLOCA/LOOP mitigation capability would be offset (in whole or in part) by the improvements in plant response capability to other, more-likely, events. This is consistent with RG 1.174 guidelines for combined change requests (CCRs).

In our approach, by definition, the LBLOCA/LOOP combination would be assumed to "go straight to core damage," i.e., without a success path in the PRA. This case would set the base PRA CDF. Then, the proposed changes in plant design and operation would be shown as offsets to this increase, with an overall goal of being risk neutral. In our "risk-balance" approach, the RG 1.174 criterion for core damage frequency of $1x10^{-6}$ per reactor year for being a "small risk increase" is assigned to the LBLOCA/LOOP event frequency and is the target improvement in reliability needed to offset this increase. Based upon other recently published studies (References

3, 8, and 10), this value is generally consistent with the results of those studies. Section 4.0 and Appendix C of this LTR discuss this more fully.

Probability of a LOOP Event

As with any use of historical data in decision-making, there is inherent uncertainty about its ability to predict the future. However, this uncertainty can be accounted for and should not preclude good judgments from being made. Specific performance monitoring and feedback can be included in the process to account for any changes in the future. This approach is consistent with other risk-informed decisions, as outlined in RG 1.1.74. As discussed in the definition of a "large break" LOCA, the approach taken here was to assume that the overall combination of the LBLOCA/LOOP results in core damage at an assumed frequency of $1x10^{-6}$ per reactor year. The implementation guide (Section 9.0 of this LTR) includes checks to ensure that the LOOP frequency inferred by historical data remains valid for the applicant plant.
In addition, with the redefinition of the design basis event, the risk assessment of the LBLOCA/LOOP combination considers scenarios previously excluded from the licensing basis, such as LBLOCA with delayed LOOP (not synchronized at the beginning of the event), which includes LBLOCA with consequential LOOP, where the loading of the large ECCS equipment onto the electrical busses causes a drop in local grid voltage, such that the plant's protective relaying in the switchyard will separate the plant from offsite power. Inclusion of these scenarios in the generic risk assessment demonstrates that the overall risk balance is improved by reducing the demands on the EDGs that can lead to performance problems in responding to these LOOP events.

Identification of the Option 3 Implementation Options

With such a fundamental change in the design basis, concerns were expressed that implementation could be too open-ended. The approach was to focus only on the largest break sizes that drive the design requirements on the mitigating equipment. Thus, in the risk-balanced approach described above, the overly restrictive design requirements that created burden should be identified, and then those requirements should be relaxed only to the point where the burden was eliminated. This relaxation should not be at the expense of no longer being able to: 1) meet the revised design basis, and 2) to show acceptable mitigation for the new beyond design basis event.

To accomplish the above, surveys were sent out to the BWR licensees to identify and rank the possible design requirements that could be revised with the proposed change in licensing basis. The survey results were screened for feasibility. Section 3.0 of this LTR is the final result of the survey. One of the results of this survey is that some items can be mutually exclusive, as discussed further in the thermal-hydraulic analysis in Section 5.0. These are identified in the implementation guide in Section 9.0.

Implementation Guide

In keeping with the general approach of an LTR, the feasibility of the changes is demonstrated as broadly as possible. To the extent practical, the evaluations should be bounding over the entire fleet of plants, such that minimal plant specific analysis is needed. With that concept, this LTR attempts to use a broadly applicable set of base cases, with sufficient sensitivity studies to ensure most, if not all, of the fleet is covered. This approach was applied to both the risk evaluations (Section 4.0) and to the thermal-hydraulic analyses (Section 5.0). The implementation guidelines provided in Section 9 will guide licensees in determining whether they are fully bounded by the LTR, or need to do specific evaluations for their plant.

Use of a "Generic" PRA Model

While the PRA model used is characterized as "generic," it is actually an operating BWR/4 plant's model, with the unique plant specific design attributes replaced with more common design features. Various configurations of the generic model were used in this LTR to ensure that relevant plant differences were addressed in the underlying analyses. It should be noted that this model has been used by the BWROG in other regulatory actions, such as Risk-Informed Technical Specification (RITS) changes. The input for the changes to "genericize" the model comes from surveys of BWR licensees to describe those attributes of their PRA models that are germane to this study (i.e., that are important to the LOCA and LOOP event sequences.) The results of these surveys were used to create the necessary permutations of the base model to capture these important attributes (Section 4.0 and App. C). Sensitivity studies are used to cover the most-common plant-to-plant design differences. These are factored into the implementation guide in Section 9.0.

Application of the "defense in depth" and "safety margin" philosophies of RG 1.174

As discussed earlier, some measure of mitigation capability should be retained for the new beyond design basis event of the LBLOCA/LOOP, assuming the other plant changes were implemented. We chose to use a classical, deterministic definition of defense in depth for fission product barriers for simplicity. Because this is now a severe accident, some level of core damage can be tolerated. In addition, by definition, a LOCA is a breach of the fission product barrier of the reactor coolant pressure boundary. Thus, to ensure that the next fission product barrier of the containment was not challenged by this new event, it was decided that retaining the core material inside the reactor vessel would be used as the acceptance criterion for mitigation, i.e., maintaining the fuel in a "coolable geometry" was chosen to ensure that the core material was retained in the vessel. As an extra conservatism (i.e., to show "safety margins" as specified in RG 1.174), the existing 10 CFR 50.46 acceptance criteria of 2200 °F and 17% cladding oxidation fraction were chosen as the figures of merit for demonstrating that a "coolable geometry" is maintained. Best-estimate thermal-hydraulic models were used to demonstrate that the coolable geometry was maintained after the assumed plant changes were implemented. Best-estimate modeling is acceptable per RG 1.174, as we are dealing with a severe accident, i.e., beyond the requirements of the design basis.

For the ease of use of the implementation guide in Section 9.0, a commonly used thermal hydraulic model for PRAs, the MAAP4 code, was utilized in the defense-in-depth thermal-hydraulic study. Because the MAAP4 code has not been subjected to regulatory review and approval, the results of the MAAP4 analysis were not used to demonstrate conformance to the acceptance criterion, but only as a guide to identifying the most challenging set of plant changes in the implementation guide. These cases were then analyzed with the TRACG02 code, an NRC-recognized model, to demonstrate that the coolable geometry was ensured. By benchmarking the MAAP4 results back to TRACG02, individual plants can assess their plant's ability to demonstrate "defense-in-depth" without having to run the more-costly TRACG02 code when applying for this regulatory change, as outlined in Section 9.0.

The following sections of this LTR discuss the specific details of how this overall approach was executed in this evaluation.

3.0 Description of Changes

The existing regulations requiring that all possible breaks be mitigated using onsite power sources drive many of the design and operating requirements and parameters. Some of these requirements and parameters exist for the sole purpose of mitigating the largest, least likely breaks in the absence of offsite electrical power. Implementation of these requirements and parameters can make the performance of mitigating systems less than optimal for many of the more likely events. If the performance of the mitigating systems can be optimized over the entire range of challenges, then overall plant risk can be reduced.

The following sections describe the plant changes that were analyzed in this report. Current requirements and the rationale for change are described. The risk impact of the changes is presented in Section 4.0.

It is possible that other plant changes that are not explicitly described in this report can be justified using the analyses presented herein. A licensee that wishes to implement another change must demonstrate that the change is equivalent to those presented in this report. If the change is not equivalent, the plant must provide a plant specific evaluation similar to that presented in this report; and the plant must obtain approval from the NRC that the analysis method of this report is applicable to that change.

3.1 Allow EDG Warm Up Prior to Loading

Current EDG start and load times are set to a time short enough that the core can be re-flooded prior to core damage for all break sizes. The start time is typically on the order of 10 seconds and is referred to as a "fast start". It is generally accepted that fast starts and loading take a toll on the components in the engine[1]. Historically, this rationale has been accepted by the NRC as a basis for reducing the number of fast start tests that are required. Fast start tests, however, have not been eliminated. In addition, inadvertent and anticipatory logic starts of the EDGs are fast starts.

The requirement for fast start tends to decrease the reliability of the EDGs due to the component degradation associated with operating the equipment in this manner. This requirement also tends to increase the unavailability of the EDGs, because many current maintenance outages are

[1] For this report, it was assumed that the reliability benefits were the same when changing from fast start to slow start as from fast load to extended load. Therefore, discussion of reliability benefits of changing from fast to slow start applies to changing from fast load to extended load. It is expected that a licensee applying for this change would determine the optimum combination.

focused on degradation associated with fast starts and loading, based on the experience of BWROG members.

If the requirement for fast start were eliminated, the EDG starting circuits could be modified to allow an interval of warm up at idle speed before accelerating the engine to full speed and loading the bus. Diesel engineers at BWROG member plants have indicated that any warm up time would be beneficial, but 30 seconds to a minute warm up duration is expected to yield quantifiable benefits. GE analysis of industry data indicates that diesel start and load times of less than 100 seconds would satisfy current 10 CFR 50.46 requirements for all but the large recirculation loop pipe breaks. (For these smaller breaks, the controlling factor is the time to depressurize to the pressure permissive; this is greater than 100 seconds.) Analyses using realistic assumptions have shown that acceptable PCT values would be maintained for breaks of all sizes.[2] Therefore, the effect of fast start elimination would be an increase in the reliability and availability of the diesel generators for nearly all accident sequences at the cost of a reduction in safety margin for a subset of the LBLOCAs when offsite power is not available.

3.2 Optimize the Loads Sequenced on to the EDGs

Currently, loads that are automatically sequenced on to the EDGs include a large number of high capacity pumps needed to rapidly reflood the core following the largest breaks. The EDGs are designed to carry these loads, but little more. If some of the high capacity pumps were removed from the EDG automatic loading, other more beneficial loads could be automatically powered by the diesels in other, more likely, loss-of-offsite power scenarios.

Current plant procedures allow operators to secure pumps not needed for core cooling, which would allow other loads to be powered by the diesels. This, however, requires the operators to take manual actions at a time when the limited operator resources could be assigned to other tasks. Further, PRAs have generally considered such manual actions to be less reliable than automatic actions.

Some members of the BWROG have identified several types of equipment that would be valuable in LOOP scenarios in which there is not a LBLOCA. These include battery chargers, drywell coolers, and some equipment closed cooling loops. Conversely, they have indicated that automatic initiation of LPCS pumps does not provide significant benefit in such a scenario. Others have indicated that starting only two of four LPCI pumps would have a negligible impact on LOOP scenarios not involving a LBLOCA.

[2] Plants must still demonstrate that small breaks are acceptable using current 10 CFR 50.46 criteria. See Section 9 for a more complete description of this requirement.

If the requirement for automatic loading of all LPCI pumps or LPCS pumps onto the diesel generators were eliminated, licensees would perform analyses to determine which equipment would be most beneficial to have automatically loaded. They may also determine that some equipment that is currently load shed upon loss of bus voltage may not need to be shed.

The potential detrimental effect of this change would be that some LBLOCA events with a concurrent loss-of-offsite power and an additional single failure may lead to core damage. The beneficial effect of this change depends on the plant's use of the additional load margin on the diesels. In Section 4.0, this report evaluates the effect of automatically powering the battery chargers from emergency onsite power and eliminating automatic load of LPCS pumps or some LPCI pumps onto the EDGs. (In this analysis, all low pressure ECCS pumps are assumed to automatically start on a LOCA signal as long as offsite power remains available.) This has the benefit of reducing operator burden during loss-of-offsite power sequences, e.g., by eliminating the need to reduce DC loads, monitor battery consumption, and manually re-start the chargers. There are also fewer pumps that need to be secured by the operators that are not needed for extended core cooling. This benefit has been evaluated using the generic PRA model discussed in Section 4.0. Other benefits that are more difficult to quantify include additional voltage margin afforded DC powered equipment if the chargers can be credited in the plant safety analysis.

3.3 Start EDGs Only When Needed

The EDGs are currently started on any single indication of a LOCA (i.e. high drywell pressure or low reactor water level), regardless of the state of the offsite power sources. This has caused actuation of the diesels when they were not needed, resulting in unnecessary demand on the equipment. Additionally, personnel are required to secure the unneeded diesel, diverting resources that could have been used to directly address the transient.

If the requirement for this anticipatory start were eliminated, licensees could change the EDG start logic so that the engine would start only on bus under-voltage or degraded voltage conditions, i.e., anticipatory starts on ECCS logic accident signals would no longer occur. The only scenarios whose response would be adversely affected by this change would be a subset of very large recirculation line breaks in which offsite power was lost as a consequence of starting the ECCS system. If the power loss occurred concurrent with or a few seconds following the LOCA, the current accident analysis would be virtually unchanged. If the power loss occurred more than a few minutes after the LOCA, reflood would have been substantially complete, peak clad temperature would have already been reduced, and the additional time to start the EDGs would not affect long-term cooling.

One of the safety benefits of eliminating the anticipatory EDG starts comes from the reduction of operator burden following accidents and transients. When a diesel generator is started, but not needed to power plant loads, it runs unloaded. The BWROG diesel generator engineers have indicated that running unloaded for an extended period of time is detrimental to the machine. Currently licensees secure unneeded diesels to prevent damage to the machine. This can take the attention of several operators. When multiple diesels start, as is the case for the anticipatory logic, the situation is exacerbated. Therefore, if the operators do not have to divert resources to tending diesel generators, one can assume that other post initiator operator actions assessed in the PRA would be somewhat more reliable. This effect was analyzed using the generic PRA model described in Section 4.0

Another safety benefit of eliminating the anticipatory EDG starts comes from an increase in diesel availability and reliability. When spurious EDG starts occur, remedial actions, such as running the EDG under load for a period of time to clear cylinder soot buildup, are typically necessary to restore the equipment to full integrity. This effect has been analyzed using the generic PRA model discussed in Section 4. Additionally, a diesel generator that has been unnecessarily started during an accident or transient and has been successfully secured may incur damage if offsite power is subsequently lost and an actual start demand occurs. One mechanism for this damage is that the hotter oil in a recently shutdown EDG does not lubricate all portions of the EDG, such as turbochargers, as well as if the oil was at normal, standby temperature.

A third safety benefit of eliminating the anticipatory EDG starts is that there should be fewer spurious EDG actuations. A reduction of the number of signals that will cause a start will result in a reduction of the number of spurious starts. Since any reduction in demands reduces wear on the equipment, unavailability should decrease and reliability should increase as a direct consequence of this change.

3.4 Simplified EDG Testing

Testing is required on the current functions of starting and loading EDGs. Changes under this option would not alleviate the need for such testing, but the acceptance criteria for such testing would be relaxed and the overall testing scope would be significantly reduced. Significant maintenance time associated with diesel generators is spent meeting these acceptance criteria. This affects both the generator and the logic associated with shedding equipment and the subsequent reload of required equipment. Redefining these acceptance criteria would not significantly affect any but the most limiting LOCA scenarios. Relaxing these acceptance criteria has the potential to reduce unavailability of the EDG function. There is an additional benefit that some of the tests could be simplified, which in turn could result in fewer operator distractions during plant operation.

For example, to satisfy the accident response assumptions associated with a LLOCA concurrent with LOOP, RHR pumps must load onto the DG-powered board immediately (typically less than 1 second) after the DG ties to the board. The DG is therefore subjected to the application of a very large load just a few seconds after its cold, fast start. Additional loads are sequenced onto the DG in fairly quick succession. The timing relays which accomplish this loading have tight tolerances, both to assure reflood times are within those assumed in the accident analyses, and also to ensure that the DG can recover adequately before the next load is applied.

Upon separation of the LOOP and LOCA events as requested by this LTR, the DG is allowed a warm-up period prior to connecting to the associated electrical board. The start times of the RHR pump and other loads are not as critical in a smaller break scenario, so the timing relays' tolerance need not be so tight, and the DG can be allowed a greater recovery time between load applications. The longer start times impose less stress upon the DG and require less timing precision in the loading sequences, while maintaining adequate margins to accident analyses assumptions.

Additionally, since DG's will no longer have to carry all of the large pump motor loads associated with LBLOCA mitigation, the DG's will have additional capacity to supply other loads that often have to be shed under the current regulatory requirements. Testing, maintenance, and plant perturbations associated with this load shedding could be greatly reduced and in some cases would no longer be necessary at all. The reduction or elimination of the complex plant equipment manipulations generally necessary for such load shed logic testing and maintenance would allow the plant staff to focus its attention on more risk-significant activities.

3.5 Increased MOV Stroke times

Current MOV stroke times are set short enough so that the core can be re-flooded prior to core damage for all break sizes following a loss-of-offsite power and subsequent EDG start and load. This has caused plant designs to require fairly large valve operators to meet the open or close times. The combination of oversized operators and severe test conditions causes significant wear on the valves. One BWR has experienced severe valve damage during a test under these conditions. In addition, the electric power requirements for the larger operators can add to EDG loading constraints.

Separating LBLOCA from LOOP will not alter test conditions, but will allow slower valve stroke times. If the stroke time were slower, lower torques would be required to change the state of the valve. This would allow slower operating speeds for the motors and gears, thus allowing smaller operators, or more margin for the current operators (Reference 9).

As with the EDGs, valves that are made more reliable through a reduction of wear also have better availability. The potential detriment would be that a subset of LBLOCA with LOOP scenarios might lead to core damage. However, as discussed in Section 5.0, thermal-hydraulic analyses using realistic analyses have shown that adequate PCT would be maintained for a wide range of stroke time relaxations. The maximum time increase would be limited by the acceptance criteria for the LBLOCA.

An additional benefit from this change would be that smaller operators would reduce seismic loading on some plant pipes. This report does not attempt to quantify the magnitude of the seismic benefit of this change.

3.6 Automatically Start One RHR Loop in Suppression Pool Cooling Mode

Currently, instrumentation logic is such that both RHR divisions start in LPCI mode on a LOCA signal. The logic signal is locked in until the core reflood is complete. Later in the postulated accident sequence, at least one division of RHR is required to be manually re-aligned to one of the containment cooling modes. While this strategy works well for LBLOCA scenarios, it is not as effective for smaller breaks and transient scenarios.

If LBLOCA scenarios did not require consideration of concurrent loss-of-offsite power, many BWRs would be able to modify their RHR logic such that at least one RHR loop would be started automatically in suppression pool cooling mode. This eliminates the need for a manual mitigation action that is important in all LOCAs and transients. To make this change, a licensee would need to deterministically demonstrate that it could still mitigate the LBLOCA with offsite power available and a single active failure.

Making this change would reduce CDF associated with sequences involving loss of containment heat removal because the main system for performing this function would be initiated automatically rather than manually. The change would increase CDF for sequences involving loss of injection, because only one division of RHR would be started automatically in LPCI mode. Further, LPCI could not be used to automatically reflood the core in those LBLOCA scenarios in which the break is in the recirculation loop that receives the LPCI flow. Because loss-of-containment heat removal sequences make up a larger fraction of the BWR risk profile than loss of injection sequences, this change would result in a net reduction in CDF.

3.7 Eliminate LPCI Loop Select

Some BWRs have a logic system called LPCI Loop-Select Logic. In the event of a Recirculation line break, this system monitors both Recirculation loops and repositions valves to isolate the broken Recirculation loop and directs all LPCI flow to the Recirculation loop that is intact, i.e., does not have the LOCA. To assure that the required valves will have power under all assumed single failures, they are powered from a "swing bus" that is power-seeking from either division of on-site or off-site AC power. This logic is very complicated, and therefore it is difficult to maintain and test. A typical test of this system includes multiple jumpers, lifted leads, and blocked relays to simulate the accident condition while protecting other equipment from damage. Elimination of this testing greatly reduces the potential for restoration errors or equipment damage during the testing.

In the current LOCA analyses for Loop-Select plants, the logic is assumed to fail (i.e. select the broken loop) for all breaks less than or equal to 0.5 ft^2. This is well into the large break range, so elimination of this function will not affect other postulated accidents. When offsite power is available, there are no single failures that would prevent reflood following a LBLOCA if Loop-Select were eliminated.

4.0 Risk Impact of Changes

The probabilistic risk effect of eliminating a large break LOCA concurrent with a loss-of-offsite power as a design consideration is addressed in this section. As indicated earlier, an increase in the calculated core damage frequency (CDF) occurs because the LBLOCA/LOOP contribution must be assumed to equal the initiating event probability, i.e., no credit is allowed for mitigation. Conversely, some beneficial effects on more likely events are anticipated, primarily due to increased emergency diesel availability and ECCS logic changes. These effects have been evaluated with a "generic PRA model".

4.1 Generic PRA Model

The generic PRA model was constructed so that it can model many different BWR configurations. To evaluate the changes outlined in this report, nine explicit BWR configurations were evaluated. Other plant configurations were addressed in sensitivity analyses. The model is an integrated Level 1 and Level 2 for internal events that can occur while the plant is operating at full power. The model explicitly covers front line systems and support systems. These systems are modeled to the super-component level (e.g., a MOV includes the valve, motor, and operator). The human error analysis includes both pre- and post-accident operator actions. Appendix C provides the details of the model.

To ensure that the conclusions drawn from the generic model are broadly applicable to the BWR fleet, the generic model results were compared to several plant specific BWR PRA models. The focus of this comparison was the LOOP event. This is because the largest portion of the risk offset is recovered from LOOP events. As shown in Appendix C, the generic model is able to approximate the results from the plant specific models.

In addition to benchmarking the generic model with various BWR PRA models, a set of uncertainty and sensitivity analyses were performed. The uncertainty study performed was a Monte-Carlo analysis of the basic event frequency and probability distributions. Sensitivity analyses were performed for several key attributes and assumptions in the model. The details of these analyses are presented in Appendix C. Assumptions or attributes that are important to the conclusions of this report are outlined in Section 9.0.

4.2 Risk Impact of LOCA/LOOP

The changes proposed do not affect the entire range of LBLOCA events. The lower end of the large LOCA break sizes and any LOCAs involving breaks above the core (i.e. steam line breaks or SORVs) would not be significantly affected by the changes described in Section 3.0.

To provide conservatism, it is assumed that all large-break LOCAs will result in core damage if offsite power is not available. Both the NRC and the industry have provided estimates of the frequency of LBLOCA and the probability of a LOOP given a LBLOCA. The product of these numbers gives the LBLOCA/LOOP frequency. Using this data, the range of LBLOCA/LOOP frequencies is estimated to be between 3.0×10^{-7} and 1.2×10^{-6} per year (References 2, 3, and 10).

The generic model uses a conservative mean value of 1.0×10^{-6} per year as the increase in CDF associated with the deterministic elimination of the capability to mitigate a concurrent LBLOCA and LOOP.

Large Early Release Frequency (LERF) contribution from LBLOCA with injection failure sequences is small at BWRs. A typical number for the conditional probability of LERF given a LBLOCA with loss-of-injection is about 0.01 (Reference 12)[3]. This small fraction is a result of several factors, in particular that vessel failure occurs at low RPV pressure and there is water on the drywell floor at the time of vessel failure. None of these factors is affected by the proposed separation of LBLOCA and LOOP. This leads to a maximum LERF increase of 10^{-8} per year, which is about an order of magnitude below the RG 1.174 guidelines.

4.3 Risk Impact of Option 3 Changes

As indicated in Section 3.0, the proposed changes may result in risk reduction. The risk reduction associated with some of these changes can be explicitly quantified, as discussed in Sections 4.3.1 through 4.3.3. Other changes cannot be explicitly quantified, but can be qualitatively evaluated in terms of plant changes and their potential for reducing the CDF, as discussed in Section 4.3.4.

4.3.1 Quantitative Impact of Optimizing EDG Loads

In the description of the changes, it was postulated that LPCS or some LPCI pumps could be eliminated as equipment that is automatically loaded on to the diesel generators, while battery chargers could be added. This provides a trade-off. If the chargers are automatically loaded on the emergency buses, the operator actions associated with DC load shed or re-start of the chargers would be rendered unnecessary. Low pressure ECCS pumps that are not automatically loaded following a loss-of-offsite power could be subsequently added manually in the event that other systems were unable to provide core injection.

The PRA model was modified to address these configurations. The 125 V DC model was changed such that if offsite power were lost, the operator action to load the chargers would be eliminated from the model. The LPCS model was changed by adding an operator action to manually start these pumps following a loss of offsite power. The LCPI model was changed in a similar manner, but the operator action was only included for two of the four LPCI pumps.

Quantification of these configurations leads to a CDF decrease of 1.2×10^{-8} per year if the battery chargers are automatically loaded instead of the LPCS pumps and 1.4×10^{-8} per year if the battery chargers are loaded instead of the LPCI pumps. In both cases, the model shows a negligible change in LERF.

[3] Large break LOCA with failure to scram has a significantly higher LERF fraction, but because of the small contribution of ATWS in LBLOCA events, the overall conditional LERF probability remains 0.01.

4.3.2 Quantitative Impact of Starting One RHR Loop in SPC Mode

Another postulated change was to start one loop of RHR in suppression pool cooling mode rather than LPCI mode. With this change, the operators would not need to start suppression pool cooling for scenarios in which the pool water temperature increased above a given value. If the other core injections were not available, the loop selected for automatic SPC could be manually aligned to the LPCI mode instead.

In the PRA model, eliminating the operator action from one loop of RHR for suppression pool cooling simulated this. An operator action to manually start the loop in LPCI mode was added to the LPCI model for that loop. The other loop retained the original configuration.

Quantification of this configuration leads to a CDF decrease of 4.1×10^{-8} per year. The model shows a negligible change in LERF.

4.3.3 Quantitative Impact of Eliminating LPCI Loop-Select Logic

In the past, LPCI Loop-Select has been removed from BWRs through elaborate redesign and reconfiguration. The change proposed here is to eliminate the feature in a straightforward manner. The crosstie valve would be locked closed, and the logic would be disabled.

Quantification of this change leads to a CDF increase of only 1×10^{-9} per year. The model shows a negligible change in LERF.

4.3.4 Qualitative Risk Reductions

As indicated above, some potential benefits associated with the changes described in Section 3.0 in terms of CDF are envisioned, but not directly quantified. Three aspects were examined:

- Diesel Generator Availability
- Diesel Generator Reliability
- Operator Action Reliability

It is difficult to predict the actual reduction in unavailability and unreliability of the diesel generators associated with the proposed changes. The effect is not expected to be large, but even small improvements can substantially offset the postulated risk increase arising from the assumption that all LBLOCA/LOOP events result in core damage.

A reasonable reduction in unavailability and failure rates as a result of the changes described in this report would be 10%. For example, the average unavailability of EDGs is 1.01×10^{-2}. Assuming an average annual required availability time of 8760 hours, the total unavailability per EDG is about 88 hours. Nearly a 10% reduction in unavailability would be realized if one shift (8 hours) of down time per year was avoided. Proposed Revision to NEI 99-02 (Reference 5) reports the probability of an EDG failure to start on demand is 1.1×10^{-2}. This translates into about one failure per plant every 3 years, assuming the generators are started monthly and the average BWR has 2.5 diesels. Extending this to one failure per 3.3 years results in a 10% reduction of the failure probability.

It is similarly difficult to predict the reduction of the operator action failure rates. Once again, though, a moderate decrease of 10% is expected if the proposed changes are made. This only applies to actions in the model that share dependence with the actions to secure a diesel generator that has spuriously started. These are defined as those actions taken outside the control room, that need to happen within one hour of the initiating event.

Rather than specifying an impact on the parameter changes and re-quantifying the model, the risk impact was evaluated examining a reasonable range of improvements of the affected parameters. Depending on the plant configuration, improvements as small as 10% in EDG related terms could offset approximately 47% of the CDF associated with eliminating the licensing requirement for LBLOCA/LOOP. These EDG improvements could completely offset the change in LERF. Plants that rely more on their EDGs (i.e. those plants with a less reliable local grid) would benefit more from these changes. The results for all analyzed plant configurations are presented in Appendix C.

In addition, improving operator performance by 10% can offset approximately 31% of the assumed increase in CDF associated with eliminating the licensing requirement for LBLOCA/LOOP.

4.4 Risk Balance

Sections 4.2 and 4.3 described the risk associated with the changes on an individual basis. When implementing this LTR, plants will combine the assumed risk increase associated with eliminating the licensing requirement for LBLOCA/LOOP analysis with the risk decreases that emerge as a result of implementing the selected Option 3 changes. It is expected that plants will implement some combination of the proposed Option 3 changes.

The risk balance will be different for each plant implementing selected changes outlined in Section 3. All implementations, however, will realize a CDF increase that is less than simply assuming that all LBLOCA/LOOP scenarios go directly to core damage[4].

For example, a BWR4, without an AC independent low pressure injection system or an additional diverse AC onsite power source, that implements the EDG reliability enhancing changes, removes two LPCI pumps from automatically loading onto onsite power, replaces the load with battery chargers, and allows one RHR loop to initiate automatically in SPC mode would have an overall CDF increase of 2.03×10^{-7} per year. The change in LERF would be negligible.

A similarly configured BWR 6 would have an overall CDF increase of 2.78×10^{-7} per year. Once again the change in LERF would be negligible. Appendix C contains a comprehensive list of all analyzed configurations and combinations of changes.

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[4] There is one exception to this statement. The CDF increase is higher than 10^{-6} by a very small amount if elimination of Loop-Select is the only change made to the plant. However, combining this change with any of the others described in this report will result in a CDF increase that is less than 10^{-6}.

As can be seen from the results presented in this section, the overall effect of implementing the changes described in Section 3 would be a change in CDF that is considered very small. Changes of this magnitude would pose no undue risk to the public.

5.0 THERMAL HYDRAULIC ASSESSMENT

As outlined in Sections 2 and 7 of this report, the defense-in-depth philosophy has traditionally been applied in reactor design and operation to provide multiple means to accomplish safety functions and prevent the release of radioactive material. In order to demonstrate defense-in depth for the LBLOCA-LOOP separation exemption, per RG 1.174, we will demonstrate that the LBLOCA/LOOP event can continue to be mitigated, given the effect of the plant modifications discussed in Section 3.0 (Option 3 changes).

Plant changes associated with implementation of Option 3 will result in a delay in the start of ECCS injection. They will also result in converting some equipment from automatic to manual operation and vice versa. Therefore, the impact of these changes was evaluated to determine the viability of implementing the changes. Given the low probability of a LBLOCA occurring concurrent with the loss of offsite power (LOOP), this event will now be considered to be a "severe accident." Consistent with this re-categorization, this evaluation was performed using realistic assumptions and methods. Consistent with the use of best estimate methods and assumptions, the acceptance criterion for the analyses was established to be maintaining the core in the reactor vessel in a coolable geometry.

As discussed earlier, some measure of mitigation capability should be retained for the new beyond design basis event of the LBLOCA/LOOP, assuming the other plant changes were implemented. We chose to use a classical definition of defense in depth for fission product barriers for simplicity. Because this is now a severe accident, some level of core damage could be tolerated. In addition, by definition, a LOCA is a breach of the fission product barrier of the reactor coolant pressure boundary. Thus, to ensure that the next fission product barrier of the containment was not challenged by this new event, it was decided that retaining the core material inside the reactor vessel would be used as the acceptance criterion for mitigation. Maintaining the fuel in a "coolable geometry" was chosen to ensure that the core material was retained in the vessel. As an extra conservatism (i.e., to show "safety margins" as specified in RG 1.174), the existing 10 CFR 50.46 acceptance criteria of 2200 °F peak cladding temperature (PCT) and 17% local cladding oxidation fraction were chosen as the figures of merit for demonstrating that a "coolable geometry" is maintained.

The TRACG02 computer code was used to evaluate the dynamic response of realistic cases against the acceptance criterion. A series of scoping analyses were performed with MAAP4 to better focus the TRACG02 analysis effort. To understand the impact of code differences, a series of benchmark analyses were performed. However, the TRACG02 and MAAP4 PCT values compared reasonably well.

Additional MAAP4 cases were analyzed to assess the impact of plant-to-plant variability on the results of the analysis. It was demonstrated that changes in RPV liquid volume of ±20% and a variation in ECCS injection flow of 10% did not generate significant changes in PCT that would alter the conclusions of this evaluation.

The details of the thermal hydraulic assessment described above are provided as Appendix B to this report.

5.1 Methodology

The specific approach used to evaluate the impact of the proposed Option 3 changes was as follows:

- Benchmark LBLOCA cases were performed with both the MAAP4 and TRACG02. The key phenomena important to a LBLOCA analysis were compared between the two codes and adjustments to the inputs and assumptions used in MAAP4 were made to achieve general agreement. The benchmark analyses are discussed in Appendix B, Section B.4.
- Base cases were identified that included the equipment and plant variations, which were expected to bound the plant variability within the BWR fleet.
- A series of MAAP4 scoping cases was performed using the various equipment combinations resulting from implementing all of the Option 3 changes. This method was used to identify the limiting cases for further TRACG analysis and is discussed in Appendix B, Section B.5.
- A series of MAAP4 sensitivity cases to evaluate the effect of plant-to-plant variability were performed. The sensitivity analysis is discussed in Appendix B, Section B.6.
- The MAAP4 scoping case with the highest PCT for the BWR4 and one for the BWR6 were then run using TRACG02 to generate the actual PCTs used to demonstrate defense-in-depth with implementation of Option 3. It should be noted that the BWR6 case, although not the highest PCT, was chosen because it offered the opportunity to evaluate the impact of the Option 3 changes in scenarios requiring both high pressure and low pressure injection sources. The difference in peak PCT between the BWR6 case selected and the BWR6 case with the highest PCT is less than 100 °F and there is significant margin to the 2200 °F acceptance criteria. The final TRACG02 analyses are provided in Appendix B, Section B.7.

5.2 Summary of Results

The representations of Option 3 plant changes are provided in Table 5-1 below. Table 5-2 provides the limiting combinations of ECCS delay and available ECCS systems derived from Table 5-1.

The TRACG02 and MAAP4 PCT values compared reasonably well in the benchmark of the two limiting cases. MAAP PCTs were higher than the comparable adjusted TRACG02 PCT results. This result provides added confidence in using the MAAP4 cases in the scoping study.

The MAAP4 sensitivity analyses demonstrated that the plant-to-plant variation in RPV liquid volume and ECCS injection flow did not generate changes in PCT that would alter the conclusions of the evaluation. This range of RPV liquid volume and ECCS injection flow encompasses the expected plant type variability.

The objective of maintaining the core in the reactor vessel in a coolable geometry was met with significant margin. The TRACG02 cases for the limiting scenarios predict a peak clad temperature that is less than 2200 °F (1951 °F for the BWR 2/3/4 and 1634 °F for the BWR 5/6) and no significant clad oxidation (much less than 1% global and less than 5% local clad reacted).

Therefore, it was demonstrated that defense-in-depth, as specified in RG 1.174, has been achieved, with the adoption of the Option 3 changes.

Table 5-1
Bounding Representation of Option 3 Plant Changes

Category	Change	Effect	Bounding Effect on ECCS		
			BWR 3/4	BWR 3/4 w/ Loop-Select	BWR 5/6
1	Eliminate EDG fast start Eliminate anticipatory EDG start Increased MOV stroke times	Delayed Injection	90 Second EDG Delay[1,2]		120 Second EDG Delay[1]
2a	Optimize EDG loading	Reduced Injection	No LPCS		Loss of 1 LPCI pump and 1 LPCS Pump
2b			Loss of 2 LPCI Pumps		
3	One loop of RHR in SPC mode	Reduced Injection	No LPCI [2]		Loss of 1 LPCI Loop
4	Eliminate LPCI LOOP Select logic	Reduced Injection	N/A	Loss of 1 LPCI Loop	N/A
N/A	Simplified EDG testing	N/A	N/A		

Notes:
1. Nominal valve stroke and pump coast-up time assumed for ECCS equipment operation is included in addition to the EDG delay.
2. Bounding for a BWR 2

Table 5-2
Limiting Combination of ECCS Delay and Reduced Injection

Case	Plant Type	Category[2] Combination	EDG Delay Time (sec)	Number of Injection Pumps		
				LPCI	LPCS	HPCS
A	BWR 3/4	1 & 2	90	2	0	N/A
B[3]		1 & 3		0	2	N/A
C	BWR 5/6	1 & 2 & 3	120	1	0	1
D				2	0	0
E				0[1]	1	1

Notes:
1. Includes an additional LPCI failure to make it a unique case.
2. Category corresponds to Table B.5-1.
3. The only combination valid for a BWR 2 plant corresponds to Case B.

6.0 Compliance with Current Regulations

The changes proposed in Section 3.0 require a specific exemption, pursuant to 10 CFR 50.12, to key provisions of 10 CFR 50.46, namely §50.46(c)(1) and (d). §50.46(c)(1) defines the spectrum of break sizes that must be postulated within the design basis, which includes the classic LBLOCA of the double-ended guillotine break of the largest pipe in the reactor coolant system. §50.46(d) specifically invokes the ECCS design requirements as currently set forth in General Design Criterion (GDC) 35 (Appendix A to 10 CFR 50), which contains the assumption of the simultaneous LOOP with the LOCA.

The proposed changes in this LTR require a limited exemption to the above requirements. Under the proposed exemption, the ECCS design basis under §50.46(c)(1) and (d) for BWRs would no longer include an assumed LOOP coincident with breaks in "large" pipes (nominally pipe diameters greater than or equal to 10 inches) in the reactor coolant pressure boundary. The coincident LOOP and LOCA assumption would, however, continue to apply for breaks smaller than this "large" break size. In addition, per GDC 35, the LBLOCA would still be postulated to occur with off-site power available.

6.1 Justification for the Exemption

10 CFR 50.12(a)(1) allows exemptions that are "Authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security." As discussed below, the requested exemption satisfies the requirements in 10 CFR 50.12:

<u>The requested exemption is not inconsistent with any law</u>

There is nothing specific in the Atomic Energy Act or other statute applicable to the NRC that requires that the LOOP assumption be applied to the design or analysis of ECCS for nuclear power plants, or that the NRC establish or implement the design criterion in question.

<u>The requested exemption would not present an undue risk to the public health and safety</u>

As described in this LTR, the overall risk to the public health and safety can be offset by the implementation of plant modifications that require issuance of the requested exemption. Approval of this exemption would make it possible for individual plants to implement the changes described in Section 3 of this LTR and thereby achieve a better balance in overall plant risk by optimizing the performance of the mitigating systems for other events that are more likely to occur than the LBLOCA/LOOP. Further, as discussed in Sections 7 and 8 of this LTR, implementation of the requested exemption will not adversely affect defense-in-depth or safety margins. Therefore, the exemption will not adversely affect the public health and safety.

The requested exemption is consistent with the common defense and security

The requested exemption does not pertain to safeguarding of Special Nuclear Material, the protection of Restricted Data, the availability of Special Nuclear Material for defense needs, or foreign control over power plant operation. Therefore, it does not have any effect on the common defense and security.

In addition, 10 CFR 50.12(a)(2) stipulates that "special circumstances" must be present before the Commission will consider granting an exemption. Special circumstances are present whenever:

1. Application of the regulation in the particular circumstances conflicts with other rules or requirements of the Commission; or
2. Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule; or
3. Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated; or
4. The exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption; or
5. The exemption would provide only temporary relief from the applicable regulation and the licensee or applicant has made good faith efforts to comply with the regulation; or
6. There is present any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption.

The BWROG believes that special circumstances, as defined above, exist, as follows:

Satisfaction of the Second Criterion for Special Circumstances

The purpose of §50.46 is to ensure a highly reliable ECCS that is capable of removing sufficient heat to assure that the fuel remains in a coolable geometry and negligible metal-water reactions take place for the postulated accident scenarios. As demonstrated herein, the underlying purpose of this regulation can still be achieved with the proposed changes.

The goal of this exemption is to improve the overall reliability of the ECCS and the supporting on-site power supply, i.e., the EDGs, by removing the existing design and testing constraints that are in place solely to mitigate the most-unlikely subset of postulated LOCA events. The evaluations in Section 4 of this LTR demonstrate that such improvements in reliability are achievable and offset the assumed increase in risk due to removing the LBLOCA/LOOP event from the design and licensing basis.

As shown in Section 5 of this LTR, the ability to achieve the underlying objective of a coolable geometry and negligible metal-water reaction remains satisfied, even with the removal of this subset of LOCA events from the design basis. Thus, the underlying objectives of §50.46 remain satisfied with the proposed exemption.

Satisfaction of the Fourth Criterion for Special Circumstances

As stated in Section 2 of this LTR, the purpose of this exemption request is to improve overall plant risk by optimizing the performance of key safety system equipment to respond to the most likely set of events, by eliminating the performance requirements on that equipment that exist solely to respond to a small subset of events that are known to be highly improbable. The evaluations in Section 4 of this LTR demonstrate that such improvements are achievable and can offset the assumed increase in risk due to removing the LBLOCA/LOOP event from the design and licensing basis.

Satisfaction of the Sixth Criterion for Special Circumstances

Since the subject regulations were promulgated in the early 1970's, a substantial amount of research and development devoted to piping failure mechanisms, accident thermal-hydraulic phenomena, and quantitative plant risk assessments has taken place. Application of these technologies demonstrates the excessive conservatism in the original regulations. As noted in Section 1 of this report, so much conservatism exists that it constitutes a burden on the licensees that warrants relief through both permanent rulemaking and, in the interim, exemption to these requirements. And, as demonstrated in Section 4 of this report, such relief could be used to make improvements in overall plant safety that would otherwise not be possible. Thus, it would be in the public interest to consider this new material information not present when the regulations were originally promulgated.

6.2 Compliance With Related Regulations

There are a number of additional regulations and GDCs that require the assumption of a LOOP, some in combination with LOCA. Because they are not the subjects of this exemption request, continued compliance with these regulations, considering the proposed changes described in Section 3 of this LTR, is demonstrated in the following subsections.

6.2.1 10 CFR 50.34(f) – Additional TMI Requirements

10 CFR 50.34(f)(1) has two requirements related to a LOOP. These are:

(iii) Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss-of-offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage, and

(ix) Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) systems, following a complete loss-of-offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI") (applicable to BWR's only).

BWROG Compliance

With respect to (iii) above, this issue is applicable only to B&W plants, see NUREG-0737, Item II.K.2.16.

With respect to (ix) above, as noted in Section 3 of this LTR, these plant loads are not targeted for removal from the on-site power supplies (EDGs). In addition, per Section 4 (and App. C) of this LTR, the space cooling capability to ensure reliable long-term cooling operation of the RCIC and HPCI for following a LOOP is not being reduced.

6.2.2 10 CFR 50.63 Loss of All Alternating Current Power

10 CFR 50.63(a)(1) requires, in part, "Each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout."

BWROG Compliance

The plant capability for coping with a SBO is not reduced by the requested exemption. Because the coping capability generally requires the assumption of the unavailability of both offsite and onsite AC power for the coping period duration, there is no change to analyzed capability, as the changes outlined in Section 3 of this LTR involve equipment that is not relied upon during this coping period (e.g., LPCS and RHR are AC-powered). Once onsite AC power is restored, plant modifications to the EDG loading may provide some additional capability during the recovery period.

6.2.3 10 CFR 50 Appendix A – General Design Criteria (GDC)

For certain BWRs, compliance with the GDC is not part of their licensing and design basis. The following discussion of conformance is not intended to mandate a change to any plant's licensing basis, but to illustrate that the Option 3 changes have no impact on the continued provision of the capabilities specified in the GDCs.

Criterion 17—Electric Power Systems

GDC 17 specifies, in part: "An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents."

BWROG Conformance

The proposed exemption is for the elimination of the assumption of a concurrent LBLOCA/LOOP. There is no change to the offsite power supplies being proposed by this LTR, and they remain capable of providing sufficient power to permit functioning of structures, systems, and components (SSCs) important to safety to meet the stated objectives of GDC 17. The restructuring and timing of large electrical loads onto the essential busses outlined in Section 3 of this LTR, when powered from onsite power, does not adversely impact the reliability of offsite power.

The onsite power supplies remain capable of automatically providing the required safety functions for a concurrent Small Break LOCA and LOOP. Even though the concurrent LBLOCA/LOOP would be considered beyond the plant's design basis, the realistic ECCS performance analyses in Section 5 of this LTR demonstrate that a coolable geometry for the core is maintained for the identified changes to the EDG loading and other potential plant modifications. As demonstrated in Section 4.0, enhancements in the reliability of onsite power are possible with the changes proposed in Section 3.0. As discussed in Section 7 of this LTR, containment integrity and other fission product barriers remain intact as a result of the proposed changes.

Therefore, the specified capabilities identified by GDC 17 continue to be provided.

Criterion 33—Reactor Coolant Makeup.

GDC 33 specifies: "A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation."

BWROG Conformance

The proposed exemption does not seek relief from the §50.46 requirement for an assumed concurrent Small Break LOCA and LOOP. In a BWR, GDC 33 conformance is met by any of several high pressure makeup systems, depending upon break size. Several of these makeup systems are independent of offsite or onsite AC power. As outlined in Section 3 of this LTR, no high pressure makeup systems are being revised.

Therefore, the specified capabilities identified by GDC 33 continue to be provided.

Criterion 34—Residual Heat Removal

GDC 34 specifies: "A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded."

"Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

BWROG Conformance

In a BWR, decay and residual heat are removed from the reactor during abnormal operating events by the RHR system, in conjunction with the RHR Service Water system, either directly, via the RHR-SDC or RHR-LPCI modes; or indirectly, via the RHR-SPC or RHR-Containment Spray modes.

The residual heat removal capability is not reduced by the proposed exemption. There are no changes to the offsite power supply system. The capability of the onsite power supply system to support the residual heat removal function remains unchanged. The EDG loading sequencing may change due to the elimination of the design requirement for the assumption of a concurrent LBLOCA/LOOP. Depending on the specific plant modifications implemented, it may be possible to automatically load a suppression pool cooling system, which would provide a reduced risk for event sequences requiring suppression pool cooling.

Therefore, the specified capabilities identified by GDC 34 continue to be provided.

Criterion 38—Containment Heat Removal

GDC 38 specifies: "A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

BWROG Conformance

The containment heat removal capability is not reduced by the proposed exemption. There are no changes to the offsite power supply system. The capability of the onsite power supply system to support the containment heat removal function remains unchanged. The EDG loading sequencing may change due to the elimination of the design requirement for the assumption of a concurrent LBLOCA/LOOP. Depending on the specific plant modifications implemented, it may be possible to automatically load a suppression pool cooling system, which would provide a reduced risk for small break LOCA sequences requiring suppression pool cooling.

Therefore, the specified capabilities identified by GDC 38 continue to be provided.

Criterion 41—Containment Atmosphere Cleanup

GDC 41 specifies: "Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure."

BWROG Conformance

The containment atmosphere cleanup capability is not changed by the proposed exemption. There are no changes to the offsite power supply system. The capability of the onsite power supply system with respect to the containment atmosphere cleanup function remains unchanged. No changes are proposed for the required systems.

Therefore, the specified capabilities identified by GDC 41 continue to be provided.

Criterion 44—Cooling Water

Criterion 44 specifies: "A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

BWROG Conformance

The cooling water capability is not reduced by the proposed exemption. There are no changes to the offsite power supply system. The capability of the onsite power supply system to provide cooling water remains unchanged. The EDG loading sequencing may change due to the elimination of the design requirement for the assumption of a concurrent LBLOCA/LOOP. Depending on the specific plant modifications implemented, it may be possible to automatically load additional cooling water systems on the EDGs, which would provide an overall reduced risk for events sequences requiring cooling water.

Therefore, the specified capabilities identified by GDC 44 continue to be provided.

6.2.4 10 CFR 50 Appendix R—Fire Protection Program For Nuclear Power Facilities Operating Prior To January 1, 1979

10 CFR 50, Appendix R, Section III, has four requirements related to a LOOP. These are:

> F. *Automatic fire detection.* Automatic fire detection systems shall be installed in all areas of the plant that contain or present an exposure fire hazard to safe shutdown or safety-related systems or components. These fire detection systems shall be capable of operating with or without offsite power.
>
> H. *Fire brigade* ... At least a 1-hour supply of breathing air in extra bottles shall be located on the plant site for each unit of self-contained breathing apparatus. In addition, an onsite 6-hour supply of reserve air shall be provided and arranged to permit quick and complete replenishment of exhausted air supply bottles as they are returned. If compressors are used as a source of breathing air, only units approved for breathing air shall be used and the compressors shall be operable assuming a loss-of-offsite power. Special care must be taken to locate the compressor in areas free of dust and contaminants.
>
> L. *Alternative and dedicated shutdown capability.*
>
> 1. Alternative or dedicated shutdown capability provided for a specific fire area shall be able to (a) achieve and maintain subcritical reactivity conditions in the reactor; (b) maintain reactor coolant inventory; (c) achieve and maintain hot standby conditions for a PWR (hot shutdown for a BWR); (d) achieve cold shutdown conditions within 72 hours; and (e) maintain cold shutdown conditions thereafter. During the post-fire shutdown, the reactor coolant system process variables shall be maintained within those predicted for a loss of normal AC. Power, and the fission product boundary integrity shall not be affected; i.e., there shall be no fuel clad damage, rupture of any primary coolant boundary, of rupture of the containment boundary. And,

3. The shutdown capability for specific fire areas may be unique for each such area, or it may be one unique combination of systems for all such areas. In either case, the alternative shutdown capability shall be independent of the specific fire area(s) and shall accommodate post-fire conditions where offsite power is available and where offsite power is not available for 72 hours. Procedures shall be in effect to implement this capability.

BWROG Conformance

With respect to Item F above, there are no identified changes to the fire detection system. Therefore, the current capability remains unchanged.

With respect to Item H above, there are no identified changes to the fire brigade capability. Therefore, the breathing air capability is not changed.

With respect to Item L.1 above, plant capability with respect to fire protection is not reduced by the proposed exemption. The EDG loading and sequencing may change due to the elimination of the design requirement for the assumption of a concurrent LBLOCA/LOOP. Depending on the specific plant modifications implemented, it may be possible to automatically load additional systems on the EDGs, which would provide an overall reduced risk for events sequences associated with fires. Option 3 changes must be evaluated on a plant specific basis to assure continued compliance with the requirements of appendix R.

With respect to Item L.3 above, above, shutdown capability for specific fire areas is not reduced by the proposed exemption. The EDG loading sequencing may change due to the elimination of the design requirement for the assumption of a concurrent LBLOCA/LOOP. Depending on the specific plant modifications implemented, it may be possible to automatically load additional systems on the EDGs, which would provide additional capability for some fire areas.

6.3 Effect on Plant Safety Analysis

The proposed exemption would relocate the LBLOCA concurrent with a LOOP from the current design basis safety analysis to a severe accident category. All other safety analyses in the current licensing basis (such as ATWS, SBO, etc.) would have to be reviewed for impact due to the implementation of the changes proposed in Section 3 of this LTR, not just those that are specific to LOCA and LOOP. This is required by the licensee's plant modification process, that implements the requirements of 10 CFR 50.54(a)(1) and Part 50 Appendix B (Quality Assurance), Criterion III for design control. Any such impacts would be evaluated for changes to the current safety analysis as required by 10 CFR 50.59 and any updates to the plant FSAR would be made pursuant to 10 CFR 50.71(e).

7.0 Defense-in-Depth

7.1 Regulatory Guidance

Regulatory Guide 1.174 (Reference 4, Section 2.2.1.1) – *Defense in Depth* provides the guidance for defense in depth evaluations relative to changes in a plant licensing basis. The guidance states:

The engineering evaluation should evaluate whether the impact of the proposed licensing basis change (individually and cumulatively) is consistent with the defense-in-depth philosophy. In this regard, the intent of the principle is to ensure that the philosophy of defense in depth is maintained, not to prevent changes in the way defense in depth is achieved. The defense-in-depth philosophy has traditionally been applied in reactor design and operation to provide multiple means to accomplish safety functions and prevent the release of radioactive material. It has been and continues to be an effective way to account for uncertainties in equipment and human performance. If a comprehensive risk analysis is done, it can be used to help determine the appropriate extent of defense in depth (e.g., balance among core damage prevention, containment failure, and consequence mitigation) to ensure protection of public health and safety. When a comprehensive risk analysis is not or cannot be done, traditional defense-in-depth considerations should be used or maintained to account for uncertainties. The evaluation should consider the intent of the general design criteria, national standards, and engineering principles such as the single failure criterion. Further, the evaluation should consider the impact of the proposed licensing basis change on barriers (both preventive and mitigative) to core damage, containment failure or bypass, and the balance among defense-in-depth attributes. As stated earlier, the licensee should select the engineering analysis techniques, whether quantitative or qualitative, traditional or probabilistic, appropriate to the proposed licensing basis change.

The licensee should assess whether the proposed licensing basis change meets the defense-in-depth principle. Defense in depth consists of a number of elements, as summarized below. These elements can be used as guidelines for making that assessment. Other equivalent acceptance guidelines may also be used.

7.2 BWROG Approach

As stated in Section 2.0, the central approach to Option 3 has been to achieve an improvement in the overall risk balance between the risk increase assumed with the relaxation of the LBLOCA/LOOP provisions against the risk decreases that can be gained by optimizing the equipment performance for a broader spectrum of postulated events. The BWROG approach includes both risk analysis and best estimate thermal-hydraulic evaluations. As demonstrated in Section 4.0, this improvement in the risk balance can be achieved, because as shown in Section 5.0, adequate mitigation capability remains for the LBLOCA/LOOP event after it has been reclassified from the design basis accident to a "severe accident" in the plant's licensing basis. That is, although the LBLOCA/LOOP event sequences are assumed to go to core damage in the risk assessment, in actuality, only a small subset of these sequences would actually be predicted to go to core damage. Thus, the enhancements proposed in Section 3.0 are evaluated against an artificially high threshold for being risk neutral.

7.3 Defense in Depth Evaluation

As stated in RG 1.174, consistency with the defense in depth philosophy can be maintained if certain conditions can be demonstrated. Each of these conditions is discussed in the following.

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.

  This is demonstrated by the risk assessment in Section 4.0 and thermal-hydraulic analysis in Section 5.0 that demonstrates the overall plant risk of severe core damage can be reduced by the plant modifications that address the more probable postulated event sequences rather than the highly improbable concurrent LBLOCA/LOOP event.

- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.

  No weaknesses in the current plant design have been identified. The proposed changes sought by the proposed exemption allow a means of achieving a better overall balance of plant safety versus being skewed to a small subset of highly improbable events (i.e., LBLOCA/LOOP).

- System redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences of challenges to the system, and uncertainties (e.g., no risk outliers).

  There is no change in system redundancy, independence or diversity introduced by the proposed exemption. No mitigating systems are being physically removed from the plants. Only their design features are being revised to achieve optimal performance for the full spectrum of postulated events. The risk assessment in Section 4.0 provides a demonstration that an improvement in the overall risk of severe core damage can be achieved.

- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.

  The possible plant modifications can be implemented consistent with the current plant defenses against common cause failures. Assessments of the plant modifications, consistent with established design practices under 10 CFR Part 50, Appendix B and §50.59, will be done to assure any potential new common cause failure mechanisms, although unlikely, are identified and dispositioned consistent with the plant's current design bases.

- Independence of barriers is not degraded.

In the BWR design there are five barriers to fission product release. These are:

  - o Fuel Matrix
  - o Fuel Cladding
  - o Reactor Coolant Pressure Boundary
  - o Primary Containment
  - o Secondary Containment

During normal operation, approximately 10 – 20% of the noble gases and 2-3% of the halogens will migrate out of the fuel matrix into the gap region. The remainder is contained within the fuel matrix. The other non-gaseous, non-volatile fission products are maintained as an integral part of the fuel matrix. As a result, the fuel matrix is capable of retaining fission products until temperatures approaching the melting point of the fuel are reached. The thermal hydraulic analyses, which are discussed in Section 5.0 and in Appendix B, indicate that the fuel cladding temperature during a LBLOCA/LOOP event with the proposed changes in Section 3.0, will not reach temperatures at which a substantial amount of the fission products in the fuel matrix would be assumed to be released. In this way, the fuel matrix continues to provide a substantial barrier to radioactivity release following a LOCA event.

The fuel cladding barrier may be breached if the cladding temperature reaches an excessive temperature. This current temperature limit below which excessive cladding failure is not expected is 2200°F (§50.46). Again, Section 5.0 and Appendix B demonstrate that, using realistic analysis assumptions, the peak cladding temperature is within this limit. Thus, excessive cladding failure is not anticipated.

The reactor coolant pressure boundary is assumed to fail as a part of the LOCA definition. This assumption is consistent with the defense in depth approach to plant safety.

The primary containment boundary is assumed to fail if the primary containment design conditions (pressure and temperature) are exceeded. The primary containment performance will be consistent with the current safety analysis, as no changes to the pressure suppression capability are proposed by this exemption request. Therefore, the primary containment barrier will continue to perform acceptably after implementation of the proposed changes in Section 3.0.

The secondary containment is assumed to fail if the secondary containment design conditions (pressure and temperature) are exceeded. Because the proposed changes do not increase the challenge to the primary containment and no new primary containment bypass sequences are created, there is no change in the challenge to the secondary containment. Therefore, the secondary containment will continue to perform acceptably after implementation of the proposed changes.

- Defenses against human errors are preserved.

  As described in this LTR, the proposed exemption provides the opportunity to reduce the operator challenges. Some operator actions are eliminated (e.g., automatic alignment for SPC), while some new operator actions are added (e.g., manually starting a LPCS pump). However, these are routine operator actions, with reasonable mission times. Thus, the probability of human error is not changed from present.

- The intent of the GDC in Appendix A to 10 CFR Part 50 is maintained.

  The GDCs establish minimum requirements for the principal design criteria for light water reactors. The principal design criteria are intended to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

  The proposed exemption is to GDC 35, which has been interpreted as requiring consideration of a concurrent LBLOCA/LOOP. However, the risk assessment in Section 4.0 demonstrates a better balance of plant safety is achievable if the design features that are focused on mitigating this event sequence are relaxed.

As described in Section 5.0, with implementation of the changes proposed in Section 3.0, reasonable assurance of the capability of the plant to mitigate LBLOCA accidents with a concurrent LOOP continues to be provided. This provides an additional layer of assurance that making these changes to a BWR is inherently consistent with the philosophy of defense-in-depth.

8.0 Safety Margins

8.1 Safety Margins

Regulatory Guide 1.174 (Reference 4, Section 2.2.1.2) – *Safety Margins* provides the guidance for engineering evaluations relative to changes in a plant licensing basis. The guidance states:

> *The engineering evaluation should assess whether the impact of the proposed licensing basis change is consistent with the principle that sufficient safety margins are maintained. Here also, the licensee is expected to choose the method of engineering analysis appropriate for evaluating whether sufficient safety margins would be maintained if the proposed licensing basis change were implemented. An acceptable set of guidelines for making that assessment is summarized below. Other equivalent acceptance guidelines may also be used. With sufficient safety margins:*
>
> - *Codes and standards or their alternatives approved for use by the NRC are met.*
> - *Safety analysis acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) are met, or proposed revisions provide sufficient margin to account for analysis and data uncertainty.*

8.2 BWROG Approach

Safety margin for this discussion must be defined in the context of the deterministic safety analysis process. In the deterministic safety analysis process, the assumed plant events are analyzed using NRC approved methods. The results of these event analyses are compared to the applicable acceptance criteria. In the deterministic analysis process, the safety margin is inherent in the selection of the acceptance criteria and the conservatism in the approved methodology.

The acceptance criteria are selected such that that there is considerable margin to the expected barrier failure. Generally, the safety margin applied is incrementally proportional to the expected frequency of occurrence. That is, the more likely the event occurrence, the more safety margin is applied. The acceptance criteria used in the safety analysis process are identified in the plant's licensing basis and generally cannot be changed without prior NRC approval. The conservatism in an event's safety analysis methodology is consistent with the category/type (e.g., anticipated operational occurrence (AOO) or Design Basis Accident (DBA)).

8.3 Safety Margin Assessment

The exemption request proposes the relocation of the assumption of a concurrent LBLOCA/LOOP from DBA category to the "severe accident" category. Thus, consistent with the above philosophy for such a change in event frequency, the incremental safety margin can be lower and less conservatism in the methodology can be utilized for a severe accident than for a DBA.

In the thermal-hydraulic analysis of the Option 3 changes, the acceptance criterion to be applied for a severe accident was chosen to be retaining the fuel in a "coolable geometry." Because defining a specific set of conditions that characterize a "coolable geometry" for the fuel is problematic, a surrogate criterion using the current acceptance criteria of §50.46 was proposed; specifically, 2200 °F for peak clad temperature and 17% local clad oxidation fraction. Use of these surrogates inherently provides safety margin in the final results.

In addition, a "best estimate" thermal-hydraulic methodology was used instead of that required by the regulations (§50.46 and 10 CFR Part 50, App. K) for design basis accidents. Again, this is in keeping with the lower frequency event category of a severe accident. While a "best estimate" model was used, it contains sufficient conservatisms, as described in App. B of this report, to provide for the uncertainties in the analysis.

Application of the above approach ensures that sufficient safety margins are preserved with the implementation of the proposed changes, consistent with the RG 1.174 guidelines.

8.4 Margins Contained in the PRA Analysis

As deemed appropriate, the risk analysis in Section 4.0 (and App. C) used bounding values, rather than more-realistic values. This was done to simplify the analysis and preclude lengthy uncertainty analyses.

One such bounding assumption is the frequency of a LBLOCA. When estimating the risk increase of eliminating the concurrent LOOP requirement, the frequency of a LBLOCA was assumed to be approximately 1×10^{-4} per year. This value is consistent with the published frequency for the total of all LOCA events that cause rapid depressurization of the reactor vessel. However, this LBLOCA frequency includes breaks in other piping systems, such as Main Steam, and intermediate-to-large break sizes that also rapidly depressurize the reactor vessel. These LOCA events are not adversely impacted by the changes proposed in Section 3.0. The LBLOCA event described in this report only occurs in the reactor recirculation piping and is greater than or equal to 10 inches in diameter in size. In the BWR Large LOCA Taxonomy (Reference 8), the fraction of LBLOCA events that meet this criterion is 0.18. This provides a considerable margin (over 80%) in the assumed risk increase (1×10^{-4} per year) used in the Section 4.0 assessment. Thus, the assumed probability for the LBLOCA event used in Section 4.0 is a bounding value that includes considerable margin.

Based on this assessment, it is concluded that sufficient safety margins are maintained.

9.0 Implementation Guide

The LBLOCA/LOOP combination is not completely removed from the licensing basis with the proposed change in regulation, but is redefined from a design basis event to a severe accident, for which some mitigation capability must be assured. The thermal-hydraulic analyses in Appendix B of this report demonstrate this mitigation capability. Also, it must be acknowledged that continued compliance with 10 CFR50.46 and all other relevant regulations must be demonstrated after any plant changes to the revised design basis (LBLOCA with offsite power available, and Small Break LOCAs, both with and without LOOP). The need to perform plant specific evaluations to assure continued compliance with the plant design basis is discussed in sub-section 9.3.

Because this report contains a bounding best-estimate analysis of the capability to mitigate LBLOCA with LOOP events, a plant would not need to provide plant specific thermal-hydraulic calculations in its application if it meets the assumptions specified under sub-section 9.2 below. Similarly, an applicant would not be required to provide plant specific PRA analyses if it meets the PRA assumptions and attributes enumerated in sub-section 9.1 below.

9.1 PRA Assumptions

The generic PRA model was adjusted to encompass some plant specific variability in terms of individual plant attributes and data ranges. However, the generic PRA model results are dependent upon a set of key assumptions that were made in performing the analysis of CDF and LERF effects of the Option 3 plant modifications. These assumptions are inherent in the event tree and fault tree construction and, if not valid for a particular plant, could alter the risk balance conclusions that resulted from the generic analysis documented in this report.

More than 70 model attributes and assumptions were assessed for their effect on the conclusions of this report. These analyses are described in Appendix C. Most of the attributes and assumptions do not have any significant effect; however some are important for the changes described in this report. To confirm that the analyses described in this report are bounding for an individual plant, it would be necessary for each plant requesting a LOCA/LOOP exemption to review each of these and to confirm that they apply on a plant specific basis.

1. HPCI (if present) is AC independent. This does not apply to any room cooling that may be required for HPCI long-term operation.
2. HPCI (if present) can be used to depressurize the reactor without uncovering the core in non-ATWS sequences. Note: HPCI is not applicable to LBLOCA events.
3. RCIC (if present) does not require room cooling for the time period considered.
4. In a BWR 5/6, one division of AC powers LPCS and one loop of LPCI, and the other division powers two loops of LPCI.
5. Plant must have the ability to provide long-term core cooling with systems that take their suction from outside the containment. For example, either LPCI or LPCS must be able to take suction from the CST (or other source of water external to containment) for long-term cooling.

6. One LPCI pump can be used for both long-term injection and for suppression pool cooling through the use of operator actions.
7. Service water can be used for injection into the reactor vessel for long-term cooling. This function must be available using either onsite power alone or offsite power alone.
8. MSIVs isolate on low water level, low steam line pressure, high steam tunnel temperature, high main steam line flow, and low condenser vacuum. Loss of offsite power also results in closure of the MSIVs.
9. Containment venting must be possible and procedures available.
10. RPS reliability must be consistent with INEEL/EXT-98-00670, General Electric RPS Unavailability, 1984-1995.
11. Each division of DC power has a dedicated charger. There is a shared "swing" charger for each DC system.
12. Plants need to review the success criteria listed in Appendix C for applicability. See Appendix C for additional guidance on this review.
13. Using information from existing pipe monitoring programs, plants need to confirm that there is no evidence of recirculation piping degradation that would result in a LBLOCA frequency in those sections significantly greater than $1x10^{-4}$.
14. Plant must demonstrate actions and programs consistent with SOER 99-01 recommendations.
15. Plants must review their fire areas. If a fire in a single area, other than the main control room, can cause both a loss of offsite power and disable a portion of the LPCI or LPCS systems, a plant specific fire risk analysis must be performed in order to make the changes to either of these systems.
16. Plants must review their flood areas. If a flood in a single area can cause both a loss of offsite power and disable a portion of the LPCI or LPCS systems, a plant specific flood risk analysis must be performed in order to make the changes to either of these systems.
17. Plants must review the seismic ruggedness of their recirculation loops and EDGs. If the ruggedness of the EDGs is greater than that of the recirculation loops, a plant specific seismic risk analysis must be performed.

9.2 Thermal-Hydraulic Assumptions

The generic thermal-hydraulic analyses performed in the scoping and sensitivity analyses in this LTR evaluated the impact of plant variability with respect to RCS volume, available ECCS injection systems, ECCS injection flow rate and the timing of ECCS injection.

The assumptions in the table below need to be validated by each applicant to ensure that the analyses in this report are applicable. The RPV volume variability sensitivity analyses encompassed a wide enough range to preclude the need for further validation.

The minimum system availability requirements are needed to preserve the defense-in-depth. In this way, a LBLOCA with an assumed LOOP must have at least this minimum set of ECCS available for injection within the assumed time delay interval, in order to conform to the assumptions of the analyses performed in Appendix B.

The ECCS injection flows shown reflect the lower end of the variability analysis range. The maximum power levels used in the analysis are provided in the table below. Plants with lower power levels can accommodate a lower ECCS injection flow.

Item #	Item	Assumption
1	BWR 2/3/4 ECC System Availability Requirements (Minimum system combinations available after time delay for a LBLOCA with a LOOP)	2 LPCI or 2 LPCS or 1 LPCI + 1 LPCS
2	BWR 5/6 ECC System Availability Requirements (Minimum system combinations available after time delay for a LBLOCA with a LOOP)	2 LPCI or 1 LPCI + 1 HPCS or 1 LPCS + 1 HPCS
3	BWR 2/3/4 EDG Time Delay	≤ 90 seconds
4	BWR 5/6 EDG Time Delay	≤ 120 seconds
5	BWR 3/4 ECCS Injection (Minimum **LPCI** Flow Requirements)	RCS P (psia) / LPCI Flow (ft^3/hr) 210.7 / 0.0 191.1 / 34184.7 171.5 / 48347.1 132.3 / 68370.3 93.1 / 83730.6 53.9 / 96678.0 34.3 / 102555.0 0.0 / 108108.0
6	BWR 2/3/4 ECCS Injection (Minimum **LPCS** Flow Requirements)	RCS P (psia) / LPCS Flow (ft^3/hr) 279.7 / 0.0 253.7 / 12350.7 226.7 / 17468.1 200.7 / 21394.8 173.7 / 24707.7 147.7 / 27623.7 127.7 / 29594.7 14.7 / 39065.4
7	BWR 5/6 ECCS Injection (Minimum **LPCI** Flow Requirements)	RCS P (psia) / LPCI Flow (ft^3/hr) 323.7 / 0.00 313.7 / 8698.59 275.5 / 21764.43 246.2 / 32700.78 194.7 / 45261.36 156.6 / 52516.17 131.6 / 56414.34 113.6 / 59265.72

8	BWR 5/6 ECCS Injection (Minimum HPCS Flow Requirements)	RCS P (psia) / LPCI Flow (ft^3/hr) 1387.0 / 0.00 1386.0 / 4330.80 1212.0 / 14436.00 1104.0 / 21654.00 996.0 / 28872.00 779.0 / 36090.00 541.0 / 43308.00 271.0 / 50526.00
9	BWR Vessel Diameter	218 to 251 inches
10	Peak Linear Heat Generation Rate	12.2 kw/ft
11	Reactor Power	BWR4: ≤3042 MWt BWR6: ≤4510 MWt

9.3 Required Plant Specific Evaluations

When a plant implements one or more of the changes contained in this report, it must identify any of its remaining licensing basis analyses that may be affected by those changes and submit them in conjunction with the exemption. For example, to demonstrate continued compliance with 10 CFR 50.46, the accident analyses including all changes must bound all small break LOCA events with a concurrent LOOP and a single active failure postulated. The analyses must also bound the double ended break of the largest primary system pipe with an additional single failure and offsite power available.

In addition other licensing basis considerations such as ATWS, Appendix R, etc. must be reviewed as required by 10 CFR 50.59.

10.0 CONCLUSIONS

To evaluate the effect of eliminating the LBLOCA/LOOP requirement, a generic probabilistic risk analysis (PRA) was performed in conjunction with a best estimate thermal hydraulic assessment. The evaluation identified a number of potential design changes and considered compliance with current regulations, defense-in-depth, and safety margins.

The results of the risk analysis showed that the probability of a large break LOCA in combination with a LOOP is consistent with the Regulatory Guide 1.174 (Reference 4) threshold for such regulatory changes (i.e., less than 1×10^{-6}) and that there was an overall risk balance between the risk increase associated with eliminating the LBLOCA/LOOP requirement and the risk reduction associated with the beneficial design changes.

The TRACG02 best estimate thermal hydraulic analysis showed that even if a LBLOCA event combined with a LOOP were to occur, with the associated ECCS injection delays and flow reductions, the core would be maintained in a coolable geometry and that 10 CFR 50.46 acceptance criteria would be met.

As a result, it is concluded that the elimination of the LBLOCA/LOOP requirement and the implementation of the identified design changes will be in compliance with existing regulations, maintain adequate defense-in-depth and preserve safety margins.

The changes in CDF and LERF from eliminating LBLOCA/LOOP requirements are acceptable and within the bounds defined by the Option 3 framework and Regulatory Guide 1.174. For some BWRs, these changes will result in a net decrease in risk to the public.

11.0 References

1. SECY-98-300, USNRC, "Options for Risk-Informed Revisions to 10 CFR Part 50 – 'Domestic Licensing of Production and Utilization Facilities'," December 23, 1998.

2 Memorandum to W. D. Travers (EDO) from A. L. Vietti (Secretary), "Staff Requirements-SECY-02-0057-Update to SECY-01-0133 'Fourth Status Report on Study of Risk Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria)" March 31, 2003.

3 Memorandum from Ashok C. Thadani, Director, Office of Nuclear Regulatory Research to Samuel J Collins, Director, Office of Nuclear Reactor Regulation, "Transmittal of Technical Work to Support Possible Rulemaking on a Risk-Informed Alternative to 10 CFR 50.46/GDC 35", July 31, 2002.

4 Regulatory Guide 1.174, USNRC, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002.

5 Nuclear Energy Institute, "Regulatory Assessment Performance Indicator Guidelines", Proposed Revision to NEI-99-02, 8/13/2002.

6 Nuclear Energy Institute, "Probabilistic Risk Assessment Peer Review Process Guidance," NEI-00-02, Revision A3, March 20, 2000.

7 NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995", (INEEL/EXT-98-00401), February 1999.

8 EPRI Report 1009044, "Boiling Water Reactor Large Loss of Coolant Taxonomy", Final Report, September 2003.

9 NEDC-32958, "Predicting Capability and Stroke Time in DC Motor-Operated Valves", Revision 0, dated March 2000 (MPR Report-2093 Revision 0)]

10 "Probability of LOOP Given Large LOCA, Results of Expert Elicitation Meeting," EPRI, Palo Alto, March 20, 2002.

11 NEI 99-02, "Mitigating Systems Performance Index," Draft 0, September 3, 2002.

12 Cooper Nuclear Station 1998 Level II PSA Update, Nebraska Public Power District, 1998.

APPENDIX A –

ECCS SYSTEM SCHEMATICS

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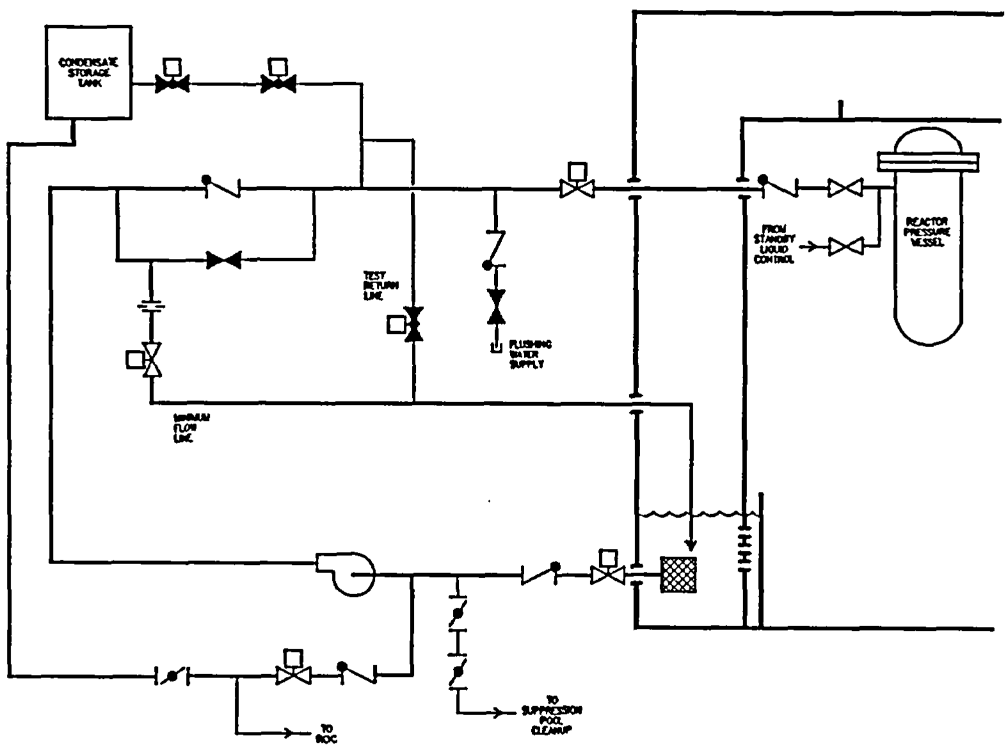

Figure A-1
High Pressure Core Spray Simplified Schematic

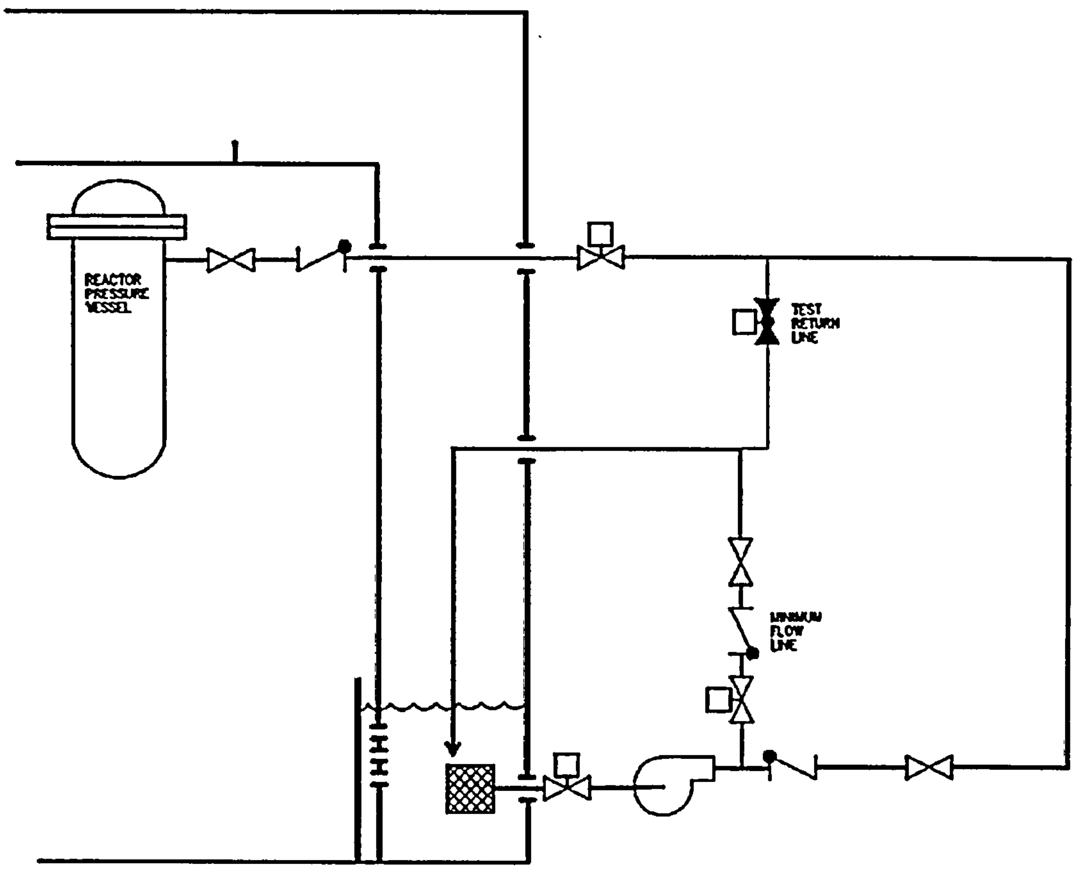

Figure A-2
Low Pressure Coolant Injection Schematic (One loop shown)

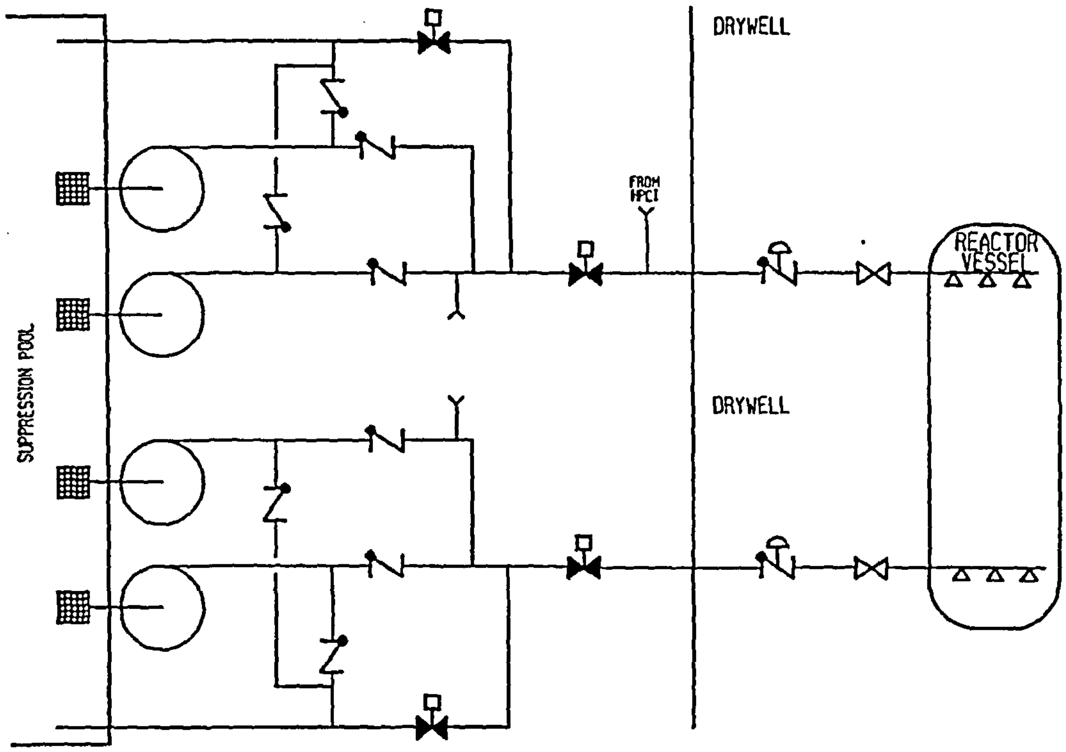

Figure A-3
Core Spray System Simplified Schematic

APPENDIX B –

THERMAL-HYDRAULIC ASSESSMENT OF OPTION 3

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B.1 PURPOSE

As outlined in Sections 2 and 7 of this report, the defense-in-depth philosophy has traditionally been applied in reactor design and operation to provide multiple means to accomplish safety functions and prevent the release of radioactive material. In order to demonstrate defense-in depth for the LBLOCA/LOOP separation exemption, per RG 1.174, this Appendix will demonstrate that the LBLOCA/LOOP event can continue to be mitigated, given the effect of the plant modifications discussed in Section 3.0 (Option 3 plant changes).

B.2 INTRODUCTION

Plant changes associated with implementation of Option 3 will result in a delay in the start of ECCS injection. They will also result in converting some equipment from automatic to manual operation and vice versa. Therefore, the impact of these plant changes was evaluated to determine the viability of implementing the changes. Given the low probability of a LBLOCA occurring concurrent with the loss of offsite power (LOOP), this event will now be considered to be a "severe accident." Consistent with this re-categorization, this evaluation was performed using best estimate assumptions and methods. Consistent with the use of best estimate methods and assumptions, the acceptance criterion for the analyses was established to maintain the core in the reactor vessel in a coolable geometry.

As discussed earlier, some measure of mitigation capability should be retained for the new beyond design basis event of the LBLOCA/LOOP, assuming the other plant changes were implemented. A classical definition of defense in depth for fission product barriers was chosen for simplicity. Because this is now a severe accident, some level of core damage could be tolerated. In addition, by definition, a LOCA is a breach of the fission product barrier of the reactor coolant pressure boundary. Thus, to ensure that the next fission product barrier of the containment was not challenged by this new event, it was decided that retaining the core material inside the reactor vessel would be used as the acceptance criterion for mitigation, i.e., maintaining the fuel in a "coolable geometry" was chosen to ensure that the core material was retained in the vessel. As an extra conservatism (i.e., to show "safety margins" as specified in RG 1.174), the existing 10 CFR 50.46 acceptance criteria of 2200 °F peak cladding temperature (PCT) and 17% local cladding oxidation fraction were chosen as the figures of merit for demonstrating that a "coolable geometry" is maintained.

The TRACG02 computer code was used to evaluate the dynamic response of best estimate cases for evaluation against the acceptance criterion. A series of scoping analyses were performed with MAAP4 to better focus the TRACG02 analysis effort. To understand the impact of code differences, a series of benchmark analyses were performed. However, the TRACG02 and MAAP4 PCT values compared reasonably well.

Additional MAAP4 cases were analyzed to assess the impact of plant-to-plant variability on the results of the analysis. It was demonstrated that changes in RPV liquid volume of ±20% and a variation in ECCS injection flow of 10% did not generate significant changes in PCT that would alter the conclusions of this evaluation.

This Appendix provides the details of this evaluation, which was summarized in Section 5 of this report.

B.3 THERMAL-HYDRAULIC ANALYSIS OVERVIEW

B.3.1 IMPORTANT LOCA ANALYSIS CONSIDERATIONS

This section identifies the considerations that are important to simulate properly in order to predict the fuel response during a LBLOCA. Each of the items listed below involve one or more phenomena that are important in the prediction of the PCT and cladding oxidation.

The aspects of a LOCA analysis that are important in predicting the timing and magnitude of the peak cladding temperature are:

1. Sources of heat
2. Blowdown heat transfer
3. Post-blowdown heat-up rate
4. Core reflood time
5. Core reflood heat transfer

Any two codes used to perform a LBLOCA analyses that agree with respect to these considerations will predict similar PCT values.

B.3.2 OVERVIEW OF COMPUTER CODES

The LBLOCA analyses discussed in this Appendix were performed with the TRACG02A (referred to as TRACG02) and MAAP 4.0.4 (referred to as MAAP4) computer codes.

A brief overview of each of these codes is provided below.

B.3.2.1 TRACG02 Overview

TRACG is a computer code for the prediction of boiling water reactor transients ranging from abnormal operational occurrences (AOOs) to design basis loss-of-coolant accidents, stability and anticipated transients without scram (ATWS). TRACG incorporates a two-fluid thermal-hydraulic model for the reactor vessel and the primary coolant system, a three-dimensional kinetics model for the reactor core and one-dimensional calculations in the other components.

The conservation equations for mass, momentum and energy are closed through an extensive set of basic models consisting of constitutive correlations for shear and heat transfer at the gas/liquid interface as well as at the wall. The constitutive correlations are flow regime dependent, and are determined based on a single-flow regime map, which is used consistently throughout the code. In addition to the basic thermal-hydraulic models, TRACG contains a set of component models for BWR components, such as recirculation pumps, jet pumps, fuel channels, steam separators and dryers. TRACG also contains a control system model capable of simulating the major BWR control systems such as the pressure, level and recirculation flow control systems.

TRACG02 is a "best estimate" system computer code applicable for analysis of loss-of-coolant accidents (LOCAs) including large and small breaks at any location.

While the application range of TRACG02 includes LOCA analysis, a detailed application methodology for LOCA has not been defined for TRACG02, as it has for evaluating Abnormal Operating Occurrences (AOOs), including Anticipated Transients Without Scram (ATWS). GE is part way through the development of a LOCA application methodology, and an estimated TRACG02 modeling uncertainty adder, based upon benchmarking with other LOCA evaluation models and testing data, is applied to the TRACG02 results.

B.3.2.2 MAAP4 Overview

MAAP 4.0.4 is an EPRI tool that is used primarily for the modeling of PRA event sequences in order to evaluate the sequence timing, to determine system success criteria and to identify if a sequence results in a core damage state. There is a BWR and a PWR version. MAAP4 includes models for the important accident phenomena that might occur within the primary system, containment and reactor/auxiliary building. MAAP4 calculates the progression of the postulated accident sequence, including the disposition of the fission products, from a set of initiating events to either a safe, stable state or to an impaired containment condition and the possible release of fission products to the environment. MAAP4 has modeling capabilities for the determination of the PCT during a LBLOCA.

The MAAP4 methodology includes the following models, which are most pertinent to the analysis of a Large Break LOCA event:

a) General Core Model

b) Break Flow Model

c) Two Phase Mixture Level

d) Overheat and Clad Oxidation of Uncovered Core Nodes

   1. Core Heat Up

   2. Metal-Water Reaction

e) Quenching for Reflooded Core Nodes

MAAP4 has the advantage of being easier and faster to execute that some other codes, which allows for the analysis of a variety of cases in a relatively short period of time. MAAP4 is used in this report to perform a scoping analysis of the various Option 3 changes to identify the limiting cases to be analyzed with TRACG02. It is also used to perform a sensitivity analysis to evaluate the expected PCT impact of the identified key plant specific variables of Reactor Coolant System (RCS) volume and ECCS flow rate to provide a justification for applicability of the results among the various plants in the BWR fleet.

B.3.3 TECHNICAL APPROACH

As outlined in Section 5 of the report, the approach used to evaluate the impact of the proposed Option 3 changes was as follows:

- Benchmark LBLOCA cases were performed with both the MAAP4 and TRACG02. The key phenomena important to a LBLOCA analysis were compared between the two codes and adjustments to the inputs and assumptions used in MAPP4 were made to achieve general agreement. The benchmark analyses are discussed in Section B.4.

- Base cases were identified which included the equipment and plant variations which we expect to bound the plant variability within the BWR fleet as follows:
    - ❖ A series of MAAP4 scoping cases was performed using the various equipment combinations remaining after implementing all of the Option 3 changes. This method was used to identify the limiting cases for further TRACG analysis and is discussed in Section B.5.
    - ❖ A series of MAAP4 sensitivity cases to evaluate the effect of plant-to-plant variability were performed. The sensitivity analysis is discussed in Section B.6.
- The MAAP4 scoping case with the highest PCT for the BWR4 and one for the BWR6 were then run using TRACG02 to generate the actual PCTs used to demonstrate defense-in-depth with implementation of Option 3. The BWR6 case selected did not have the highest PCT. It was chosen because it offered the opportunity to evaluate the impact of the Option 3 changes in scenarios requiring both high pressure and low pressure injection sources. The difference in peak PCT between the BWR6 case selected and the BWR6 case with the highest PCT is less than 100 °F and there is significant margin to the 2200 °F acceptance criteria. The final TRACG02 analyses are provided in Section B.7.

B.4 MAAP4/TRACG02 BENCHMARK

B.4.1 Introduction

Key parameters from the final TRACG02 analyses (refer to Section B.7) and the MAAP4 cases selected for final analysis (refer to Section B.5) are compared in this section. Transient plots comparing these data are included as Figures B.4-1 to B.4-5 for the BWR4 and Figures B.4-6 to B.4-10 for the BWR6.

B.4.2 BWR4 Case A Comparison

B.4.2.1 *Overview* (Case A)

This section provides a discussion of how MAAP4 compares with TRACG02 for the evaluation of a BWR4 LBLOCA with 2 LPCI pumps injecting into the intact recirculation loop and a 90 second EDG delay time (Case A from the MAAP4 scoping analyses summarized in Table B.5-2). The comparison is provided for reactor power, reactor vessel pressure, break flow rate, ECCS flow rate and peak cladding temperature. Table B.4-1 below provides a comparison of the timing of key events in both the MAAP4 and TRACG02 analyses.

As can be seen, the Level 1 setpoint is reached slightly sooner in MAAP4. This is due to a somewhat faster drop in level in MAAP4. This would result in an earlier LPCI injection time by about 3 seconds. Another difference is that TRACG02 accounts for the refill of the recirculation loop before directing any LPCI flow into the vessel, which takes less than 20 seconds with two LPCI pumps injecting into the intact loop. In order to account for this in MAAP4, the LPCI injection valve delay time of 37 seconds was increased to 57 seconds. The pressure response between the codes is very similar. Further discussion of the reactor vessel pressure comparison is provided later in this section.

In order to obtain the same adiabatic heatup rate in both TRACG02 and MAAP4, the heat generation in the "hottest node" must match. In the TRACG02 cases four limiting bundles are modeled, each with different limiting fuel parameters (either Critical Power Ratio (CPR) or Linear Heat Generation Rate (LHGR) on limits, and top or bottom peaked power shapes). The remainder of the core has a nominal, center peaked power shape. In the MAAP4 model a center fuel ring was added that matches the power shape in the TRACG02 bundle with the highest LHGR for any node and is top peaked. This fuel ring contains the equivalent of four bundles.

The MAAP4 axial power distribution for the various core rings is depicted in Figure B.4-11 below. The hot channel is top peaked with the highest core peaking factors.

TRACG02 uses a two-phase discharge model with multiple break locations to simulate a recirculation suction line large break LOCA. The environment is used as the break downstream volume to minimize the backpressure. Use of these break elements shows a close match to experimental data. A similar break nodalization scheme is implemented for the MAAP4 analysis using the default LOCA and adding generalized openings to model the jet pump break locations. A reasonably close match to the TRACG02 data was achieved.

An additional sensitivity analysis was performed that varied the break discharge coefficient as a function of time over the first 20 seconds in an attempt to better match the liquid and vapor mass and energy release rates. Approximately the same total break flow was achieved for the first 20 seconds of the analysis. However, since there was no significant change in the PCT, the MAAP4 default break discharge coefficient of 0.7 was used for the final benchmark analysis.

Table B.4-1
TIMING OF KEY EVENTS IN THE BWR4 MAAP -TRACG COMPARISON

EVENT	MAAP (sec)	TRACG (sec)
Feed water Pump Trip	0.0	0.00
Reactor Scram	0.5	0.0
Level 1	4.6	7.76
MSIV initiation (4 sec stroke)	5.2	0.5
ECCS Actuation	151.6	134.8

B.4.2.2 Reactor Power (Case A)

The reactor power comparison is provided in Figure B.4-1. As can be seen in the figure, the power response between the two codes is very similar. While the scram time is slightly sooner in TRACG02, the overall impact on the reactor power is not significant.

B.4.2.3 Reactor Vessel Pressure (Case A)

The reactor pressure comparison is provided in Figure B.4-2. The pressure comparison between the two codes is quite good. MAAP4 exhibits a small pressure increase at the beginning of core reflood, which is not exhibited by TRACG02. This small pressure spike causes a reduction in the LPCI injection flow over a period of about 50 seconds, but overall the pressure effect does not have a significant impact on the results of the analysis. The cause of the pressure spike is associated with a rapid energy transfer that occurs when the core begins to reflood.

B.4.2.4 Break Flow Rate (Case A)

Since the reactor pressures matched closely, it would be expected that the break flow rate would also be a close comparison. The break flow rate comparison is shown in Figure B.4-3 and shows a very close similarity. In order to assure that this parameter matched fairly closely, the MAAP4 liquid region and steam region break areas were increased slightly to provide a closer approximation to the TRACG02 break flow rate.

B.4.2.5 ECCS FLOW (Case A)

Figure B.4-4 shows the LPCI flow into the intact loop. The flow rates are prescriptive functions of vessel pressure and are the same between the codes. In MAAP4, ECCS flow begins 20 seconds later than in TRACG02. This delay was added to the MAAP4 input to simulate the refill of the intact recirculation loop. This phenomenon is explicitly calculated in TRACG02 and not in MAAP4. The reduction in flow in the MAAP4 case at about 200 seconds is due to the pressure increase during reflood that was described in Section B.4.2.3.

B.4.2.6 Peak Cladding Temperature (Case A)

Overall, with the changes made to the MAAP analysis discussed above, MAAP predicts higher peak cladding temperatures than TRACG02 until the core quenching phase of the event. These results are presented in Figure B.4-5.

There are three phases in the scenario to consider. First is the PCT response during the blowdown (0 to 20 seconds). Both codes show an initial drop in clad temperature. When the recirculation suction line becomes uncovered, however, the clad temperature in MAAP4 would normally increase by several hundred degrees because there is logic in MAAP4 to change the heat transfer mode to film boiling when the core flow is reduced below 10% of the full power flow. In order to match the TRACG02 model, which is viewed as more best estimate, the film boiling heat transfer coefficient in MAAP4 was increased to 1170 btu/hr-ft^2-°F, which corresponds to the heat transfer prior to the mode change to film boiling. It is maintained until the core reflood begins, at which time the film boiling heat transfer coefficient is returned to its normal value. The result is a relatively close match for the PCT at about 20 seconds, about 500°F, when the fuel heatup begins.

In the next phase, most of the core is uncovered and there is little steam flow through the core. In this phase TRACG02 and MAAP4 PCT predictions compare rather well. The rate of heatup is similar between the codes, with MAAP4 showing about a 10% higher heatup rate.

The final phase models the quench of the core during reflood. In MAAP4, this occurs rapidly. Note that if the PCT is above 1800°F, there is a brief increase in temperature when there is steam available for metal-water reaction (See Fig B.4-5).

In the end, the PCT predicted by MAAP4 (about 2000°F) is about 240°F higher than TRACG02 (1758°F). However, from previous studies, there are known model uncertainties in the TRACG02 code that impact the prediction of the PCT during a LBLOCA event. To account for these uncertainties, a PCT adder[1] of 193°F was added to the TRACG02 BWR4 analysis PCT results. This results in a TRACG02 PCT of about 1951°F versus a MAAP4 PCT of about 2000°F. This shows that the PCT predicted by MAAP is very close, but somewhat more conservative, than the adjusted TRACG02 PCT.

[1] While the application range of TRACG02 includes LOCA analysis, a detailed application methodology for LOCA has not been defined for TRACG02, as it has for evaluating Abnormal Operating Occurrences (AOOs), including Anticipated Transients Without Scram (ATWS). GE is part way through the development of a LOCA application methodology, and an estimated TRACG02 modeling uncertainty adder, based upon benchmarking with other LOCA evaluation models and testing data, is applied to the TRACG02 results.

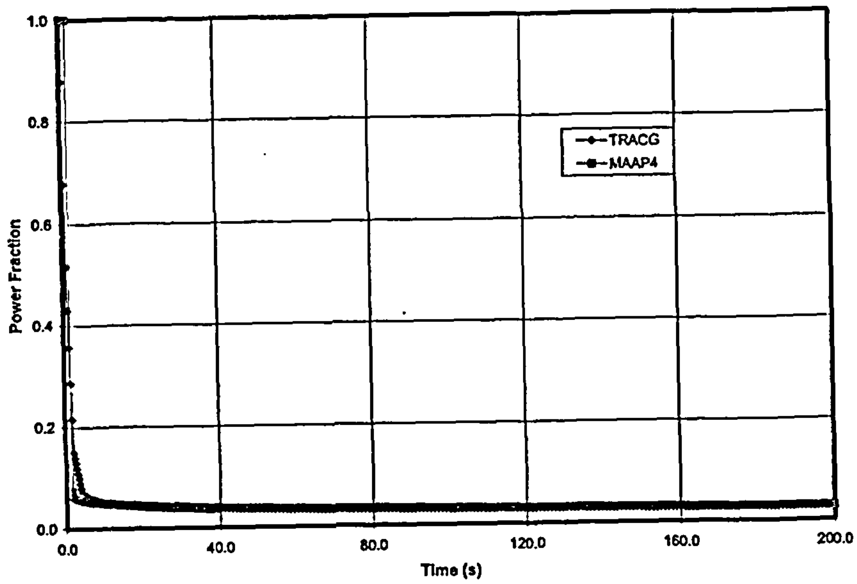


Figure B.4-1
BWR4 Case A: Reactor Power Comparison

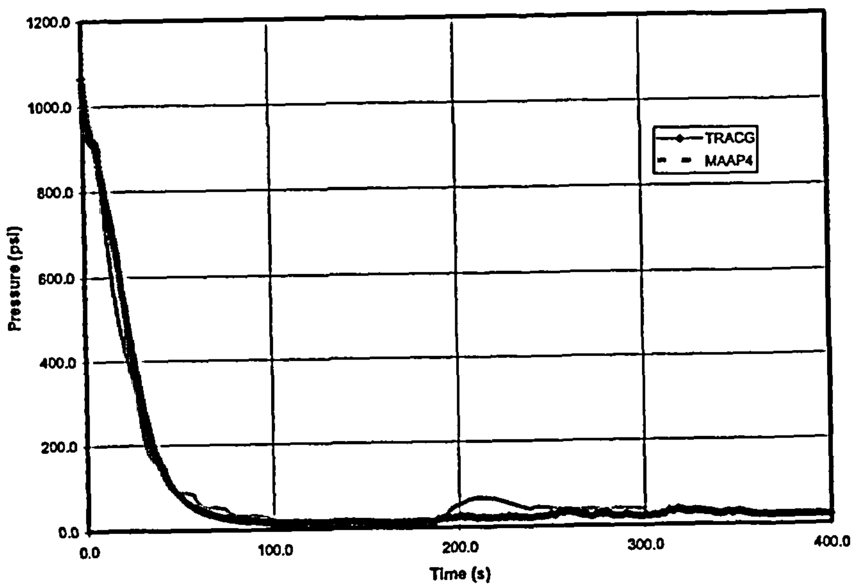


Figure B.4-2
BWR4 Case A: Reactor Vessel Pressure Comparison

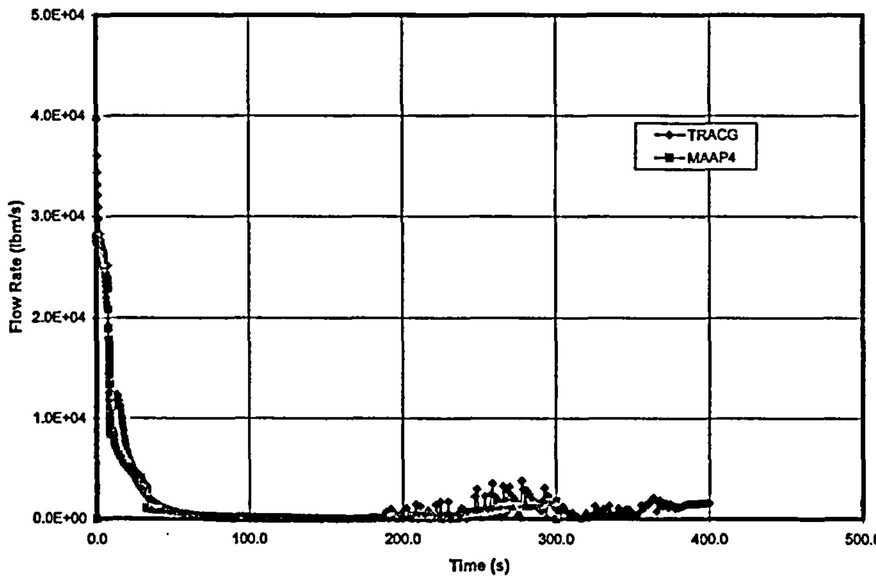


Figure B.4-3
BWR4 Case A: Break Flow Rate Comparison

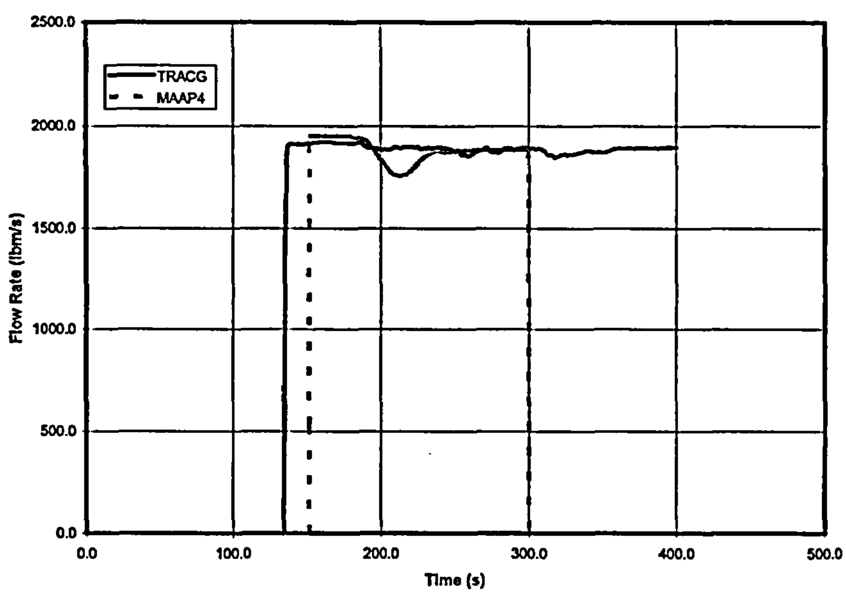


Figure B.4-4
BWR4 Case A: ECCS Flow Comparison

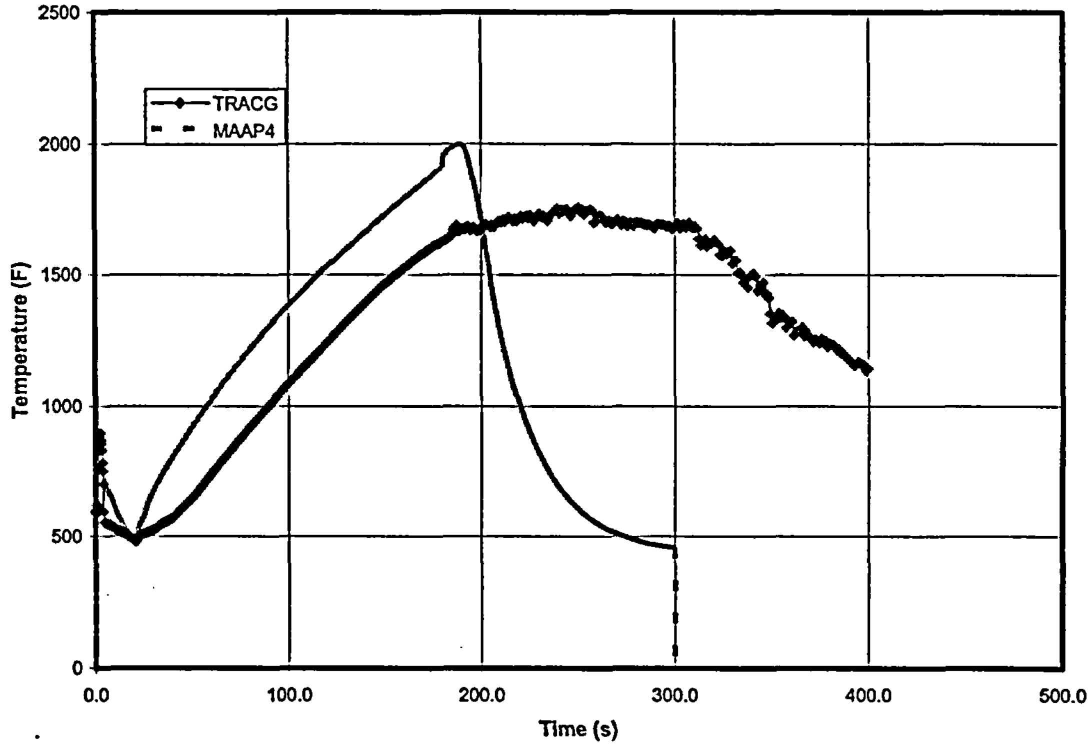


Figure B.4-5
BWR4 Case A: Peak Cladding Temperature Comparison

B.4.3 BWR6 Case C Comparison

B.4.3.1 Overview (Case C)

This section provides a discussion of how MAAP4 compares with TRACG02 for the evaluation of a BWR6 LBLOCA with 1 LPCI pump and 1 HPCS pump injecting into the core. A 120 second EDG delay time is assumed. This event is Case C from the MAAP scoping analyses summarized in Table B.5-2. The comparison is provided for reactor power, reactor vessel pressure, break flow rate, ECCS flow rate and peak cladding temperature. Table B.4-2 below provides a comparison of the timing of key events in both the MAAP and TRACG02 the analyses.

The same adjustments to the core model and break area were made for the BWR6 cases as were made for the BWR4.

The time of ECCS injection is very critical to the analysis results. MAAP depressurizes somewhat faster than TRACG02 so the LPCI pressure permissive is reached about 5.4 seconds sooner but this is not important because the time of LPCI injection is very close. The HPCS injection time for both codes is also within about 1 second. The main reason for this difference is the time to reach Level 1, which was used as the HPCS initiator for both analyses.

Table B.4-2
TIMING OF KEY EVENTS IN THE BWR6 MAAP-TRACG COMPARISON

EVENT	MAAP (sec)	TRACG (sec)
Feed water Pump Trip	0.0	0.0
Reactor Scram**	4.0	1.157
Low Level HPCS Setpoint*	7.1	7.72
Low Level LPCI Setpoint (L1)	7.1	7.72
MSIV Initiation (5 sec stroke)	7.1	7.72
LPCI Pressure Permissive	37.0	39.23
HPCS Injection Begins	154.1	154.8
LPCI Injection Begins	164.2	164.3

* TRACG02 and MAAP4 initiated HPCS on L1
** TRACG02 scram on Hi DW P.

B.4.3.2 Reactor Power (Case C)

The reactor power comparison is provided in Figure B.4-6. As can be seen in the figure, the power response between the two codes is very similar. While the scram time is slightly sooner in TRACG02, the overall impact on the reactor power is not significant.

B.4.3.3 Reactor Vessel Pressure (Case C)

The reactor pressure comparison is provided in Figure B.4-7. MAAP4 results in a more rapid pressure reduction during blowdown than TRACG02 for the BWR6. This difference is not critical to the analysis results. There is an increase in the MAAP4 vessel pressure at about 150 seconds, which is coincident with the time at which HPCS injects into the core. The MAAP4 code provides a much more rapid core heat transfer response to the HPCS injection than TRACG02. The result is additional steam generation that is the cause of the pressure increase. The higher core heat transfer also results in a more rapid reduction of the fuel cladding temperature, which is discussed below.

B.4.3.4 Break Flow Rate (Case C)

The break flow rate comparison for the BWR6 analyses is shown in Figure B.4-8 and shows a very close similarity between the codes. In order to assure that this parameter matched fairly closely, the MAAP4 liquid region and steam region break areas were increased slightly to provide a closer approximation to the TRACG02 break flow rate similar to what was done for the BWR4 MAAP4 analysis.

B.4.3.5 ECCS Flow (Case C)

Figure B.4-9 shows the HPCS injection at about 154 seconds followed by the LPCI injection at about 164 seconds. Because the LPCI and HPCS flows are a function of vessel pressure, the ECCS flow rates in MAAP4 are initially less than the TRACG02 flow rate because the MAAP4 vessel pressure at the time of ECCS injection is greater than the TRACG02 vessel pressure. As the pressures between the codes equalize, the MAAP4 ECCS flow rates increase to more closely match the TRACG02 ECCS flow rates.

B.4.3.6 Peak Cladding Temperature (Case C)

Overall, with the changes made to the MAAP4 analysis discussed above, MAAP predicts higher peak cladding temperatures than TRACG02 until the core quenching phase of the event. These results are presented in Figure B.4-10.

There are three phases in the scenario to consider. First is the PCT response during the blowdown (0 to 20 seconds). Both codes show an initial drop in clad temperature. When the recirculation suction line becomes uncovered, however, the clad temperature in MAAP4 would normally increase by several hundred degrees because there is logic in MAAP4 to change the heat transfer mode to film boiling when the core flow is reduced below 10% of the full power flow. In order to match the TRACG model, which is viewed as more best estimate, the film boiling heat transfer coefficient in MAAP4 was increased to 1170 btu/hr-ft^2-°F, which corresponds to the heat transfer prior to the mode change to film boiling. This condition is maintained until the core reflood begins, at which time the film boiling heat transfer coefficient is returned to its normal value. After 20 seconds, MAAP4 PCT reaches a minimum value of about 535°F and then begins to increase while the TRACG02 PCT continues to decrease until approximately 60 seconds at which time the minimum PCT is about 400°F.

In the next phase, most of the core is uncovered and there is little steam flow through the core. In this phase TRACG02 and MAAP4 PCT predictions compare rather well. The rate of heatup is virtually the same in both codes.

The final phase models the quench of the core during reflood. In MAAP4, this occurs rapidly. In this case, the PCT is fairly low so MAAP4 does not show any increase from clad oxidation during the initial reflood. In TRACG02, the initiation of core spray does not immediately quench the upper part of the core. It does, however, slow the rate of increase.

In the end, the PCT predicted by MAAP4 (1580°F) is about 158°F higher than TRACG02 (1422°F). This temperature difference is very close to the results that were seen in the BWR4 analysis. For the BWR6, this difference can be attributed in part to the temperature difference following the blowdown phase as discussed above.

As with the BWR4 results (refer to Section B.4.2.6), to account for the model uncertainties for the BWR6, a PCT adder of 212°F was added to the TRACG02 BWR6 analysis PCT results. This results in a TRACG02 PCT of about 1634°F versus a MAAP4 PCT of about 1580°F. This shows that the PCT predicted by MAAP for the BWR6 is close, but somewhat less conservative, than the adjusted TRACG02 PCT. Both codes, however, predict a significant margin to a PCT of 2200 °F.

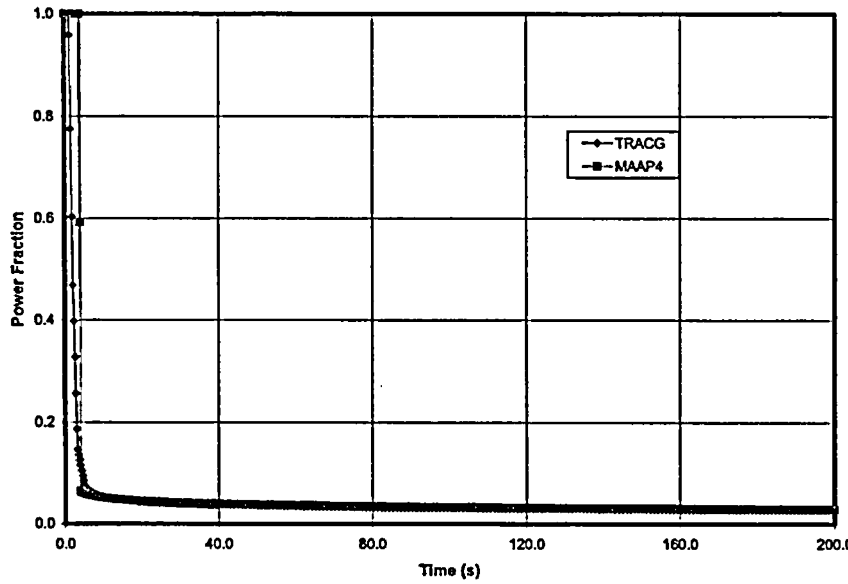


Figure B.4-6
BWR6 Case C: Reactor Power Comparison

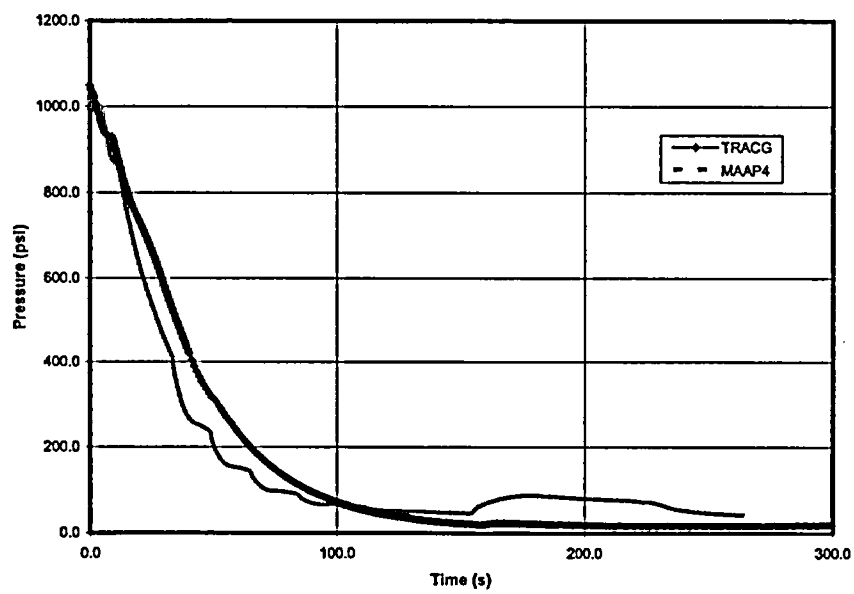


Figure B.4-7
BWR6 Case C: Reactor Vessel Pressure Comparison

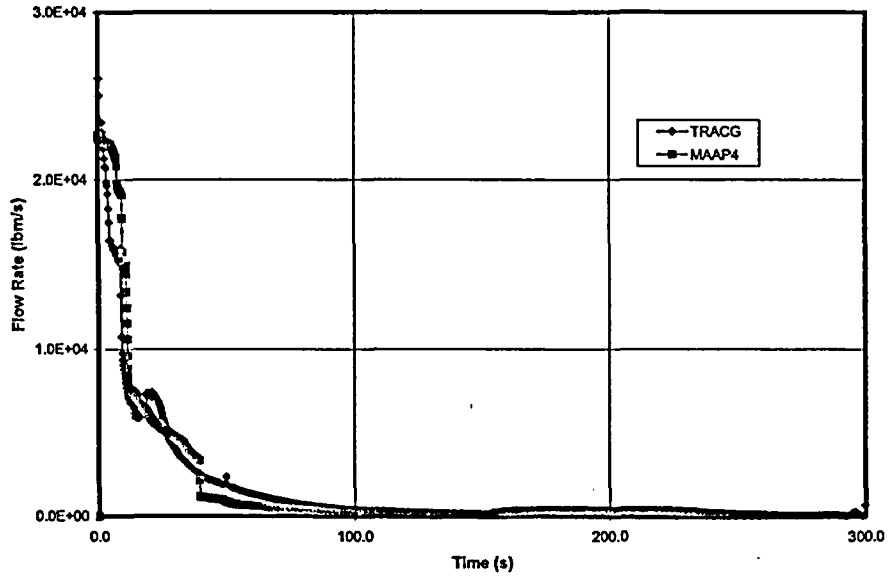


Figure B.4-8
BWR6 Case C: Break Flow Rate Comparison

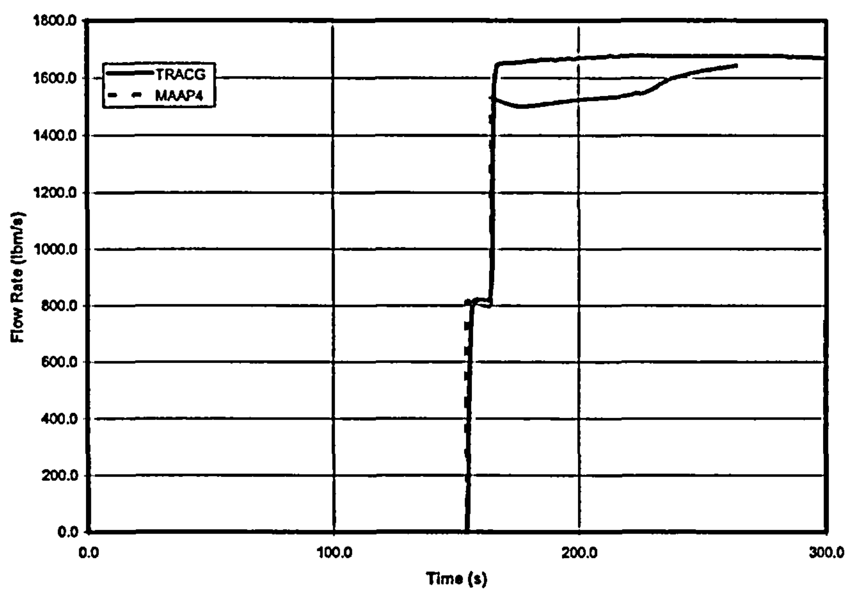


Figure B.4-9
BWR6 Case C: Total (HPCS & LPCI) ECCS Flow Comparison

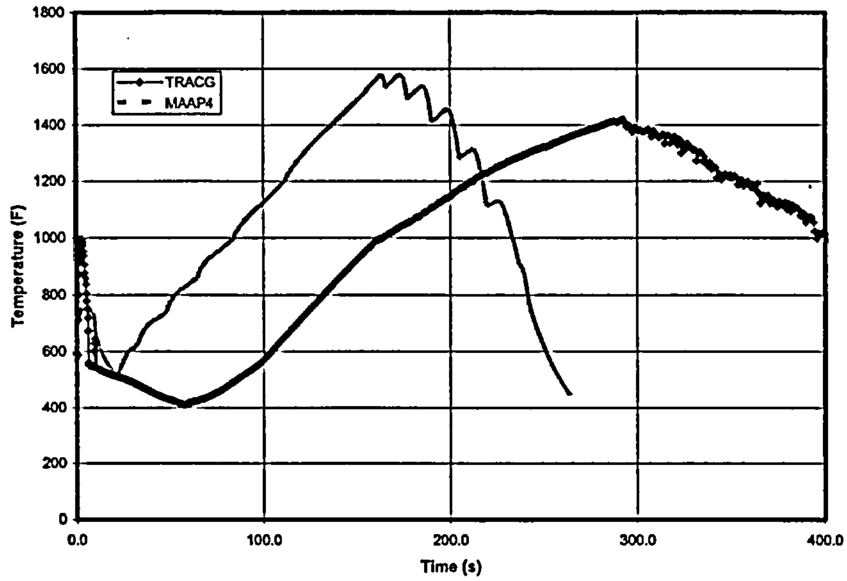


Figure B.4-10
BWR6 Case C: Peak Cladding Temperature Comparison

B.4.4 Benchmark Conclusions

There are several points of note from the MAAP4 vs. TRACG02 comparison. First, MAAP4 does not predict the same temperature drop during the blowdown as TRACG02. This response is due to the MAAP4 heat transfer mode change to film boiling when the core flow decreases to 10% of the full power flow, which occurs from 10 to 20 seconds. To compensate for this behavior the film boiling heat transfer coefficient was increased during this period, which improved the results. However, the core heatup in MAAP4 still starts from a higher temperature than in TRACG02. This difference is more pronounced for the BWR6 than the BWR4.

Another difference was noted in the time to begin vessel reflood. In MAAP4, LPCI flow is immediately directed into the lower plenum upon system actuation. The nodalization in TRACG02 forces the recirculation loop to be refilled prior to directing any LPCI flow into the jet pumps. This phenomenon causes a delay in core reflood for scenarios that involve LPCI. This phenomenon only affects plants in which LPCI is injected into the recirculation loops.

Another important set of parameters is the fuel peaking. In order to obtain the same adiabatic heatup rate in both TRACG02 and MAAP4, the heat generation in the "hottest node" must match. In the TRACG02 cases four limiting bundles are modeled, each with different parameters (CPR and LHGR limited, top and bottom peaked). The remainder of the core has a nominal, center peaked power shape. In the MAAP4 model a center fuel ring was added that matches the power shape in the TRACG02 bundle with the highest LHGR for any node and is upper peaked. This fuel ring contains the equivalent of four bundles.

The TRACG02 results should be modified to account for model uncertainties in the prediction of the PCT. To accomplish this, a PCT adder of 193°F was included for the BWR4 and an adder of 212°F was included for the BWR6.

Overall the results compare favorably given the differences in methodology between the two codes. The results of these benchmark analyses conclude that MAAP4 can be used to perform the scoping and sensitivity cases and identify the bounding cases for evaluation by TRACG02. The scoping analysis using MAAP4 is provided in Section B.5 and the TRACG final analysis of the bounding cases is provided in Section B.7.

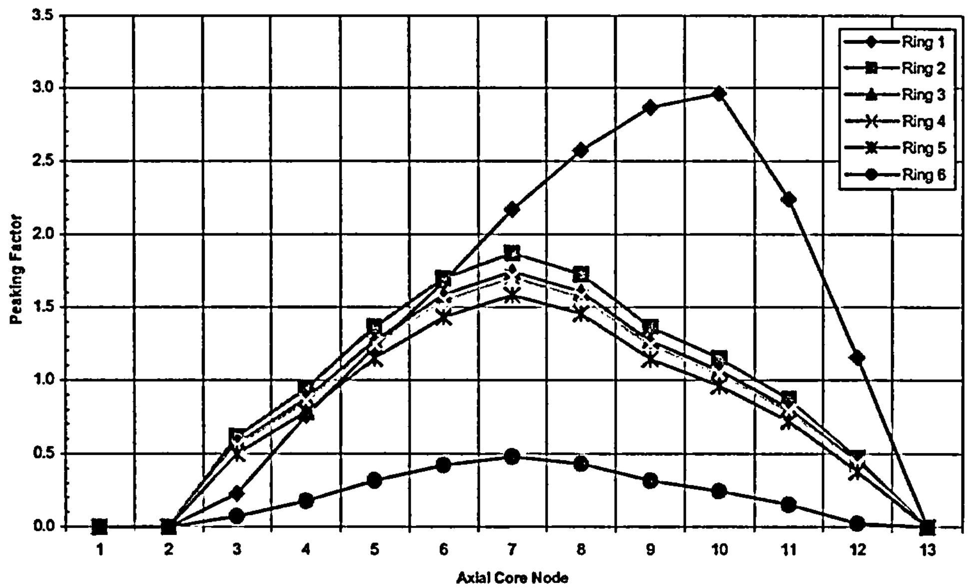


Figure B.4-11
MAAP4 Core Peaking Factors

B.5 MAAP4 SCOPING ANALYSES

B.5 1 Introduction

The purpose of this section is to evaluate the plant response using the MAAP4 computer code including timing of injection and the number of injection systems. Each of the Option 3 changes and combinations were considered. The acceptance criterion for this evaluation is that plants must retain mitigation capability with the objective of maintaining the core in the reactor vessel in a coolable geometry.

The event analyzed was the double-ended guillotine break of a recirculation loop suction line. This break generates mass and energy releases and resulting peak cladding temperatures that are bounding to smaller breaks. Representative cases were evaluated for the various BWR plant types with the objective of maintaining the core in the reactor vessel in a coolable geometry. The peak cladding temperature (PCT) was defined as the figure of merit to demonstrate that the acceptance criterion was met.

B.5.2 Evaluation of Option 3 Changes

Utilities may choose to implement the identified plant changes in Section 3 of this report as a result of separation of LOOP and large break LOCA. In order to develop a set of cases for analysis it was important to evaluate the effects on ECCS equipment operation from these proposed plant changes. The anticipated plant changes, including their likely affect on ECCS equipment, are as follows:

- <u>Eliminate diesel fast starts:</u>

  The expected diesel start time to facilitate warm-up would be 30 – 60 seconds.

- <u>Optimize diesel generator (EDG) loading:</u>

  Automatic loading of the LPCS pump(s) or two of the LPCI pumps on the diesel would be eliminated. Manual start of these pumps would be expected in cases where the EPGs require maximum injection. However, this operator action was not credited in the analysis.

- <u>Eliminate anticipatory diesel start:</u>

  A LOCA signal would no longer cause the start of the EDG. The EDG would start only on bus under voltage or degraded voltage conditions.

- <u>Simplified EDG testing:</u>

  There are no effects on ECCS equipment response during LOCA as a result of the proposed EDG testing.

- <u>Increased MOV stroke times:</u>

  Longer valve stroke times could lead to delayed injection start time.

- Automatically start one loop of RHR in suppression pool cooling (SPC) mode:

  Only one loop of RHR would be available for automatic LPCI. Availability of LPCI would depend on the break location since LPCI may not be available in the broken loop.

- Eliminate LPCI Loop Select logic:

  One loop of LPCI would inject into the broken loop. Certain scenarios could result in a loss of all LPCI.

The effects on ECCS equipment operation can be grouped into categories based on delayed or reduced injection, with a further distinction between LPCS or LPC injection. In Section 3 of this report, an EDG warm up time of 30 – 60 seconds is anticipated for the elimination of the fast start capability. In this analysis, EDG delay times of 90 seconds for the BWR 3/4 and 120 seconds for the BWR 5/6 were included to take into account either implementation of multiple changes or other phenomena, such as delayed LOOP. The difference in EDG delay is associated with the recirculation loop fill time in the BWR3/4 that is not present in the BWR6 LPCI design. Table B.5-1 shows the effects categorized and related to the BWR 2/3/4 and BWR 5/6 plant types.

B.5.3 Case Development

Based on the three (3) categories of effects identified in Table B.5-1, a series of cases was developed. Only realistic combinations of ECCS equipment availability were included. Combinations which result in a total loss of injection were not included. For instance, the combination of all three categories for a BWR 3/4 would result on a total loss of injection. The limiting combinations are listed in Table B.5-2. The following assumptions were made in developing these cases:

- Recirculation suction line break:
  The event analyzed is a double-ended guillotine rupture of the recirculation suction line. A break at this location generates bounding mass and energy release rates and vessel depressurization. These conditions require the maximum ECCS flow rates to reflood the reactor core. To insure a bounding result for the BWR 3/4 plant, the break in the recirculation loop was also assumed to disable one loop of LPCI injection.

- Loss of Offsite Power:
  A LOOP is assumed to occur concurrent with the ECCS injection signal, to simulate the delayed LOOP. The main feedwater pump trip is conservatively assumed to occur at the start of the event.

- ECCS Equipment Only:
  No credit is taken for non-ECCS injection sources.

- Power Level:
  Cases were analyzed at 105% and 125% original licensed thermal power to bound nominal and uprated plants.

- Fuel:
  All cases were analyzed with GE 14 fuel at an end-of-life core with a top peaked power shape with the limiting bundle on thermal limits.

- Negligible Initial Clad Oxidation:
  Low initial clad oxidation leads to higher clad oxidation rates during the event.

B.5.4 ECCS Injection Line LOCAs

Breaks in an ECCS injection line were also considered in this evaluation. While it was postulated that such breaks would be less severe than the design basis recirculation line break LOCA, an analysis of a BWR6 LPCI line break was performed with MAAP4. The peak cladding temperature for this case was substantially less than any of the recirculation line break LOCA cases analyzed. This combination would be applicable to all BWR plant types.

On the basis of this analysis, it can be concluded that ECCS injection line breaks are not limiting and do not require further consideration in this evaluation.

B.5.5 Results

The PCT values for the MAAP4 scoping analyses are compared in Figure B.5-1. The highest PCT case for the BWR4 was Case A and the highest PCT case for the BWR6 was Case D. These are both cases that assume 2 LPCI pumps are the only ECCS injection available. The Case C PCT for the BWR6 is not significantly different than Case D, but it provides injection with 1 LPCI and 1 HPCS. In order to account for the effects of HPCS in the BWR6 analysis, it was judged that Case C should be used as the limiting analysis case for the BWR6. Therefore, Cases A and C were selected as the limiting cases for final analysis by TRACG02. The TRACG02 final analyses are provided in Section B.7.

Table B.5-1
Bounding Representation of Option 3 Plant Changes

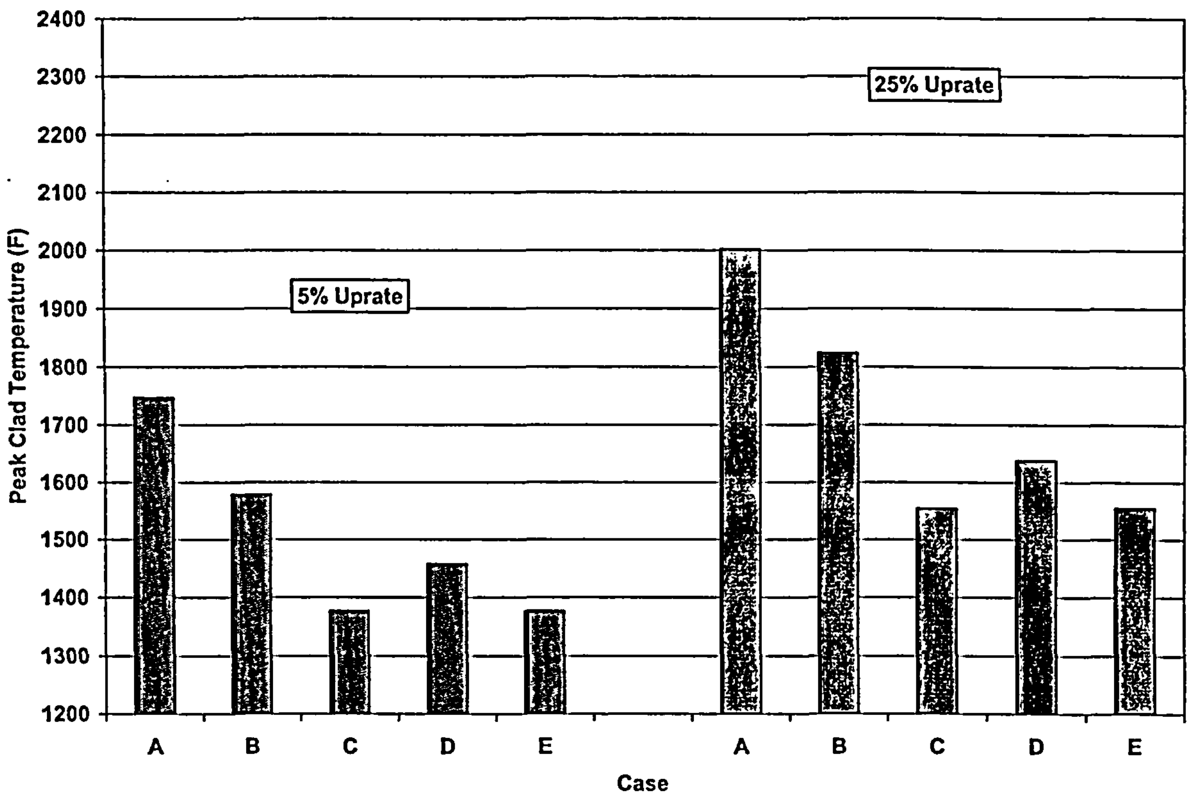
<table>
<tr><th rowspan="2">Category</th><th rowspan="2">Change</th><th rowspan="2">Effect</th><th colspan="3">Bounding Effect on ECCS</th></tr>
<tr><th>BWR 3/4</th><th>BWR 3/4 w/ Loop-Select</th><th>BWR 5/6</th></tr>
<tr><td rowspan="3">1</td><td>Eliminate EDG fast start</td><td rowspan="3">Delayed Injection</td><td colspan="2" rowspan="3">90 Second EDG Delay[1,2]</td><td rowspan="3">120 Second EDG Delay[1]</td></tr>
<tr><td>Eliminate anticipatory EDG start</td></tr>
<tr><td>Increased MOV stroke times</td></tr>
<tr><td>2a</td><td rowspan="2">Optimize EDG loading</td><td rowspan="2">Reduced Injection</td><td colspan="2">No LPCS</td><td>Loss of 1 LPCI Pump and 1 LPCS Pump</td></tr>
<tr><td>2b</td><td colspan="3">Loss of 2 LPCI Pumps</td></tr>
<tr><td>3</td><td>One loop of RHR in SPC mode</td><td>Reduced Injection</td><td colspan="2">No LPCI[2]</td><td>Loss of 1 LPCI Loop</td></tr>
<tr><td>4</td><td>Eliminate LPCI LOOP Select logic</td><td>Reduced Injection</td><td>N/A</td><td>Loss of 1 LPCI Loop</td><td>N/A</td></tr>
<tr><td>N/A</td><td>Simplified EDG testing</td><td>N/A</td><td colspan="3">N/A</td></tr>
</table>

Notes:
1. Nominal valve stroke and pump coast-up time assumed for ECCS equipment operation is included in addition to the EDG delay.
2. Bounding for a BWR 2

Table B.5-2
Limiting Combination of ECCS Delay and Reduced Injection

Case	Plant Type	Category[2] Combination	EDG Delay Time (sec)	Number of Injection Pumps		
				LPCI	LPCS	HPCS
A	BWR 3/4	1 & 2	90	2	0	N/A
B[3]		1 & 3		0	2	N/A
C	BWR 5/6	1 & 2 & 3	120	1	0	1
D				2	0	0
E				0[1]	1	1

Notes:
1. Includes an additional LPCI failure to make it a unique case.
2. Category corresponds to Table B.5-1.
3. The only combination valid for a BWR 2 plant corresponds to Case B.



Notes:

1. Refer to Table B.5-2 for a description of the Cases.
2. ECCS injection flows were not changed with power level.
3. The purpose of this plot is to show the relative differences between the cases only. The numerical magnitude of the PCT is not utilized in this analysis.

Figure B.5-1
MAAP4 PCT Comparisons of Option 3 Changes

B.6 MAAP SENSITIVITY ANALYSES

MAAP4 sensitivity cases were run to evaluate the impact of critical assumptions on the analyses. The following conditions, which were identified during the benchmark and scoping analyses, were varied:

- Quench Time:
  The injection flow was reduced to 5-10% of design flow when the collapsed liquid level reached the core support plate. After approximately 500 seconds the injection flow was restored to the design value. The purpose of this study was to evaluate the impact of forced steam cooling on total clad oxidation with the core temperatures at or near the PCT for an extended duration. There was only a slight increase in the peak clad temperature and amount of clad oxidation. The worst case total clad oxidation remained below 0.2%.
- Break Size
  A number of smaller break size cases were performed to evaluate the RPV depressurization rate against meeting the LPCI permissive. Slower depressurization acts to delay receipt of the LPCI permissive and to delay LPCI injection. This study demonstrated that the highest peak clad temperature occurred for the double-ended guillotine break.

There was no change to the conclusion drawn from the base cases.

B.6.1 Effect of Plant Variability

MAAP4 Scoping Cases A and D (refer to Table B.5-2) at the 25% uprated condition generated the highest PCT for the BWR4 and BWR6, respectively (refer to Figure B.5-1). Obviously; core power level has a strong effect on PCT following a LBLOCA with reduced and delayed ECCS injection flow. This sensitivity provides an idea of how much PCT might be expected to increase for a given plant from Original Licensed Thermal Power (OLTP) to Extended Power Uprate (EPU) conditions, which will vary depending upon specific core operating conditions.

Within a plant type there is variability in the RPV size and total ECCS injection flow. A series of runs was performed using MAAP4 to evaluate the impact of a change in RPV liquid volume with the core power level at the 25% uprated condition and ECCS injection flow unchanged. The base cases for this series of runs were Cases A and D, discussed above. RPV liquid volume, which includes the shroud, lower plenum, shroud head, separators, and active core regions, was varied ±20%. The change in PCT for these cases was limited to approximately 100 °F for the BWR4 and approximately 200 °F for the BWR6. The largest increase in PCT occurred at lower RPV volumes (refer to Figure B.6-1).

Additional cases were analyzed to evaluate the effect of a 10% reduction in LPCI flow in conjunction with the change in RPV liquid volume. As can be seen in Figure B.6-1, the variation in LPCI flow accounted for a PCT change of approximately 25 °F for the BWR4 and less for the BWR6.

Plants with a smaller RPV tend to have a lower power level and plants with a larger RPV tend to have a higher power level. The variability cases performed in this evaluation maintained a constant power level. The effect of power level will tend to offset the impact of the variability shown in Figure B.6-1. The changes in PCT that were observed with expected variability in RPV liquid volume and LPCI flow do not alter the conclusion that margin to the objective of maintaining the core in the reactor vessel in a coolable geometry is preserved (refer to Sections B.4.2.6 and B.4.3.6). Therefore, the results of this evaluation are applicable to all BWR 3/4 and BWR 5/6 plants.

The BWR 2 plant type more closely resembles the BWR 3/4 used in this evaluation. However, the BWR 2 plants only have LPCS as an ECCS injection source. Therefore, the only applicable combinations of the Option 3 changes identified in Table B.5-1 are Categories 1 & 3, because automatic start of LPCS cannot be removed. This combination leaves Case B (refer to Table B.5-2) as the limiting case. Given that Case B was not the bounding case and the acceptable effect of plant variability in RPV liquid volume and ECCS injection flow, the results of this assessment are also applicable to the BWR 2 plants.

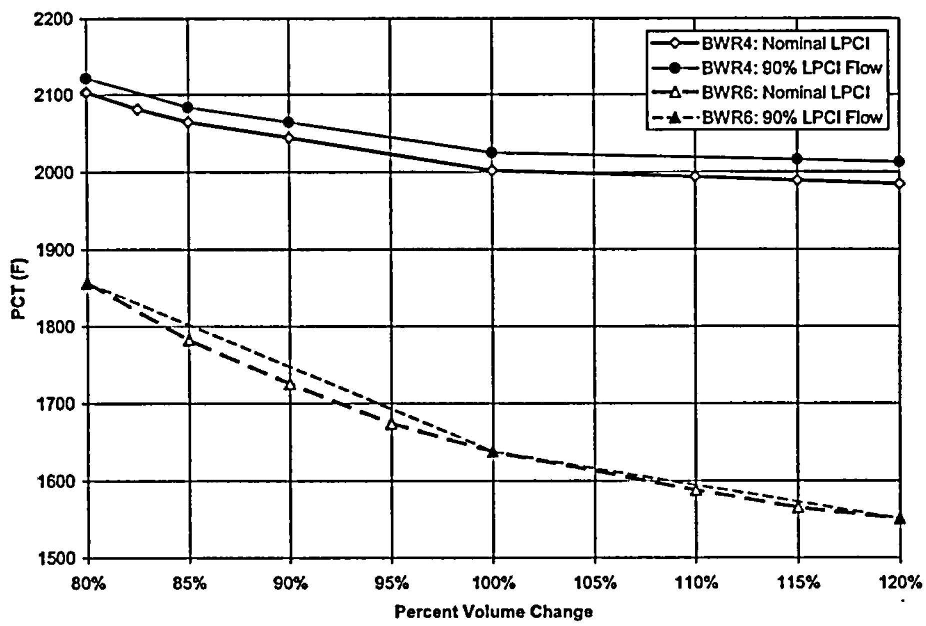


Figure B.6-1
MAAP4 Effect of Plant Variability on PCT

B.6.2 Cases Selected for Final Analysis

MAAP4 Scoping Cases A (BWR4) and C (BWR6), both at the 25% uprated condition, were selected for analysis using the TRACG02 computer code. Case A generated the highest PCT for the BWR 4 plants, so it was selected. The PCT in Case C was not the highest for the BWR 6 plants, but it was chosen to ensure that at least one TRACG02 case included a High Pressure Core Spray (HPCS) system. There is ample margin in the BWR6 cases to support this selection.

B.7 FINAL TRACG02 ANALYSES

B.7.1 Introduction

Consistent with the approach used to evaluate the impact of the proposed Option 3 changes, a series of MAAP4 scoping cases using the limiting equipment combinations remaining after implementing Option 3 was performed. The cases identified in Table B.5-2 were performed with MAAP4. Each case was run with and without a power uprate assumption.

The limiting MAAP4 scoping cases selected (Cases A and C) were then run using TRACG02 to generate the PCT associated with implementation of Option 3.

The limiting PCT case for the BWR4 was Case A at an uprate power of 25% and the PCT case that was run for the BWR6 was Case C at an uprate power of 25%. As indicated in Table B.5-2, Case A is characterized by injection using 2 LPCI with a 90 second emergency diesel generator (EDG) start delay and Case C is characterized by injection using 1 LPCI and 1 HPCS with a 120 second EDG start delay.

The PCT for BWR6 Case D (2 LPCI) was actually the highest PCT case for the BWR6. This can be seen from Figure B.5-1. It was decided to utilize Case C (1LPCI, 1HPCS) for the TRACG02 final analysis in order to be able to include the phenomenology associated with the injection of HPCS. Since the PCT between the cases was not substantially different, as can be seen in Figure B.5-1, it was judged to be more important to account for the phenomenology in TRACG02. This also provides a more meaningful reference comparison between TRACG02 and MAAP4 as described in the following section of this report.

B.7.2 LOCA Event Description

The event that is modeled in TRACG02, for both the BWR4 and the BWR6, is a double-ended guillotine rupture of a recirculation suction line coincident with a loss of offsite power (LOOP). In the analysis, the EDG is assumed to start on a Level 1-signal (L1) and an EDG delay following L1 is assumed in accordance with the Option 3 changes discussed earlier. A 25% power uprate is assumed in order to be consistent with the need to model the more limiting PCT case as identified in the MAAP4 scoping analyses. It should be noted that the L1 anticipatory start signal is not critical to the analysis since EDG start would also occur on bus undervoltage. The delay time is the important consideration, not the anticipatory start signal.

The system assumptions for the BWR4 and BWR6 analysis cases are different, reflecting the design differences between the plants. The specific cases analyzed and the results are discussed in the following sections.

Key parameter values used for the BWR4 and the BWR6 analyses are provided in Table B.7-1.

Table B.7- 1
Key Parameter Values Used in the BWR4 and BWR6 TRACG02 Analyses

Parameter	Description	BWR4 Analysis Value	BWR6 Analysis Value
RPV Dome Pressure, (Pa)		7348122.0	7232505.0
Reactor Core Power, (W)		3042400000.0	4510400000.0
Steam Flow, (kg/sec)		1674.0	2494.35
Feed water Flow (kg/sec)		1679.1	2492.02
Feed water enthalpy (J/kg)		0.9632E+06	0.9632E+06
Jet Pump #1 Discharge Flow Rate, (kg/sec)		4846.6	6446.3
Jet Pump #2 Discharge Flow Rate, (kg/sec)		4852.8	6494.3
Radial peaking	TRAC CPR limited channel	1.5085	1.3903
	TRAC PLHGR limited channel	1.4925	1.3617
ECCS Systems Available		Two LPCI in the same loop (intact loop)	1 LPCI 1 HPCS
ECCS Systems start signal	LPCI	L1	L1 (Both)
ECCS Systems delay time from L1 signal (seconds)	Delay time consists of an EDG start delay and a LPCI loop fill time delay	127	147 (HPCS) 157 (LPCI
ECCS Pump Curve	Two LPCI into one loop performance. The curve shown reflects the LPCI flow for one system only	Head(psia) Flow(ft^3/hr) 210.7 0.0000E+00 191.1 1.8992E+04 171.5 2.6860E+04 132.3 3.7984E+04 93.1 4.6517E+04 53.9 5.3710E+04 34.3 5.6975E+04 0.0 6.0060E+04	NA

ECCS Pump Curve (Cont'd)	One HPCS Pump	NA	Head(psia) Flow(ft^3/hr) 1.001E+7 0.0000E+00 1215 0.0000E+00 1215 2.646E+03 1162 9.67E+03 214.7 4.003E+04 14.65 4.811E+04 14.65 4.811E+04 14.65 4.811E+04
	One LPCI Pump into shroud	NA	Head(psia) Flow(ft^3/hr) 870.2 0.0000E+00 239.3 0.0000E+00 211.8 1.361E+04 194.4 1.894E+04 172.6 2.440E+04 149.4 2.899E+04 127.2 3.302E+04 104.7 3.668E+04 82.24 4.001E+04 59.76 4.311E+04 34.66 4.631E+04 14.65 4.992E+04

B.7.3 BWR4 Analysis Results

The limiting MAAP scoping analysis case which was run with TRACG02 for the BWR4 is the LBLOCA with 2 LPCI injecting into the intact loop and a 90 sec EDG delay time. The timing of various key events for the BWR4 analysis is provided in Table B.7-2. The time of ECCS actuation includes the time to reach L1 plus the additional time from L1 to the LPCI pressure permissive plus a 90 second EDG start delay plus a 20 sec LPCI fill time assumption. Injection is provided by two LPCI systems into the intact recirculation loop.

Figures B.7-2 and B.7-3 show the reactor power and reactor vessel pressure as a function of time. The scram occurs very early in the event and reactor blowdown occurs in less than 100 seconds. The break flow in Figure B.7-4 is negligible at approximately 50 seconds into the scenario. The period of time between the end of blowdown and the reflooding of the core following LPCI injection is characterized by a rapid core heatup as can be seen in the peak cladding temperature depicted in Figure B.7-5.

A plot of LPCI flow is provided in Figure B.7-1. As can be seen, injection by both systems into the intact loop begins at about 137 seconds and the peak cladding temperature does not turn around until approximately 225 seconds. PCT is shown on Figure B.7-5

The first 20 seconds of the blowdown result in a cladding temperature decrease because there is sufficient heat transfer from the fuel to the coolant and the coolant temperature is dropping as a result of the rapid depressurization. As the blowdown proceeds further, there is insufficient heat transfer to remove the stored energy and the fuel begins to heat up.

As can be seen in Figure B.7-5, the peak cladding temperature is about 1758°F at about 250 seconds into the event. In order to account for known TRACG02 model uncertainties in the calculation of the PCT, a PCT adder of 193°F was added to the PCT. This would result in a PCT of 1951°F. This is well below the PCT required to assure that the core is maintained in a coolable geometry.

Table B.7- 2
Timing of Key Events in the BWR4 TRACG02 Analysis

Reactor Scram	0.0	sec
Feed water Pump Trip	0.0	sec
MSIV Initiation (4 sec stroke time)	0.5	sec
Level 1	7.76	sec
ECCS Actuation	134.8	sec

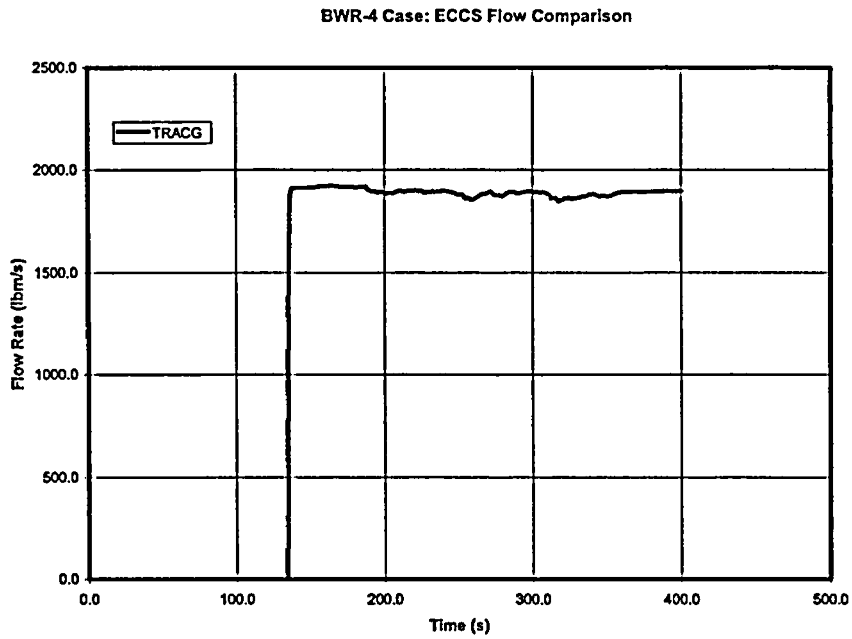


Figure B.7-1
BWR4 Case A: ECCS Flow

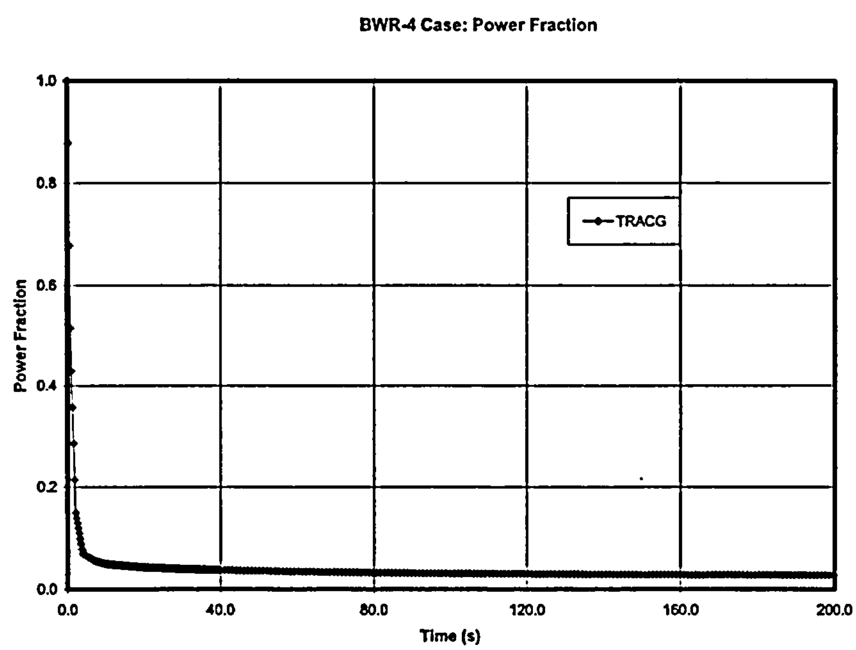


Figure B.7- 2
BWR4 Case A: Reactor Power

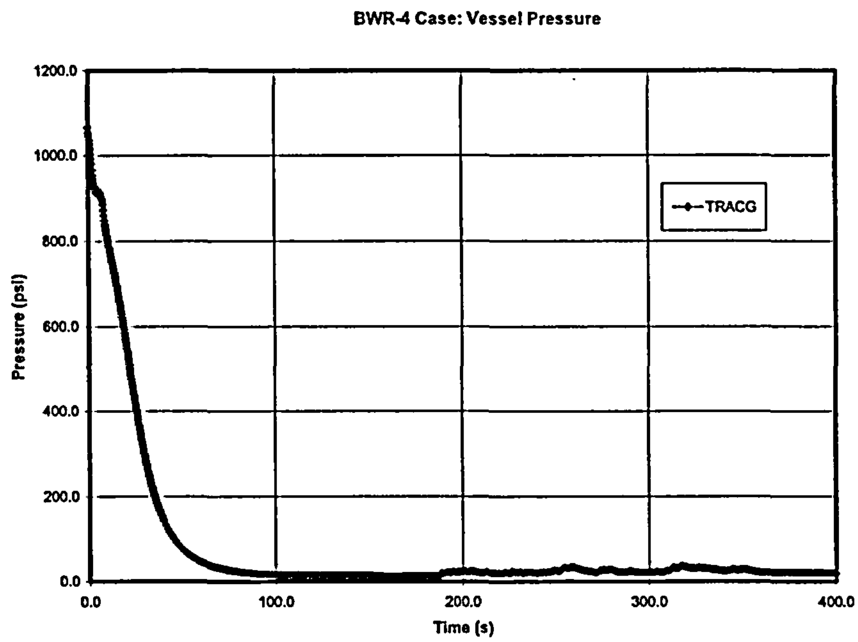


Figure B.7-3
BWR4 Case A: Reactor Vessel Pressure

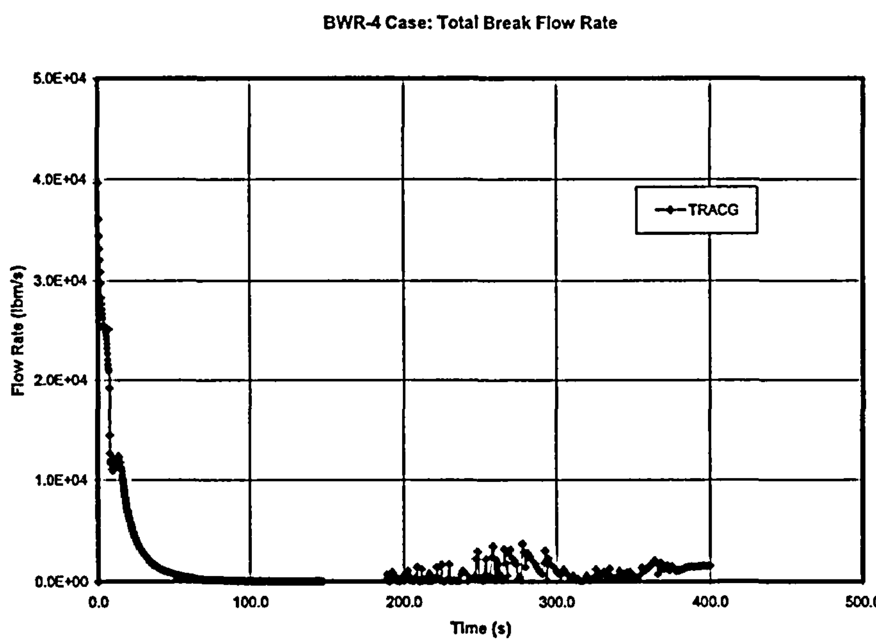


Figure B.7-4
BWR4 Case A: Break Flow Rate

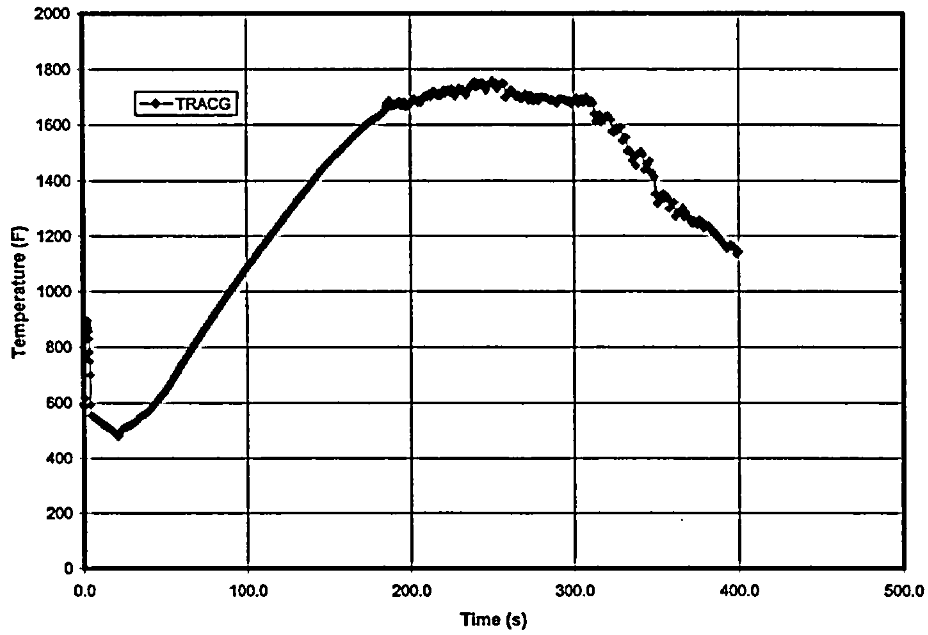


Figure B.7-5
BWR4 Case A: Peak Cladding Temperature

B.7.4 BWR6 Analysis Results

The limiting MAAP scoping analysis case, which was run with TRACG02 for the BWR6, is the LBLOCA with 1 LPCI injecting into the vessel and 1 HPCS injecting into the core with a 120 sec EDG delay time.

The timing of various key events for the BWR6 analysis is provided in Table B.7-3. The time of LPCI actuation includes the time to reach L1 plus the additional time from L1 to the LPCI pressure permissive plus a 120 second EDG start delay time assumption. The HPCS injection start signal was also assumed to be L1 for this analysis, rather than L2 plus high drywell pressure, in order to provide a more conservative response. A 120 second EDG time delay was also assumed for HPCS injection.

Figures B.7-7 and B.7-8 show the reactor power and reactor vessel pressure as a function of time. The scram occurs very early in the event and the reactor blowdown duration, based on the vessel pressure, is about 150 seconds. However, the break flow in Figure B.7-9 is very small by approximately 100 seconds. The period of time between the end of blowdown and the reflooding of the core following ECCS injection is characterized by a rapid core heatup as can be seen in the peak cladding temperature depicted in Figure B.7-10. Core heatup actually begins before the end of blowdown, at about 60 seconds, when there is insufficient core heat transfer to remove the stored energy in the fuel.

Figure B.7-6 shows the ECCS flows. As can be seen, injection by HPCS begins at about 154 seconds. The flow curve is flat after HPCS injection and before LPCI injection. The addition of LPCI flow can be seen at about 165 seconds.

The first 60 seconds of the blowdown result in a cladding temperature decrease because there is sufficient heat transfer from the fuel to the coolant and the coolant temperature is dropping as a result of the rapid depressurization. As the blowdown proceeds further, there is insufficient heat transfer to remove the stored energy and the fuel begins to heat up. This occurs at about 60 seconds into the blowdown as can be seen in Figure B.7-10.

As can be seen in Figure B.7-10, the peak cladding temperature is about 1422°F at about 292 seconds into the event. In order to account for known TRACG02 model uncertainties in the calculation of the PCT, a PCT adder of 212°F was added to the PCT. This would result in a PCT of 1634°F. This is well below the PCT required to assure that the core is maintained in a coolable geometry.

Table B.7- 3
Timing of Key Events In The BWR6

Feed water Pump Trip	0.0	sec
Reactor Scram	1.157	sec
Level 1	7.72	sec
MSIV Initiation (5 sec stroke time)	7.72	sec
LPCI Pressure Permissive	39.23	sec
HPCS Injection Begins	154.8	sec
LPCI Injection Begins	164.3	sec

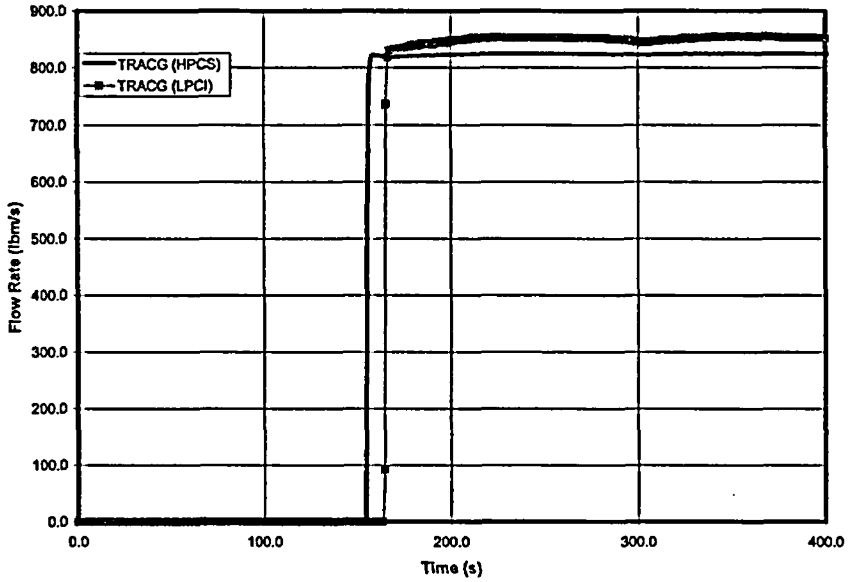


Figure B.7-6
BWR6 Case C: ECCS Flow

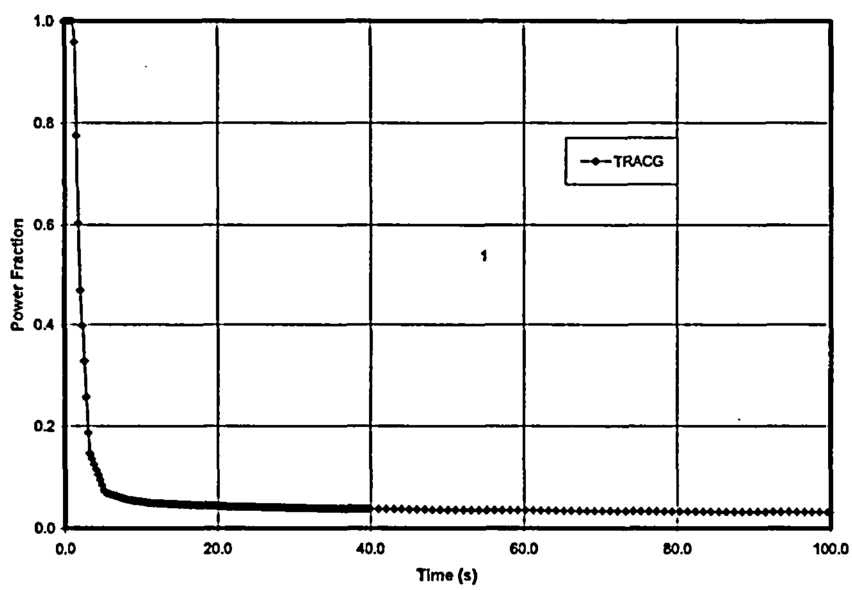


Figure B.7-7
BWR6 Case C: Reactor Power

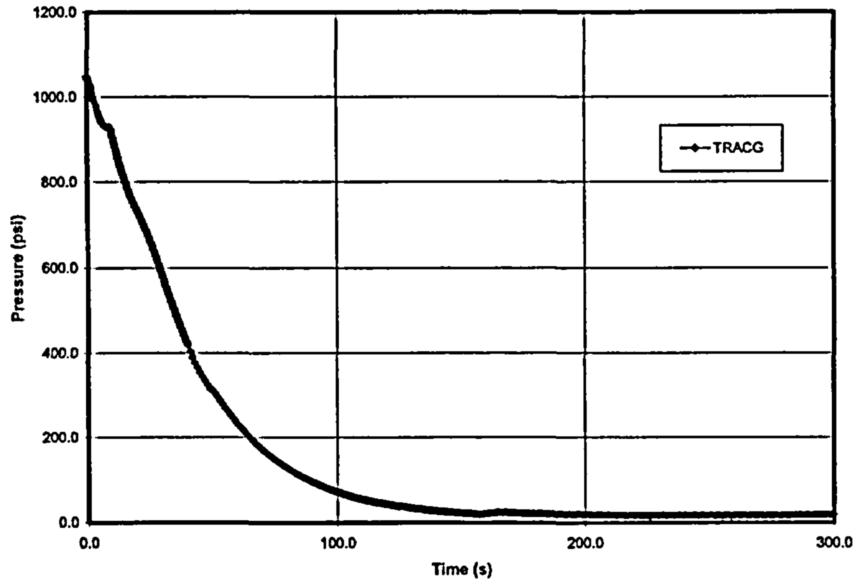


Figure B.7-8
BWR6 Case C: Reactor Vessel Pressure

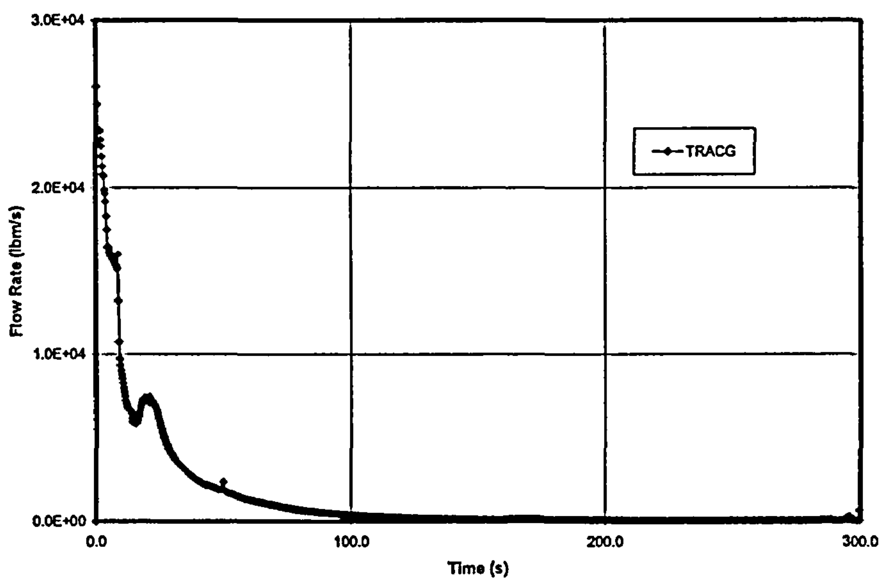


Figure B.7-9
BWR6 Case C: Break Flow Rate

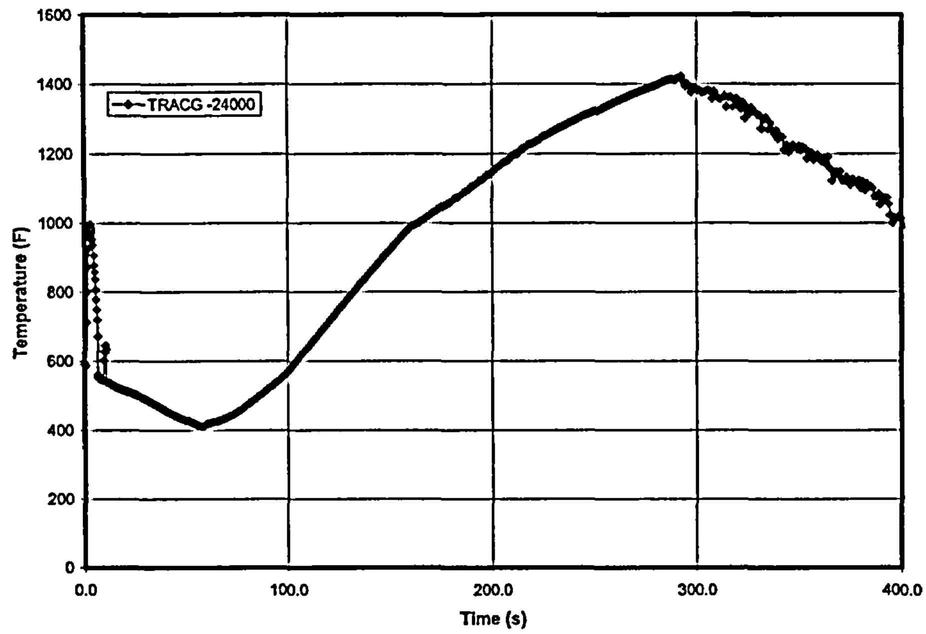


Figure B.7-10
BWR6 Case C: Peak Cladding Temperature

B. 8 SUMMARY & CONCLUSIONS

The objective of maintaining the core in the reactor vessel in a coolable geometry was met with significant margin. The TRACG02 cases for the limiting scenarios predict a peak clad temperature that is less than 2200 °F and no significant clad oxidation (much less than 1% global and less than 5% local clad reacted). Therefore, we have demonstrated that defense-in-depth, as specified in RG 1.174, has been achieved, with the adoption of the Option 3 changes.

APPENDEX C --

GENERIC PRA MODELING FOR LBLOCA/LOOP EXEMPTION

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C.1 PURPOSE

The purpose of this appendix is twofold. First, the generic BWR PRA model will be described. This will be followed by a risk analysis of the changes described in this report using the generic BWR PRA model. The appropriate uncertainty and sensitivity analyses are also included in this appendix.

C.2 INTRODUCTION

In order to evaluate the risk changes associated with the Option 3 implementation described in this report, the BWROG created a quantitative tool. It is called the generic BWR PRA model. It is a detailed model that can be configured to represent, and provide risk insights for, most operating US BWR plants. This analysis incorporates nine different configurations of the generic BWR PRA model.

The generic BWR PRA model was created from an existing, a plant specific, peer reviewed BWR4 model. Many of the plant specific features (i.e. those that were not broadly applicable) were removed from the model. Several features that are available at many plants were either retained or added as configuration options. This resulted in the set of models that was used to evaluate the changes in risk (CDF and LERF) that result from the Option 3 changes. The base BWR4 configuration is presented in section C.3.

In the course of developing this report, the key attributes of the model were compared with the plant specific modeling of the member BWR plants. These comparisons guided the development of the various configurations of the generic model. Section C.4 describes how the generic model was configured so that the results of this are applicable to all BWROG member plants.

Section C.5 of this appendix presents the quantitative and qualitative estimations of the risk deltas associated with the Option 3 changes. Each of the changes are evaluated independently, and in combination with other applicable changes.

Finally, sensitivity studies were performed on the key model attributes to ensure broad coverage of the BWR fleet. These are presented in Section C.6. This section also presents an evaluation of the assumptions in the model. The model attributes and assumptions that govern the results are reflected in the Implementation Guide in Section 9.

C.3 BASE PRA MODEL

The generic PRA model consists of a detailed level 1 internal events model that covers events that may occur while the plant is operating at full power. There is a simplified level 2 model that is based on an actual Mark I containment at an operating US plant. For the purposes of this application, shutdown events are not included because the LBLOCA with a consequential LOOP is not credible from the shutdown condition. In addition, external events are treated qualitatively.

C.3.1 System Models

This section provides a summary of the systems used in the generic PRA model. The description of each system contains attributes of the system that were considered at the time of model creation. Section C.6 provides an analysis of each of these attributes with respect to the Option 3 changes contained in this report.

C.3.1.1 High Pressure Injection Systems

There are four systems that can provide core coolant at high pressure. These are Feedwater, HPCI, RCIC, and CRD hydraulics.

The Feedwater system is assumed to be steam driven. In other words, the main steam lines must remain open in order to take credit for this method of injection. BOP support systems, such as TBCCW and non-1E electrical busses, are also required for Feedwater. Feedwater injection also requires Condensate and Main Condenser cooling in order to operate.

BWR 4 plants that have a motor-driven Feedwater system can get more credit in their PRA than the generic model because the main steam lines are not required to be open. The remaining support systems, however, remain the same. Therefore, the generic Feedwater model bounds all plant designs.

HPCI is a steam driven high pressure injection system that is designed to provide emergency high pressure injection during small LOCA accidents. It injects into the vessel outside the shroud. All of the controls for HPCI are DC powered, so the system is considered AC independent. In the generic model, however, room cooling (which is AC dependent) is required for HPCI to operate longer than 2 hours. It is possible for the operators to set up augmented cooling for the HPCI room that is independent of AC power. While this action can be taken in any accident or transient scenarios, the generic model only takes credit for this in SBO sequences.

The HPCI pump initially takes its suction from the CST. When the suppression pool level increases by about 4 inches, the suction automatically switches from the CST to the suppression pool. The time that the swap occurs is dependent on the specific accident or transient, but it is assumed to occur in less than 2 hours for all scenarios. For long-term core cooling with HPCI, SPC is required. The operators have the option to revert the water source to the CST via manual actions from the control room.

HPCI also uses a substantial amount of steam during its operation. Therefore, the generic model considers HPCI operation to be a valid means of depressurization for the plant.

RCIC is a steam driven high pressure injection system that is designed to provide emergency high pressure injection following transients. It injects into the vessel outside the shroud. All of the controls for RCIC are DC powered, so the system is considered AC independent. RCIC does not require room cooling in order to operate for the long-term.

The RCIC pump initially takes its suction from the CST. When the suppression pool level increases by about 4 inches, the suction automatically switches from the CST to the suppression pool. The time that the swap occurs is dependent on the specific accident or transient, but it is assumed to occur in less than 2 hours for all scenarios. For long-term core cooling with RCIC suppression pool cooling is required. The operators have the option to revert the water source to the CST via manual actions from the control room, but containment high pressure may cause the isolation of RCIC. Containment pressure must be kept low for RCIC to operate independent of the RCIC suction source. RCIC does not draw enough steam to depressurize the vessel to allow low pressure injection systems to operate.

The CRD system has two pumps, with one normally in operation. Following a loss of offsite power, the system must be restarted manually. This PRA model does not take credit for CRD to prevent core damage if it is the only high pressure injection source available.

In sequences where core injection is initially successful but subsequently fails because of suppression pool water temperature, the generic PRA model takes credit for CRD as a long-term injection system. This system takes its suction from the CST and must be aligned manually for 2 pump operation. Reactor depressurization is not required for CRD to be successful. The CRD system can be powered by both onsite and offsite electrical sources and it requires instrument air to be successful.

C.3.1.2 Low Pressure Injection Systems

There are four systems that can provide core coolant at low pressure. These are Condensate, LPCS, RHR in LPCI mode, and Service Water.

The Condensate system has three pumps that can provide low pressure injection as long as offsite power is available. The system takes its suction from the main condenser and makeup to the condenser is provided from the CST. When the plant is at low pressure, the condensate system can inject through the Feedwater lines.

No credit is taken for Condensate as an early injection source in LOCA sequences. If initial injection is successful, Condensate can be used for long-term cooling.

LPCS is a two pump system that is normally in a standby condition. Upon a low reactor level signal, the system automatically starts and injects inside the shroud. This system is powered by the safety related electrical system (each division being totally independent), so it can be run using either onsite or offsite power. Room cooling is not required for the LPCS system to operate for 24 hours.

The LPCS system normally takes its suction from the suppression pool, therefore suppression pool cooling is required for LPCS to provide long-term cooling for this alignment. There is also an alternate connection to the CST that can be aligned manually. This alternate connection can only be successful for long-term cooling because of the length of time it takes to establish the lineup.

LPCI is one of the modes of the RHR system. It is normally in a standby condition. Upon a low reactor water level signal, the system automatically starts and injects into the recirculation loops. This system is powered by the safety related electrical system, so it can be run using either onsite or offsite power.

The configuration for the LPCI system is consistent with the "LPCI Mod" configuration. Each of the two loops of LPCI has two pumps, but they are each powered by a different electrical division. The injection valve for the loop is powered by 250 V DC Power. There is a crosstie between the two LPCI loops such that any of the pumps can inject into either recirculation loop. In the "LPCI Mod" configuration, however, this crosstie is locked closed. No credit is taken for the crosstie in the LPCI model.

Room cooling is not required for the LPCI system to operate for 24 hours.
The LPCI system normally takes its suction from the suppression pool, therefore suppression pool cooling is required for LPCI to provide long-term cooling. RHR is a multi-mode system that also provides suppression pool cooling. In the PRA model, it is assumed that one division of RHR can provide both coolant injection and suppression pool cooling. This can be done by either alternating the mode of the loop or by passing LPCI flow through the RHR heat exchangers. There is also an alternate connection to the CST for two of the RHR pumps, one in each loop, that can be aligned manually. This alternate connection can only be successful for long-term cooling because of the length of time it takes to establish the lineup.

In circumstances where all other low pressure injection systems are unavailable, it is possible to align service water to inject into the vessel. The injection path is shared with the LPCI system. This alignment requires local manual actions, therefore is not used for early injection in LOCA sequences.

The service water system is powered by safety related electrical power, so it can be operated on either onsite or offsite power.

The event trees contain a place holder for low pressure injection using an AC independent fire water system. In the base model, this node is modeled as being unavailable. It is only used in the plant configurations described in Section C.4.

C.3.1.3 Depressurization Systems

The reactor can be depressurized using any of three systems. These are the Main Steam system, SRVs, or HPCI.

The Main Steam system (which is also designated as the Power Conversion System in the logic model) normally conveys the steam from the reactor to the turbine generator. Following a plant shutdown, the turbine can be bypassed and steam is provided directly to the main condenser. In the PRA model, this system is only considered available if there is a path for water to be removed from the condenser back to the reactor vessel using Condensate pumps.

The Main Steam system operates automatically to control reactor pressure near the normal operating range for the reactor. The operators can manually adjust the system to provide depressurization.

Main Steam is automatically isolated if there is a low reactor water level signal, so no credit is allowed in small or medium LOCA sequences. In a small LOCA, Main Steam remains available as long as the high pressure injection systems prevent the low water level signal. Main Steam is also isolated if there is a low reactor pressure if operators do not take action to prevent this isolation. This mode of isolation is not modeled because it would be modeled the same as the low level isolation. Finally, Main Steam will be isolated if the condenser is not cooled by the Circulating Water systems. Circulating Water is not powered by safety related electrical systems, therefore Main Steam is assumed to be isolated following a loss of offsite power.

SRVs provide a safety related means of depressurization. They discharge steam from the reactor to the suppression pool. The valves are DC powered and are controlled from the control room. There is an automatic mode for this system, but the PRA model assumes that this mode will be disabled by the operators (per procedure) shortly after the initiation of the accident or transient.

The HPCI system is described in section C.3.1.1.

C.3.1.4 Containment Control Systems

Containment control can be provided by either the RHR system or the containment vent.

The purpose of the containment control systems is twofold. First, the pressure inside the containment must not exceed the ultimate pressure capability of the structure. Second, the containment control systems keep the suppression pool water temperature low enough that the ECCS systems can continue injecting water into the vessel. The SPC and CSS modes of RHR perform both of these functions. The SDC mode of RHR provides both of these functions plus the water injection function by removing heat directly from the primary system, thus preventing heat from being deposited into the containment. The containment vent system only addresses the pressure control function.

The RHR system consists of two loops with two pumps and one heat exchanger in each loop. All of the containment control functions of RHR must be manually initiated. Cooling for the heat exchangers is provided by the service water system.

In SPC mode, the pumps take their suction from the suppression pool, pass the water through the heat exchangers, and return it to the suppression pool. All of the pumps and valves in the system can be powered by both onsite and offsite electrical sources.

CSS mode is similar to SPC, except the water is sprayed into the containment atmosphere rather that being returned directly to the suppression pool. The CSS mode is only included in the LOCA sequences. The containment spray valves can be powered by both onsite and offsite electrical power sources. In the PRA model, CSS must be cooled by the RHR heat exchangers for the system to be considered successful.

In SDC mode, pump suction is taken directly from the reactor vessel. Flow is passed through the RHR heat exchangers and is returned to the vessel via the LPCI injection valves. Even though the shutdown cooling suction valves are not safety related, they are powered such that offsite electrical power is not required. The SDC mode is only used in non-LOCA sequences.

The containment vent can be used to control containment pressure and allow injection systems that take their suction from outside containment to be successful. The vent provides a flow path from the suppression pool air space to the plant elevated release point. The valves required for the system to succeed are a mixture of AC and DC powered valves.

C.3.1.5 Reactivity Control Systems

The primary reactivity control system is RPS. This system provides the signal and motive power to insert the control rods into the reactor. It is modeled in two parts. First is the mechanical portion that causes control rod motion. Second is the electrical portion that provides the signals to the control rod drives. These functions are independent of the other systems modeled, therefore they are modeled as undeveloped events. The probabilities for these undeveloped events are based on the Cooper Nuclear Station evaluation of RPS reliability.

The electrical portion of RPS is backed up by a manual scram of the reactor and the ARI system. These provide independent and diverse means of initiating control rod movement. The mechanical portion of RPS does not have any backup systems.

If RPS fails, reactor power can be reduced by tripping the reactor recirculation pumps, reducing water level in the reactor, and by adding dissolved boron to the reactor coolant. The RPT function is automatic and shares much of its logic with the ARI system. It is assumed that if the signal is generated, trip of the pumps will succeed with certainty.

Reducing water level to control power is an ATWS mitigating strategy contained in the Emergency Operating Procedures. In the model, only the human action portion is included. Reducing water level may impact the way that other systems in the plant operate. The event tree models take this interaction into account.

SLC can also be used to reduce reactor power in the event that the other RPS systems fail. The system has two pumps that inject into the vessel via a single penetration. If the Main Steam system is available, one SLC pump will be sufficient to reduce power to the point where the containment integrity is protected. In Main Steam is not available, two SLC pumps are required to reduce reactor power fast enough to protect containment in all cases.

The SLC system can be powered by either onsite of offsite electrical power sources. The system must be manually initiated.

C.3.1.6 Other Front Line Systems

There are three other functions included in the PRA model. These are reactor over pressurization protection, re-closure of the SRVs if they open, and the Pressure Suppression System. On the event trees, these are designated M, P, and PSS respectively. All of these systems are modeled as a point estimate, and do not have detailed fault trees.

The reactor over pressurization protection function represents the probability that one or more SRVs will open following the closure of MSIVs. If the SRVs do not open, there is a chance that the reactor vessel or the recirculation piping can be damaged or rupture. The SRVs in this function operate in safety mode, and no electrical or pneumatic support is required.

If the SRVs are postulated to open during the accident or transient, the P function models the probability that one or more of the SRVs will stick open. If this happens, some of the high pressure injection systems are no longer considered viable. No support systems are required for the SRVs to re-close.

PSS models the chance that the pressure suppression containment may not function as designed. In LOCA sequences, this could happen if one or more wetwell to drywell vacuum breakers do not seat. If this happens, much of the steam from the reactor will not be directed below the surface of the suppression pool water and the containment pressure will quickly rise to the failure pressure. In a stuck open relief valve case, the phenomenon is a little different, but the probability is nearly the same. For these sequences, PSS is the probability that the steam discharges into the suppression pool air space. For this to happen, the tailpipe connected to the SORV would need to fail.

C.3.1.7 Support Systems

The PRA model includes several support systems. Four of these systems will be described here. These are the service water system, onsite AC power, offsite AC power, and DC power.

The service water system contains four pumps that provide water from the ultimate heat sink to cool various plant components. The system is cross connected such that any one service water pump can provide cooling to any equipment. There is an interlock in the system such that header low pressure will cause the isolation of the cross connect. In the PRA model, it is assumed that this cross connect will always receive a signal to isolate. If the isolation fails, two pumps per loop are required for success. Otherwise, one pump per loop is sufficient.

When the service water system is providing cooling to the RHR heat exchangers or providing vessel injection, an additional set of booster pumps (two per loop) are required to increase the pressure in the RHR heat exchanger.

Onsite AC power is provided by two diesel generators. Each DG can provide power to only one safety related bus. The DGs require service water and room ventilation in order to be successful.

Offsite AC power is provided by two nearly independent sources. First, following a turbine trip the offsite AC power is automatically transferred from the main generator to the startup transformer. This transformer is powered by the offsite grid. This transfer, called the "Fast Transfer" is bumpless and does not require any load shedding or re-starts to keep running plant equipment in their operating state. If for any reason the startup transformer fails to provide power, there is a "Slow Transfer" of the power source to the emergency transformer, which is also powered by the offsite grid, but from a different substation located several miles from the plant. The slow transfer requires that many large loads be shed from the safety busses and then be restarted in a predetermined sequence. Also, the emergency transformer only provides power to the safety related busses.

As stated above, these offsite AC power subsystems are nearly independent. If the reason for the loss of offsite power is associated with equipment failures between the plant switchyard and the plant itself, the emergency transformer provides an independent power source. If the reason that offsite power is unavailable is a result of weather (i.e. ice storms, tornado, etc.) or associated with a grid failure, then the emergency transformer is considered completely coupled with the normal offsite source.

There are two DC subsystems in the plant model. One is a 125 V system that is used for most control systems and for various valves in the plant. The other is a 250 V system that is chiefly used for HPCI and the LPCI injection valves. These are completely independent systems; they do not share any battery cells. Each subsystem has two independent divisions.

The batteries are kept in a charged state during normal plant operation through the station battery chargers. There are three battery chargers per subsystem. Each battery has one charger in operation. Each subsystem has a swing charger that can be aligned to either battery. The swing charger is normally in a standby state. In the event of a loss of offsite power, the battery chargers are one of the loads that are stripped from the safety buses. The chargers must be manually re-aligned to their respective batteries.

C.3.2 Event Sequences

The generic model contains nine event trees. Each tree models the plant response for one or more initiating events. The nine event trees are:

- Transients
- Loss of Offsite Power
- Loss of Service Water
- Large LOCA
- Medium LOCA
- Small LOCA
- ATWS Following a Transient
- ATWS Following a Loss of Offsite Power
- ATWS Following a Large or Medium LOCA

The success criteria for each of the event trees are presented in Table C.3-1. The following sections provide a brief summary of each event tree model.

C.3.2.1 Transient Event Tree

The Transient event tree is used for initiators that involve a plant trip with offsite power available and service water available. These initiating events are:

- Transients with the Condenser Available
- Transients with the Condenser Unavailable
- Loss of Feedwater
- Inadvertent / Stuck Open Relief Valve
- Loss of Reactor Building Closed Cooling Water
- Loss of Instrument Air
- Loss of a Single 125 V DC Bus
- Loss of a Single Safety Related 4160 V AC Bus

- Loss of Turbine Building Closed Cooling Water
- Loss of Instrument AC Power
- Reactor Coolant Leaks

The plant response to each of these initiators is similar; however the equipment available to respond to the transient is different. The quantification of the model appropriately considers what equipment is not available as a consequence of the initiating event. The Transient Event Tree logic is shown in Figure C.3-1.

The tree starts out by determining the state of the plant. Sequences where the SCRAM function fails are treated in the ATWS event tree (Section C.3.2.7). Sequences that involve a loss of offsite power caused by the plant trip are modeled in the LOOP event tree (Section C.3.2.2). Sequences in which the primary system ruptures due to over-pressurization are treated in the LLOCA event tree (Section C.3.2.4)[1]. Finally, the state of the PCS and SRVs are determined so the remainder of the progression can be determined.

For sequences with PCS available, either high pressure injection systems need to function or the operators need to depressurize the plant. This would allow Condensate (which is implied successful by PCS success) to provide long-term injection, rejecting heat to the UHS.

For sequences in which the PCS is not available and all SRVs successfully re-close, HPCI and RCIC can provide coolant injection at high pressure. The alternative is for the operators to depressurize the plant, allowing Condensate, LPCS, LPCI, or Service Water Injection to provide coolant at low pressure. Fire water injection is also shown on the event trees, but it is only available in certain plant configurations.

For sequences in which one or more SRVs stick open (or SORV initiated events), the short-term injection response is assumed to be similar to those sequences where all SRVs re-close.

All sequences in which the PCS is not available require long-term core cooling. This involves removing heat from the containment and providing injection to the core. Containment cooling can be provided by the SPC and SDC modes of RHR. If either of these is successful, any low pressure system available or CRD can provide long-term injection. If the containment cooling systems are not available, venting can provide containment pressure control while long-term injection must be provided from a source that is external to the containment (i.e. CST).

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[1] If the transient initiator is a SORV, the pressure relief function (M) is assumed to be successful. To account for this, SORV initiated sequences that also have failure of M are excluded from the solution.

C.3.2.2 Loss of Offsite Power Event Tree

The LOOP event tree is used for initiators that are caused by an independent loss of offsite power, or by a transient with an induced loss of offsite power. These initiating events are:

- Grid Centered Loss of Offsite Power
- Plant Centered Loss of Offsite Power
- Weather Related Loss of Offsite Power

The following transients, combined with a consequential LOOP, are also included:

- Transients with the Condenser Available
- Transients with the Condenser Unavailable
- Loss of Feedwater
- Inadvertent Open Relief Valve
- Loss of Reactor Building Closed Cooling Water
- Loss of Turbine Building Closed Cooling Water

These transients are assumed to result in a plant trip. All other transients would involve an orderly plant shutdown, making it less susceptible to an induced LOOP.

The plant response to each of these initiators is similar; however the equipment available to respond to the transient is different. The quantification of the model appropriately considers what equipment is not available as a consequence of the initiating event. The Loss of Offsite Power Event Tree logic is shown in Figure C.3-2.

The tree starts out by determining the state of the plant. Sequences where the SCRAM function fails are treated in the ATWS Following Loss of Offsite Power event tree (Section C.3.2.8). Sequences in which the primary system ruptures due to over-pressurization are treated in the LLOCA event tree (Section C.3.2.4)[2]. The state of the SRVs is determined so the remainder of the progression can be determined. The PCS is not available in LOOP sequences. Finally, the availability of onsite power is checked to separate the SBO from non-SBO sequences.

For non-SBO sequences in which all SRVs successfully re-close, HPCI and RCIC can provide coolant injection at high pressure. The alternative is for the operators to depressurize the plant, allowing LPCS, LPCI, or Service Water Injection to provide coolant at low pressure. Fire water injection is also shown on the event trees, but it is only available in certain plant configurations.

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[2] If the transient initiator is a SORV, the pressure relief function (M) is assumed to be successful. To account for this, SORV initiated sequences that also have failure of M are excluded from the solution.

For non-SBO sequences in which one or more SRVs stick open (or SORV initiated events), the short-term injection response is assumed to be similar to those sequences where all SRVs re-close.

All non-SBO sequences require long-term core cooling. This involves removing heat from the containment and providing injection to the core. Containment cooling can be provided by the SPC and SDC modes of RHR. If either of these is successful, any low pressure system available or CRD can provide long-term injection. If the containment cooling systems are not available, venting can provide containment pressure control while long-term injection must be provided from a source that is external to the containment (i.e. CST).

In SBO sequences, only AC independent systems (and Fire Water, if the specific configuration includes that system) are available. These systems must provide both short-term cooling and long-term injection. If AC power is not recovered, these sequences will result in core damage.

C.3.2.3 Loss of Service Water Event Tree

The Loss of Service Water event tree provides a simplified evaluation of this event. The logic is presented in Figure C.3-3.

Short-term injection is not included the model because the effects of LOSW would not be realized for several hours into the event. The equipment failures are already included in the Transient event tree, which has an initiating event frequency that is more than three orders of magnitude higher than the LOSW event. Therefore, including those same failures in this model would not provide any significant change in CDF or provide any additional insights. ATWS and LOOP sequences combined with the LOSW are also implicitly treated elsewhere in the model.

Long-term cooling can only be provided by the containment vent in conjunction with a low pressure injection source from outside the containment.

C.3.2.4 Large LOCA Event Tree

The Large LOCA event tree models events initiated by breaks in the primary system that are large enough to depressurize the plant to the point where low pressure systems can inject prior to core damage. No high pressure injection is credited in this analysis. The logic for LBLOCA events is presented in Figure C.3-4.

ATWS following LBLOCA is analyzed in section C.3.2.9.

The break in the primary system is assumed to be in a recirculation line. Any injection system that relies on the integrity of that line is assumed to be unavailable in the analysis.

Short-term low pressure injection can be provided by LPCS or LPCI, with suction from the suppression pool only. Containment cooling can be provided by the SPC and Spray modes of RHR. If short-term injection is successful, but containment cooling is not, long-term injection can be provided by venting the containment and aligning any of the low pressure systems that take suction from outside the containment.

C.3.2.5 Medium LOCA Event Tree

The Medium LOCA event tree models events initiated by breaks in the primary system that are large enough that RCIC cannot prevent core damage, but small enough that HPCI can provide enough flow to prevent core damage prior to low pressure system initiation. The logic for MBLOCA events is presented in Figure C.3-5.

ATWS following MBLOCA is analyzed in section C.3.2.9.

The location of the break in MBLOCA is assumed to be in a recirculation line, but this assumption does not have any affect on the progression of the event.

Short-term high pressure injection can be provided by HPCI. If it fails, the plant must be depressurized through the SRVs in order for low pressure systems to be successful. Short-term low pressure injection can be provided by LPCS or LPCI, with suction from the suppression pool only. Containment cooling can be provided by the SPC and Spray modes of RHR. If short-term injection is successful, but containment cooling is not, long-term injection can be provided by venting the containment and aligning any of the low pressure systems that take suction from outside the containment.

C.3.2.6 Small LOCA Event Tree

The Small LOCA event tree models events initiated by breaks in the primary system that are large enough that CRD cannot provide makeup and small enough that RCIC can provide enough flow to prevent core damage. The logic for the SBLOCA events is presented in Figure C.3-6.

ATWS following SBLOCA is analyzed along with the transients in section C.3.2.7.

The location of the break in SBLOCA is assumed to be in a recirculation line, but this assumption does not have any affect on the progression of the event.

The PCS and Feedwater systems can be successful in preventing core damage in a SBLOCA. Other high pressure systems available are HPCI and RCIC. If the high pressure injection systems fail, the plant must be depressurized either by the PCS or via the SRVs. This will allow low pressure systems to provide core cooling. In this event tree, the only low pressure systems that are available for short-term low pressure injection are LPCS and LPCI with suction from the suppression pool.

Containment cooling can be provided by either the SPC or Spray modes of RHR. If these are not successful, long-term cooling can be satisfied by venting the containment and injecting water from a source outside containment. The available long-term injection systems include Condensate, service water, and fire water (if present in the specific configuration), in addition to LPCS and LPCI with suction from the CST.

C.3.2.7 ATWS Following a Transient Event Tree

The ATWS following a transient event tree models the response of the systems and operator actions needed to control reactivity in the core and the amount of heat deposited into the containment. The logic for this event tree is presented in Figure C.3-7.

The function of the control rods is split into two different top events: mechanical and electrical. Mechanical includes the reliability of the rods themselves and the scram discharge volume. It is assumed that mechanical failure of the control rods does not have any backup. Electrical failure is backed up by ARI and by a manual scram signal.

It is assumed that failure of the electrical portion of RPS and its backups will lead directly to core damage. This is conservative, because as can be seen below, there are other means of controlling power. This assumption is tolerable because of the high reliability of the electrical portion of RPS and its backup systems.

If the control rods do not stop the nuclear reaction in the core, power can be reduced by several other means, including trip of the recirculation pumps, injection of boron into the primary system coolant, or reduction of water level in the reactor.

Short-term injection systems available are Feedwater and HPCI, and LPCS and LPCI provided the reactor is depressurized. It is assumed that these are only available if boron is injected into the coolant. Containment cooling is provided by the SPC mode of RHR.

If boron is injected into the reactor coolant and at least one of the injection systems listed above is available, long-term cooling can be provided by any of the available low pressure systems that take suction from outside the containment in conjunction with the containment vent.

C.3.2.8 ATWS Following a Loss of Offsite Power Event Tree

The ATWS following a LOOP event tree models the response of the systems and operator actions needed to control reactivity in the core and the amount of heat deposited into the containment. The logic for this event tree is presented in Figure C.3-8.

This event tree is similar to the ATWS following a transient event tree, except Condensate is not available for long-term cooling.

C.3.2.9 ATWS Following a Large or Medium LOCA Event Tree

The ATWS following a LBLOCA or MBLOCA event tree models the response of the control rods following one of these LOCAs. The logic for this event tree is presented in Figure C.3-9.

It is assumed that if the control rods do not go into the core and shut down the nuclear reaction following large or medium LOCA, core damage will occur.

C.3.3 Initiating Events

The initiating events that were evaluated in the generic model were the following:

- Transients with the Condenser Available
- Transients with the Condenser Unavailable
- Loss of Feedwater
- Inadvertent Open Relief Valve
- Loss of Reactor Building Closed Cooling Water
- Loss of Instrument Air
- Loss of a Single 125 V DC Bus
- Loss of a Single Safety Related 4160 V AC Bus
- Loss of Turbine Building Closed Cooling Water
- Loss of Instrument AC Power
- Reactor Coolant Leaks
- Small LOCA
- Medium LOCA
- Large LOCA
- Loss of Service water

The values used for these initiators were derived from NUREG/CR-5750 (Reference 3). Table C.3-2 provides the frequencies and distributions for all of the initiating events. The BWROG has determined that this reference provides the best representation of initiator values that are applicable across the fleet.

As part of the benchmark analysis (Section C.4), BWROG member plants were asked to provide the values that they used for several initiating events. These were Large LOCA, Medium LOCA, Small LOCA, and LOOP. Figures C.3-10 through C.3-13 show the ranges of these initiator values.

These figures show that there is quite a wide variation in many of the initiator frequencies, especially in the area of LOCA. Medium LOCA has the widest variation. This variation is expected because of the lack of LOCA data. Many of the plants use values that are consistent with the values used in the generic model. A sensitivity analysis was performed to analyze the variation in the LOCA frequencies.

C.3.4 Quantification

The generic model and its variations were quantified using a linked fault tree method. All of the event trees described in Section C.3.2 were quantified with the top event fault trees fully linked into the model. Deleting the cutsets that would be subsumed by the applicable top event fault tree simulated success branches. Complement events were not used in generating the model solution.

The truncation value used in the quantification was 1×10^{-12} per year. In order to retain all of the necessary terms for sensitivity analyses, in addition to the low truncation level the initiating event frequencies were set to the highest value reported by either BWROG member utilities or the 95th percentile from NUREG/CR-5750, whichever is higher. Following quantification, the initiator values were then restored to their mean values.

Following generation of cutsets for each sequence, all of the sequence results were combined into a single cutset list. This list was then passed through the "subsume" process to ensure that only minimal cutsets were retained. This step is necessary because of the way that success branches in the event tree are solved.

Table C.3-3 shows the results for all of the model configurations that were quantified.

C.3.5 Simplified Level 2

The generic model uses a simplified Level 2 analysis. It is derived from a plant specific Level 2 analysis of a member utility, which was developed using a fully developed set of containment event trees. The simplified analysis used in the generic model represents a Mark I containment configuration. It is assumed that the use of this configuration bounds the LERF value for other BWR containment designs.

The Level 2 model only estimates the LERF value for the generic model. No other release modes are addressed.

LERF is estimated as follows. First all of the core damage sequences are gathered into plant damage state bins. All of the sequences in a particular bin have a similar progression of events and phenomena. The frequency of core damage for each of the bins is estimated by summing the CDF for the sequences collected into the bin. A conditional probability of large early release given core damage of the type defined by the bin is obtained from the plant specific Level 2 analysis. The frequency of the PDS bin is then multiplied by the conditional probability of a large early release associated with the bin. Total LERF is obtained by summing the LERF values for all PDS bins.

The definition of the PDS bins and the associated conditional probability of large early release is shown in Table C.3-4.

C.3.6 External Events

There are several initiating events that are not affected by the changes proposed by this report. These are internal fires, internal flood, and seismic events. This section provides a qualitative discussion of these events. ISLOCA is also discussed in this section. Other external events have been postulated for BWRs, but they do not significantly impact CDF or LERF[3].

There are no credible fire events that can cause a LBLOCA, therefore the proposed changes to the operation and initiation of diesel generators will not have any detrimental affect. The improved reliability of the EDGs, however will have a beneficial effect on any fire event that results in a loss of offsite power. The changes that involve optimizing EDG loads and the configuration of RHR systems only affect a small portion of the ECCS available during LOOP events, but provide other benefits to offset the reconfiguration. A review of the importance measures for the base model and the variations confirms that these changes also have a beneficial effect of LOOP events, and by implication, fire events that result in a LOOP. It is judged that the changes proposed in this report will result in a reduction in risk for all fire initiating events.

Internal flood events are similar. There are no credible flood events that can cause a LBLCOA, therefore the proposed changes to the operation and initiation of diesel generators will not have any detrimental effect. It is also not likely that a flood will cause a loss of offsite power. If it did, the improved reliability of the EDGs will be beneficial in these particular flood sequences. It is judged that the changes proposed in this report will result in a reduction in risk for all internal flooding events.

Seismic events can cause a loss of offsite power at a BWR. For the same reasons described above, the changes proposed in this report will have a beneficial effect on these seismic events. If the magnitude of the event is large enough, it is possible that the primary system piping could be damaged[4]. The typical construction of BWRs is such that any earthquake that can cause a LBLOCA will also cause failure of the EDGs. Therefore, there would be no change in the CDF associated with very large seismic events. It is judged that the changes proposed in this report will result in a reduction in risk for seismic initiated events.

NUREG/CR-5750 cites the ISLOCA event frequency for BWRs as being less than 1×10^{-8} per year. This combined with the probability of failure to isolate the break (typically 0.1) and the conditional probability of LOOP (0.01) places this event frequency well beyond the realm of consideration for this report. It is judged that the changes proposed in this report will result in a negligible change in risk due to ISLOCA events.

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[3] Severe weather, such as hurricanes, is included in the Loss of Offsite Power initiating event.
[4] Seismic events of a magnitude great enough to damage the primary system is beyond the design basis at nuclear power plants.

C.3.7 Component Failure Data

The failure data for ECCS equipment used in this evaluation was taken from the draft MSPI user manual, Appendix F (Reference 11). This data is considered to be acceptable performance for ECCS equipment at nuclear power plants. In addition, any other non-ECCS components that were the same type (e.g. MOV) as ECCS equipment used the same failure rates.

The unavailability data for ECCS equipment used in this evaluation was derived from the 3nd quarter 2003 Reactor Oversight Process reported data for BWRs.

The balance of the failure and unavailability data was taken from the plant from which the generic model was derived. In addition, common cause factors were taken from the underlying plant specific model.

All mission times were assumed to be 24 hours, even though not all equipment is required to run for the full 24 hours for each sequence. This provides a conservative estimate of the associated CDF.

C.3.8 Uncertainty Analysis

In order to provide more confidence in the model results, a numerical uncertainty analysis was performed on each of the BWR 4 and BWR 6 model configurations. The method chosen was Monte Carlo. Figures C.3-14 through C.3-19 show the results of these evaluations.

The results of this analysis are as expected. The ratio of mean to 5th percentile ranges from a factor of 5 to 15, and the ratio of 95th percentile to the mean is approximately a factor of 3. The CDF calculated using the mean values of all data lies near the mean value of the calculated distributions. This is typical of internal events PRA analyses.

C.4 BENCHMARK WITH BWROG PLANT SPECIFIC ANALYSES

In order to ensure that the conclusions of this report are applicable to all BWRs, the generic model results were compared to the plant specific PRA results for several BWRs. When a plant attribute was identified that caused significant differences in the calculate CDF, a version of the generic model was created that would address that particular difference. In all, four distinct plant configurations (in addition to the base model described in C.3) were identified:

- BWR 5/6 ECCS Configuration
- BWR 3/4 with LPCI Loop-Select Logic
- AC Independent Low Pressure Injection System
- Additional Independent AC Power Source

These are described in the sections below.

In addition, one other factor was identified that has an affect on CDF. This is station battery life. The generic model assumes a nominal 4 hour battery life for the station batteries. Plants whose batteries discharge over a greater time period tend to have a significantly lower CDF, while those that do not last as long tend to have a higher CDF. The reason for this difference is associated with the value applied for offsite AC power recovery. This factor was not included as a distinct version of the generic model, but was addressed in the sensitivity analyses.

The comparisons performed in this section were done based on conditional core damage probability (CCDP). This was done to eliminate the effects of different initiating event data, and is more useful for comparing the model logic. In this section, CCDP is calculated by taking the CDF for a particular event tree and dividing by the value of the initiating event frequency.

Most of the comparisons were made using the Loss of Offsite Power event tree. The reason for this is that, in general, this initiator is responsible for the greatest fraction of CDF at BWRs. In addition, many of the changes described in this report affect only the LOOP events. It is assumed that if the CCDP for LOOP events match between a plant specific model and the generic model, the CCDP for other transients will also be similar.

C.4.1 BWR 6 Model

One obvious configuration difference is the design of the ECCS in the BWR 5/6 class of plants. In this configuration, there is only one LPCS pump. It is assumed to be powered by electrical division I.

There are 3 LPCI pumps; one powered by division I and two by division II. These pumps inject directly into the core via three separate injection lines, rather than into the recirculation loops. In this configuration, none of the LPCI loops are affected by the LBLOCA initiator. Two of the pumps, one on each division, can also be used for suppression pool cooling, shutdown cooling, or containment spray.

The BWR 6 does not have a HPCI pump. Instead, it has a High Pressure Core Spray (HPCS) pump that is powered by offsite power or by a dedicated diesel generator. Because it is a motor driven pump, it is capable of long-term injection in all sequences, including LBLOCA and MBLOCA.

C.4.2 BWR 4 with LPCI Loop-Select Model

The LPCI loop-select model was created to explicitly evaluate the elimination of the loop-select function. In this model, flow from any of the four RHR pumps can be directed to the core via the intact recirculation loop. A basic event is included that accounts for the failure of the loop-select logic, which would fail all LPCI injection during a LBLOCA.

The other difference is that the RHR pumps are not arranged with a cross-divisional electrical configuration. All of the equipment in loop A is powered by division I electrical, and all of the equipment in loop B is powered by division II electrical. The loop-select portion was only included in the LPCI fault tree for LBLOCA early injection. The electrical configuration was included for all modes of RHR operation.

C.4.3 AC Independent Low Pressure Injection Model

Many BWRs have the capability to inject at low pressure using an AC independent system, such as a diesel driven firewater pump. This capability can significantly reduce the CDF for many core damage sequences, especially station blackout sequences. The generic PRA model includes a configuration that takes credit for this capability.
The AC independent system is modeled using a diesel driven pump that takes a suction from the CST. It shares an injection path with a loop of LPCI, incorporating a local manual action to open the injection valve. This system is started and aligned manually. In order for it to be successful, the reactor vessel must be depressurized and the containment vent must operate successfully. In order to assure the condition of containment venting, the vent model was changed so that it could be operated using only DC electrical power.

The AC independent low pressure injection model is not used as an early injection system for LOCA, SORV, or ATWS events. It is assumed that the local manual actions that must be taken to align the system could not be successful within the timeframe needed for early injection.

C.4.4 Additional Independent AC Power Source Model

Some BWRs have additional capability to provide AC power to some of their plant equipment during station blackout events. Examples of these are SBO diesels or black start combustion turbine generators. In order for this capability to significantly reduce the CDF at a plant, it must be independent of the station's emergency diesel generators to the extent that modeling common cause between the extra AC power source and the EDGs would not be warranted.

The additional independent AC power source in the generic model has the capacity to power one division of safety related electrical equipment. Alignment of the system must be performed manually, and it can be accomplished within a short period following the SBO such that early injection can be performed successfully. This last assumption has a small non-conservative effect for LBLOCA and MBLOCA events, however the contribution of LBLOCA SBO and MBLOCA SBO events is so small that the effect is negligible.

C.4.5 Offsite Power Configuration

There are several different configurations for connecting to the offsite power grid at the various BWR plants. All of them, however, must comply with GDC-16 requirements. The relevant attribute is that there are two independent means of providing offsite AC power to the safety related buses. This section provides an overview of different configurations and their affect on the PRA results.

The generic PRA model has an offsite power model that has the following attributes:

- Main switchyard with multiple connections to a 345 kV grid
- Alternate switchyard with a single connection to a 69 kV grid
- During operation, all onsite power is provided by the main transformer
- Following a turbine trip, transfer relaying is required to supply offsite power
- Interlocks prevent bus connections to multiple sources, either onsite or offsite
- DC power is required to transfer the power source

There are three ways that offsite power can be lost at a plant. These are called "Plant Centered", "Grid Centered", and "Weather Related" events. Each one has a different potential for recovery of offsite power. Additionally, the connection to offsite power can be lost following a plant transient because of electrical transfer failures.

Plant centered events are caused by faults or failures that may occur between the main switchyard and the main or startup transformer while the plant is operating. For this type of event, the alternate switchyard is a completely independent offsite power source.

Grid centered events are caused by instabilities or breakdown of the 345 kV grid. The alternate switchyard is assumed to be completely dependent on the 345 kV grid, so if the event is grid centered, the alternate source is not available. The conditional loss of offsite power events (LOOP given LOCA and LOOP given transient) are modeled as grid centered events.

Weather related events are caused by severe weather in the vicinity of the plant. In this case, the alternate switchyard is also disabled with certainty by any severe weather events that cause a loss of the main switchyard.

When an accident or transient event occurs at the plant, the main turbine is tripped. This causes a transfer of the AC power source to the offsite power grid. If the equipment that is required for this transfer fails to operate correctly, the event is treated as a plant centered loss of offsite power in the generic model. This portion of the model includes the transfer breakers, under-voltage relays, and interlocks. Note that these events are not the same as conditional LOOP events mentioned above, because they affect only plant equipment and not the grid as a whole.

The transfer from the main transformer to offsite power sources requires several breakers to change state. These breakers use 125 VDC for control power, so if the batteries fail or are not available, the transfer cannot occur. If the initiating event is a loss of DC power, it is assumed that the operators can attempt a manual reconfiguration of offsite power prior to tripping the main generator.

Figure C.4-3 shows the generic model logic for loss of offsite power events.

Review of the generic model cutsets indicates that the specific configuration of the connection from the switchyard(s) to the plant power system is not critical to the results, so long as it is reliable and robust. The generic model contains the same types of failure modes that are present at all plants, so it is considered applicable across the fleet.

Some plants have multiple main switchyard configurations. These configurations can introduce a different dominant LOOP initiating event, the partial LOOP. This configuration has a significant effect only on the plant centered loss of offsite power events. For plants with this configuration, the frequency of the partial LOOP tends to be higher that a full LOOP. Based on the survey responses, plants with this configuration have a partial LOOP frequency that looks similar to the full LOOP frequency used in the generic model. If there is a partial LOOP, the plant retains approximately half of its offsite powered equipment available for mitigation, therefore the CCDP for these sequences would be smaller. It is appropriate, for the purposes of this analysis, to use a single full LOOP frequency in the generic model to cover both full and partial LOOP events.

C.4.6 Benchmark Summary

The CCDP estimated for different plant configurations were compared to plant specific results for several BWROG member plants. The comparison was made using the LOOP and MBLOCA event trees. The response to the LOOP initiating event provides most of the CDF difference for the changes described in this report. The MBLOCA comparison was done chiefly to validate that the BWR 6 configuration gave appropriate results.

Figure C.4-1 shows the comparison results for the LOOP event trees. With two exceptions, the generic PRA model results are consistent with all plant configurations. The two exceptions are different because of conservative modeling assumptions associated with the recovery of failed onsite power equipment. This analysis shows that the generic model can accurately estimate a plants response to LOOP events, as long as the key assumptions are consistently applied. The variations within the plant configuration groups are mainly a function of station battery life. If the plant is able to cope longer without AC power, the non-recovery factors that can be applied to station blackout sequences would be lower. The range of CCDP values seen are consistent with the time dependence on the offsite power non-recovery factors used in the generic model.

Figure C.4-2 shows the comparison results for the MBLOCA event trees. Once again, the comparison holds for most plants. The magnitude of the two exceptions is not sufficient to affect the overall model results. Because the conclusions drawn in this report are not as sensitive to the MBLOCA model, no further breakdown in configuration type is necessary.

This benchmark study, combined with the sensitivity and uncertainty analyses performed, demonstrate that the generic BWR PRA model is sufficient to accurately predict plant specific risk changes to the Option 3 changes described in this report.

C.5 RISK CALCULATION FOR PLANT CHANGES

This section outlines the changes that were made to the model to address the Option 3 modifications described in this report. They are first addressed individually in sections C.5.1 through C.5.5, then the combination of changes are discussed in C.5.6.

Each change is evaluated in the following manner:

(1) Calculate the CDF and LERF for the base (unaltered case)
(2) Calculate the CDF and LERF following the change
(3) Delete any contribution for LBLOCA-LOOP from the changed CDF calculated in (2)
(4) Subtract the results of (1) from (3). This represents the CDF improvements caused by the change that offsets a portion of the assumed 10^{-6} increase associated with eliminating the LBLOCA-LOOP requirement
(5) Subtract the LERF results of (1) from (2). This is an approximation of the offset in LERF

In this report, any values less than 5×10^{-10} will be reported as "ε" and should be considered negligible. Offsets are assumed to be risk decreases. Any change that results in a risk increase will be presented in parentheses.

C.5.1 Optimize Diesel Generator Loads

In the description of the changes, it was postulated that LPCS or some LPCI pumps could be eliminated as equipment that is automatically loaded on to the diesel generators, while battery chargers could be added. This provides a trade-off. If the chargers are automatically loaded on the emergency buses, the operator actions associated with DC load shed or re-start of the chargers would be rendered unnecessary. Low pressure ECCS pumps that are not automatically loaded following a loss-of-offsite power could be subsequently added manually in the event that other systems were unable to provide core injection.

The PRA model was modified to address these configurations. The 125 V DC model was changed such that if offsite power were lost, the operator action to load the chargers would be eliminated from the model. The LPCS model was changed by adding an operator action to manually start these pumps following a loss of offsite power. The LCPI model was changed in a similar manner, but the operator action was only included for two of the four LPCI pumps. Either the LPCS or the LPCI change can be made, but not both.

Eight plant configurations were evaluated for each of the optimizations. These are presented below. In the case of the BWR6 model, the elimination of the automatic start of LPCS following LOOP includes one division of LPCI. This is to balance the load on the EDGs.

Configuration	CDF Offset for Disabling LPCS Automatic Start Following LOOP	CDF Offset for Disabling LPCI Automatic Start Following LOOP
BWR 3/4 Base	$1.2x10^{-8}$	$1.4x10^{-8}$
BWR 3/4 with AC Independent LP Injection	$3.0x10^{-8}$	$3.0x10^{-8}$
BWR 3/4 with Independent AC Power Source	$3x10^{-9}$	$4x10^{-9}$
BWR 5/6 Base	$1.3x10^{-8}$	$1.3x10^{-8}$
BWR 5/6 with AC Independent LP Injection	$3.3x10^{-8}$	$3.3x10^{-8}$
BWR 5/6 with Independent AC Power Source	$3x10^{-9}$	$4x10^{-9}$
BWR 3/4 with LPCI Loop-Select and AC Independent LP Injection	$2.0x10^{-8}$	$2.1x10^{-8}$
BWR 3/4 with LPCI Loop-Select and Independent AC Power Source	$2x10^{-9}$	$3x10^{-9}$

All LERF offsets associated with this change are negligible.

C.5.2 Start One Loop of RHR in Suppression Pool Cooling Mode

Another postulated change was to start one loop of RHR in suppression pool cooling mode rather than LPCI mode. With this change, the operators would not need to start suppression pool cooling for scenarios in which the pool water temperature increased above a given value. If the other core injection systems were not available, the loop selected for automatic SPC could be manually aligned to the LPCI mode instead.

In the PRA model, eliminating the operator action from one loop of RHR for suppression pool cooling simulated this change. An operator action to manually start the loop in LPCI mode was added to the LPCI model for the affected loop. The other loop retained the original configuration.

Eight plant configurations were evaluated for this change. These are presented below.

Configuration	CDF Offset for Automatically Starting One Loop of RHR in SPC Mode
BWR 3/4 Base	4.1×10^{-8}
BWR 3/4 with AC Independent LP Injection	6×10^{-9}
BWR 3/4 with Independent AC Power Source	3.6×10^{-8}
BWR 5/6 Base	3.7×10^{-8}
BWR 5/6 with AC Independent LP Injection	2×10^{-9}
BWR 5/6 with Independent AC Power Source	3.5×10^{-8}
BWR 3/4 with LPCI Loop-Select and AC Independent LP Injection	3×10^{-9}
BWR 3/4 with LPCI Loop-Select and Independent AC Power Source	3.5×10^{-8}

All LERF offsets associated with this change are negligible.

C.5.3 Eliminate LPCI Loop-Select Logic

In the past, LPCI Loop-Select has been removed from BWRs through elaborate redesign and reconfiguration. The change proposed here is to eliminate the feature in a straightforward manner. The crosstie valve would be locked closed, and the logic would be disabled. If there is a LBLOCA in one of the recirculation lines, all of the ECCS flow through that line is assumed to be lost.

For this change, only two configurations needed to be evaluated.

Configuration	CDF Offset for Eliminating LPCI Loop-Select Logic
BWR 3/4 with LPCI Loop-Select and AC Independent LP Injection	(1×10^{-9})
BWR 3/4 with LPCI Loop-Select and Independent AC Power Source	(2×10^{-9})

All LERF offsets associated with this change are negligible.

Even though this change presents a very small increase in CDF, it is noteworthy that the final configuration is still less than the base case for the "LPCI Mod" plants. Therefore, eliminating LPCI in this manner seems to be more risk beneficial than what has been done in the past in a deterministic manner.

C.5.4 Risk Offsets Associated with Diesel Generator Reliability Changes

It is difficult to predict the actual reduction in unavailability and unreliability of the diesel generators associated with the proposed changes. The effect is not expected to be large, but even small improvements can substantially offset the postulated risk increase arising from the assumption that all LBLOCA/LOOP events result in core damage.

A reasonable reduction in unavailability and failure rates as a result of the changes described in this report would be 10%. For example, the average unavailability of EDGs is 1.01×10^{-2}. Assuming an average annual required availability time of 8760 hours, the total unavailability per EDG is about 88 hours. Nearly a 10% reduction in unavailability would be realized if one shift (8 hours) of down time per year was avoided. Draft NEI 99-02 (Reference 5) reports the probability of an EDG failure to start on demand is 1.1×10^{-2}. This translates into about one failure per plant every 3 years, assuming the generators are started monthly and the average BWR has 2.5 diesels. Extending this to one failure per 3.3 years results in a 10% reduction of the failure probability.

The basic events that are assumed to be affected by the changes are:

- EDG Fails to Start
- EDG Fails to Run
- EDG Unavailability
- EDC Common Cause

Rather than specifying an impact on the parameter changes and re-quantifying the model, the risk impact was evaluated examining a reasonable range of improvements of the affected parameters. The range chosen was from a 5% reduction to a 15% reduction. Figure C.5-1 shows the effect of varying EDG reliability and availability parameters for the BWR 4. The results are fairly linear in the range of improvement expected. Similar results would be obtained for the other configurations.

The following table presents the CDF offsets for eight configurations with a 10% EDG reliability and availability improvement.

Configuration	CDF Offset Assuming a 10% Reduction in EDG Unavailability and Failure Rates
BWR 3/4 Base	4.48×10^{-7}
BWR 3/4 with AC Independent LP Injection	3.48×10^{-7}
BWR 3/4 with Independent AC Power Source	5.2×10^{-8}
BWR 5/6 Base	4.72×10^{-7}
BWR 5/6 with AC Independent LP Injection	3.37×10^{-7}
BWR 5/6 with Independent AC Power Source	5.4×10^{-8}
BWR 3/4 with LPCI Loop-Select and AC Independent LP Injection	3.41×10^{-7}
BWR 3/4 with LPCI Loop-Select and Independent AC Power Source	4.9×10^{-8}

Plants that rely more on their EDGs (i.e. those plants with a less reliable local grid) would benefit more from these changes. Plants that have implemented a backup to their onsite AC power sources do not benefit as much from this particular change.

LERF quantification was performed in combination with removing the anticipatory start.

C.5.5 Risk Offsets Associated with Removing Anticipatory Start of EDGs

It is similarly difficult to predict the reduction of the operator action failure rates. Once again, though, a moderate decrease of 10% is expected if the proposed changes are made. This only applies to actions in the model that share dependence with the actions to secure a diesel generator that has spuriously started. These are defined as those actions taken outside the control room, that need to happen within one hour of the initiating event.

The operator actions in the generic model that are assumed to be affected are:

- Failure to locally align control panels and MCCs
- Failure to recover from electrical bus testing following the transient
- Failure to operate breakers locally & manually
- Failure to bypass failed instrument air components
- Failure to align instrument air to pneumatic equipment located in the drywell
- Failure to manually operate containment vent valves
- Failure to recover the PCS within 4 hours
- Failure to align service water injection

The following table presents the CDF offsets for eight configurations with a 10% improvement in the operator error rates for the actions listed above.

Configuration	CDF Offset Assuming a 10% Reduction in Human Error Probabilities
BWR 3/4 Base	3.09×10^{-7}
BWR 3/4 with AC Independent LP Injection	4.00×10^{-7}
BWR 3/4 with Independent AC Power Source	2.27×10^{-7}
BWR 5/6 Base	2.13×10^{-7}
BWR 5/6 with AC Independent LP Injection	3.03×10^{-7}
BWR 5/6 with Independent AC Power Source	1.32×10^{-7}
BWR 3/4 with LPCI Loop-Select and AC Independent LP Injection	3.10×10^{-7}
BWR 3/4 with LPCI Loop-Select and Independent AC Power Source	2.22×10^{-7}

LERF quantification was performed in combination with changing practices that improve diesel generator reliability and availability.

C.5.6 Combinations of Option 3 Changes

It is expected that plants that implement the changes in this report will select one or more of the changes. Therefore, the combinations of changes were evaluated to get an overall impact. Table C.5-1 shows the offsets from combining changes for the BWR 3/4 model; Table C.5-2 shows the offsets for the BWR 5/6; and Table C.5-3 shows the offsets for the BWR 4 with LPCI Loop-Select logic. As can be seen in the tables, combining the changes provides a greater offset.

C.6 KEY MODEL ATTRIBUTES

Several assumptions were made when the generic model was created. The purpose of this section is to describe those assumptions and to assess the overall affect of those assumptions on the model results. The assessments in this section consider the affect on the Option 3 changes described in this report.

Table C.6-1 contains the results of all of the assessments.

Several sensitivity analyses were performed to confirm the affects of key model attributes. These analyses are presented in section C.6.2.

C.6.1 Plant Specific Validations

The following model attributes and assumptions must be validated by a utility that wishes to implement the changes described in this report.

1. HPCI (if present) is AC independent. This does not apply to any room cooling that may be required for HPCI long-term operation.
2. HPCI (if present) can be used to depressurize the reactor without uncovering the core in non-ATWS sequences. Note: HPCI is not applicable to LBLOCA events.
3. RCIC (if present) does not require room cooling for the time period considered.
4. In a BWR 5/6, one division of AC powers LPCS and one loop of LPCI, and the other division powers two loops of LPCI.
5. Either LPCI or LPCS must be able to take suction from the CST (or other source of water external to containment) for long-term cooling.
6. One LPCI pump can be used for both long-term injection and for suppression pool cooling through the use of operator actions.
7. Service water can be used for injection into the reactor vessel for long-term cooling. This function must be available using either onsite power alone or offsite power alone.
8. MSIVs isolate on low water level, low steam line pressure, high seam tunnel temperature, high main steam line flow, and low condenser vacuum. Loss of offsite power also results in closure of the MSIVs.
9. Containment venting must be possible and procedures available.
10. RPS reliability must be consistent with INEEL/EXT-98-00670, General Electric RPS Unavailability, 1984-1995.
11. Each division of DC power has a dedicated charger. There is a shared "swing" charger for each DC system.
12. Plants need to review the success criteria listed in Appendix C for applicability. See Appendix C for additional guidance on this review.
13. Using information from existing pipe monitoring programs, plants need to confirm that there is no evidence of recirculation piping degradation that would result in a LBLOCA frequency in those sections significantly greater than $1x10^{-4}$.
14. Plant must have the ability to provide long-term core cooling with systems that take their suction from outside the containment.

15. Plant must demonstrate actions and programs consistent with SOER 99-01 recommendations.
16. Plants must review their fire areas. If a fire in a single area, other than the main control room, can cause both a loss of offsite power and disable a portion of the LPCI or LPCS systems, a plant specific fire risk analysis must be performed in order to make the changes to either of these systems.
17. Plants must review their flood areas. If a flood in a single area can cause both a loss of offsite power and disable a portion of the LPCI or LPCS systems, a plant specific flood risk analysis must be performed in order to make the changes to either of these systems.
18. Plants must review the seismic ruggedness of their recirculation loops and EDGs. If the ruggedness of the EDGs is greater than that of the recirculation loops, a plant specific seismic risk analysis must be performed.

C.6.2 Sensitivity Analyses

During the evaluation of the relevant model attributes, there were several items that could not be easily dispositioned. This section provides quantitative analyses of these attributes. Each section provides a conclusion concerning the degree to which each implementing plant needs to address the issue on a plant-specific basis.

The quantitative results of the sensitivity analyses in this section cannot be directly compared to the quantitative results presented in other sections of this appendix. The reason is that these sensitivities do not include the assumption that all LBLOCA-LOOP initiating events result in core damage.

C.6.2.1 RHR Pump Room Cooling

The generic BWR PRA model makes the assumption that the RHR pumps can be effectively cooled by natural circulation of air in the reactor building. This assumption may not be valid at all plants. A sensitivity analysis was performed to determine the affect of this assumption on the changes outlined in this report.

This sensitivity analysis was conducted by adding the LPCS room cooling logic to each RHR pump's failure logic. The BWR 4 base model and the BWR 4 base model with LPCI removed from onsite power (Section C.5.1) were re-quantified to evaluate the impact of the room cooling assumption.

Assumption	CDF Decrease	
	LPCI Does Not Start With On Offsite Power	Increased DG Reliability
RHR Does Not Require Room Cooling	1.2×10^{-8}	4.54×10^{-7}
RHR Requires Room Cooling	1.0×10^{-8}	4.86×10^{-7}

C.6.2.2 Early Service Water Injection

The generic BWR PRA model makes the assumption that service water injection can be aligned as the only low pressure injection source in transient events. This assumption may not be valid for all plants due to the plant specific nature of the operator actions necessary to complete the alignment. A sensitivity analysis was performed to determine the affect of this assumption on the changes outlined in this report.

This sensitivity analysis was performed by removing early SW injection (i.e., assumed failed) from all applicable event trees. The BWR 4 base model and the BWR 4 base model with LPCS removed from onsite power (Section C.5.1) were re-quantified to evaluate the impact of the room cooling assumption.

Assumption	CDF Decrease	
	LPCS Does Not Start With On Offsite Power	Increased DG Reliability
Early Service Water Injection Capability	6×10^{-9}	4.54×10^{-7}
No Early Service Water Injection Capability	7×10^{-9}	4.53×10^{-7}

C.6.2.3 Service Water Injection Power Source

The generic model assumes that the pumps used for service water injection can be powered by onsite power, thus the function is available in LOOP sequences. Because of the various service water configurations at plants, a sensitivity analysis was performed to determine the importance of this system during LOOP sequences.

This sensitivity analysis was performed by forcing the Service Water Injection to be unavailable in all LOOP sequences. The BWR4 base model and the BWR 4 base model with LPCS removed from onsite power (Section C.5.1) were re-quantified to evaluate the impact of this assumption.

Assumption	CDF Decrease	
	LPCS Does Not Start With On Offsite Power	Increased DG Reliability
Early Service Water Injection Capability	6×10^{-9}	4.54×10^{-7}
No Early Service Water Injection Capability	1×10^{-8}	4.82×10^{-7}

C.6.2.4 Automatic Function of ADS

The generic model assumes that the operators, per EPG implementing procedures, will inhibit ADS logic. Past studies by the BWROG have found that the CDF effect of this procedure is plant specific. A sensitivity analysis was performed to investigate this affect on the changes recommended in this report.

This sensitivity analysis was performed by setting the manual depressurization operator action basic event to FALSE. This simulates the reliability of the automatic initiation logic. The BWR 4 base model and the BWR 4 base model with LPCS removed from onsite power (Section C.5.1) were re-quantified to evaluate the impact of the assumption that automatic ADS is always disabled.

Assumption	CDF Decrease	
	LPCS Does Not Start With On Offsite Power	Increased DG Reliability
Early Service Water Injection Capability	$6x10^{-9}$	$4.54x10^{-7}$
No Early Service Water Injection Capability	$-4x10^{-9}$	$4.52x10^{-7}$

C.6.2.5 Service Water System Reliability

There are many different service water arrangements at BWRs. The range from a single common system to several systems dedicated for different purposes. The generic model uses a configuration that provides a high degree of intra-system dependence, and a high degree of dependence for the supported systems. In the PRA, this amounts to a limiting configuration.

The conclusions of this report are not sensitive to changes in the reliability of the service water system, so long as the system does not fail altogether. A sensitivity analysis was performed to confirm this. The specific service water configuration at a plant that is different from the generic model would not affect the conclusions of this report.

Assumption	CDF Decrease	
	LPCS Does Not Start With On Offsite Power	Increased DG Reliability
Base Service Water System Reliability	$6x10^{-9}$	$4.54x10^{-7}$
Service Water Pump Failure Rate Increased by 500%	$7x10^{-9}$	$4.62x10^{-7}$

C.6.2.6 Battery Cells Shared Between 125 V and 250 V DC Systems

The generic model assumes that the 125 V and 250 V DC systems are independent. Some plants use a single battery set to provide both functions. A sensitivity analysis was performed to address this feature.

This sensitivity analysis was performed by applying the same common cause basic event to all battery division (both 125 V and 250 V batteries) fault trees. This simulates the sharing of cells between the different DC systems. The BWR 4 base model and the BWR 4 base model with LPCS removed from onsite power (Section C.5.1) were re-quantified to evaluate the impact of the assumption that the DC systems are independent.

Assumption	CDF Decrease	
	LPCS Does Not Start With On Offsite Power	Increased DG Reliability
125 V and 250 V DC systems are independent	$6x10^{-9}$	$4.54x10^{-7}$
125 V and 250 V DC batteries share some cells	$6x10^{-9}$	$4.53x10^{-7}$

C.6.2.7 Initiating Event Frequencies

The generic model uses the mean values of the initiating event frequencies from NUREG/CR-5750. An additional sensitivity was performed to address the variability of plant specific initiating event frequencies. In this analysis, the LOCA and LOOP frequencies were raised to the highest values from the BWROG survey. These values were all greater than the 95th percentile from NUREG/CR-5750 for LOCA and LOOP events. All of the rest of the initiators were raised to the 95th percentile reported in NUREG/CR-5750.

The results of this sensitivity were consistent the conclusions of this report. The change in CDF associated with the plant configuration changes was similar to the main analysis. The CDF decrease associated with the EDG improvements scaled directly with the increase in the LOOP initiating event frequencies.

Assumption	CDF Decrease	
	LPCS Does Not Start With On Offsite Power	Increased DG Reliability
Initiating Event Frequencies Based on NUREG/CR-5750	$6x10^{-9}$	$4.54x10^{-7}$
Maximum Initiating Event Frequencies	$1x10^{-8}$	$1.04x10^{-6}$

C.7 CONCLUSIONS

This analysis demonstrates that a generic BWR risk analysis, when accompanied by the appropriate sensitivity analyses, can provide the necessary insights to justify changes to plant configurations under the Option 3 and Regulatory Guide 1.174 guidelines.

All of the changes evaluated in this appendix result in a CDF change that is either a risk benefit, or is negligible. Therefore, when these changes are combined with the assumption that all LBLOCA-LOOP events lead directly to core damage (Appendix B demonstrates that this assumption is very conservative), it can be clearly demonstrated that the CDF and LERF increases are well within the Regulatory Guide 1.174 guidelines.

The most beneficial of the changes, in terms of CDF and LERF, are those that affect the availability and reliability of the EDGs.

Table C.3-1 Base Model Success Criteria

Function	Transient	Transient with SORV	Loss of Offsite Power	Loss of Service Water[1]	Large LOCA
High Pressure Injection	1 Feedwater RCIC HPCI	1 Feedwater RCIC HPCI	RCIC HPCI	n/a	n/a
Low Pressure Injection	1 LPCS Loop 1 LPCI Pump 1 Condensate Pump 1 RHR Service Water Pump 1 Fire Water Pump[3]	1 LPCS Loop 1 LPCI Pump 1 Condensate Pump 1 RHR Service Water Pump 1 Fire Water Pump[3]	1 LPCS Loop 1 LPCI Pump 1 RHR Service Water 1 Fire Water Pump[3]	n/a	1 LPCS Loop 2 LPCI Pumps
Depressurization	Main Steam 1 SRV	Main Steam 1 SRV RCIC HPCI	1 SRV	1 SRV	n/a
Containment Heat Removal	Main Steam 1 SPC 1 SDC Vent	Main Steam 1 SPC 1 SDC Vent	1 SPC 1 SDC Vent	Vent	1 SPC 1 Containment Spray Vent
Long-Term Core Cooling	Main Steam + 1 Condensate 1 SDC 1 SPC + RCIC 1 SPC + HPCI 1 SPC + 1 LPCS 1 SPC + 1 LPCI Vent + 1 Condensate Vent + 1 RHR Service Water Vent + 2 CRD Vent + 1 LPCS from CST Vent + 1 LPCI from CST Vent + 1 Fire Water Pump[3]	Main Steam + 1 Condensate 1 SDC 1 SPC + 1 LPCS 1 SPC + 1 LPCI Vent + 1 Condensate Vent + 1 RHR Service Water Vent + 1 LPCS from CST Vent + 1 LPCI from CST Vent + 1 Fire Water Pump[3]	1 SDC 1 SPC + RCIC 1 SPC + HPCI 1 SPC + 1 LPCS 1 SPC + 1 LPCI Vent + 1RHR Service Water Vent + 2 CRD Vent + 1 LPCS from CST Vent + 1 LPCI from CST Vent + 1 Fire Water Pump[3]	Vent + 1 LPCS from CST Vent + 1 LPCI from CST Vent + 1 Fire Water Pump[1]	1 SPC + 1 LPCS 1 SPC + 1 LPCI 1 Cont Spray + 1 LPCS 1 Cont Spray + 1 LPCI Vent + 1 Condensate Vent + 1 RHR Service Water Vent + 1 LPCS from CST Vent + 1 LPCI from CST Vent + 1 Fire Water[3]
Reactivity Control[2]	n/a	n/a	n/a	n/a	n/a
Pressure Relief	Main Steam 1 SRV Opens	n/a	1 SRV Opens	1 SRV Opens	n/a
Pressure Suppression	n/a	SRV Tailpipe Intact	SRV Tailpipe Intact if SORV	SRV Tailpipe Intact if SORV	7 of 8 Vacuum Breakers Seat

Table C.3-1 Base Model Success Criteria (continued)

Function	Medium LOCA	Small LOCA	ATWS (Transient)	ATWS (Loss of Offsite Power)	ATWS (Large or Medium LOCA)
High Pressure Injection	HPCI	1 Feedwater RCIC HPCI	1 Feedwater HPCI	HPCI	n/a
Low Pressure Injection	1 LPCS 1 LPCI	1 LPCS 1 LPCI	1 LPCS 1 LPCI	1 LPCS 1 LPCI	n/a
Depressurization	HPCI 1 SRV	Main Steam RCIC HPCI 1 SRV	Main Steam 1 SRV	1 SRV	n/a
Containment Heat Removal	1 SPC 1 Containment Spray Vent	1 SPC 1 Containment Spray Vent	1 SLC + 1 SPC 1 SLC + Vent	1 SLC + 1 SPC 1 SLC + Vent	n/a
Long-Term Core Cooling	1 SPC + 1 LPCS 1 SPC + 1 LPCI 1 Cont Spray + 1 LPCS 1 Cont Spray + 1 LPCI Vent + 1 Condensate Vent + 1 Service Water Vent + 1 LPCS from CST Vent + 1 LPCI from CST Vent + 1 Fire Water Pump[3]	Main Steam + 1 Condensate 1 SPC + RCIC 1 SPC + HPCI 1 SPC + 1 LPCS 1 SPC + 1 LPCI 1 Cont Spray + RCIC 1 Cont Spray + HPCI 1 Cont Spray + 1 LPCS 1 Cont Spray + 1 LPCI Vent + 1 Service Water Vent + 1 LPCS from CST Vent + 1 LPCI from CST Vent + 1 Fire Water Pump[3]	Main Steam + 1 Condensate Main Steam + HPCI 1 SPC + HPCI 1 SPC + 1 LPCS 1 SPC + 1 LPCI Vent + 1 Service Water Vent + 1 LPCS from CST Vent + 1 LPCI from CST Vent + 1 Fire Water Pump[3]	1 SPC + HPCI 1 SPC + 1 LPCS 1 SPC + 1 LPCI Vent + 1 Service Water Vent + 1 LPCS from CST Vent + 1 LPCI from CST Vent + 1 Fire Water Pump[3]	n/a
Reactivity Control[2]	n/a	n/a	RPS ARI Manual SCRAM Recirc Pump Trip + Level Control + ADS Inhibit AND 1 SLC Main Steam Available 2 SLC Main Steam Unavailable	RPS ARI Manual SCRAM Recirc Pump Trip + Level Control + ADS Inhibit AND 1 SLC Main Steam Available 2 SLC Main Steam Unavailable	RPS ARI Manual SCRAM
Pressure Relief	n/a	1 SRV Opens	1 SRV Opens	1 SRV Opens	n/a
Pressure Suppression	7 of 8 Vacuum Breakers Seat	SRV Tailpipe Intact	n/a	n/a	n/a

Table C.3-1 Base Model Success Criteria (continued)

Notes for Success Criteria Table:

1) Short-term injection systems were not asked in the Loss of Service Water event tree. These failures are assumed to generate the same sequences as short-term injection failures in the Transient event tree. Because the frequency of a Loss of Service Water event is much less than that of a Transient, the results of the model are virtually unchanged by making this simplification.

2) Reactivity Control is addressed explicitly in the ATWS event trees. It is simply shown for illustrative purposes on the other event trees.

3) Fire water is not available in many plant configurations. It is only included in the cases in which an AC Independent Low Pressure injection system is indicated.

Table C.3-2 Initiating Event Frequencies and Distributions

Initiator	Mean	95th	Lognormal Error Factor
Large LOCA	3.00E-05	1.00E-04	6.1
Medium LOCA	4.00E-05	1.00E-04	3.2
Small LOCA	5.00E-04	1.00E-03	2.3
Leak	6.20E-03	1.20E-02	2.2
Stuck/Inadvertent Open Relief Valve	4.60E-02	7.10E-02	1.6
Loss of Offsite Power	4.60E-02	1.10E-01	3.0
Main Steam Isolation	2.90E-01	3.90E-01	1.4
Loss of Feedwater	8.50E-02	2.50E-01	4.4
Transient with Condenser Available	1.50E+00	2.50E+00	1.8
Loss of a Single DC Bus	2.10E-03	5.40E-03	3.4
Loss of a Single AC Division	1.90E-02	2.80E-02	1.5
Loss of Instrument Power	4.80E-03	9.70E-03	2.3
Loss of Service Water (any system)	9.70E-04	2.50E-03	3.4
Loss of Instrument Air	2.90E-02	5.50E-02	2.1

Table C.3-3 Quantification Results

Plant Type	Configuration	Change	CDF	LERF
BWR 4	No AC Independent Injection No Diverse Onsite AC Power		9.45×10^{-6}	7.62×10^{-8}
		LPCS Does Not Start Automatically Following LOOP	9.45×10^{-6}	7.62×10^{-8}
		2 LPCI Pumps Do Not Start Automatically Following LOOP	9.44×10^{-6}	7.63×10^{-8}
		1 Loop of RHR Automatically Starts in SPC Mode	9.41×10^{-6}	7.62×10^{-8}
		2 LPCI Pumps Do Not Start Automatically and 1 Loop of RHR Automatically Starts in SPC Mode	9.40×10^{-6}	7.62×10^{-8}
	AC Independent Injection No Diverse Onsite AC Power		6.83×10^{-6}	4.45×10^{-8}
		LPCS Does Not Start Automatically Following LOOP	6.81×10^{-6}	4.46×10^{-8}
		2 LPCI Pumps Do Not Start Automatically Following LOOP	6.80×10^{-6}	4.45×10^{-8}
		1 Loop of RHR Automatically Starts in SPC Mode	6.83×10^{-6}	4.46×10^{-8}
		2 LPCI Pumps Do Not Start Automatically and 1 Loop of RHR Automatically Starts in SPC Mode	6.80×10^{-6}	4.46×10^{-8}
	Diverse Onsite AC Power		4.63×10^{-6}	6.51×10^{-8}
		LPCS Does Not Start Automatically Following LOOP	4.63×10^{-6}	6.51×10^{-8}
		2 LPCI Pumps Do Not Start Automatically Following LOOP	4.62×10^{-6}	6.51×10^{-8}
		1 Loop of RHR Automatically Starts in SPC Mode	4.59×10^{-6}	6.51×10^{-8}
		2 LPCI Pumps Do Not Start Automatically and 1 Loop of RHR Automatically Starts in SPC Mode	4.59×10^{-6}	6.51×10^{-8}

Table C.3-3 Quantification Results (continued)

Plant Type	Configuration	Change	CDF	LERF
BWR 6	No AC Independent Injection No Diverse Onsite AC Power		7.64×10^{-6}	6.32×10^{-8}
		LPCS Does Not Start Automatically Following LOOP	7.63×10^{-6}	6.32×10^{-8}
		2 LPCI Pumps Do Not Start Automatically Following LOOP	7.63×10^{-6}	6.32×10^{-8}
		1 Loop of RHR Automatically Starts in SPC Mode	7.61×10^{-6}	6.32×10^{-8}
		1 LPCS and 1 LPCI Pumps Do Not Start Automatically and 1 Loop of RHR Automatically Starts in SPC Mode	7.60×10^{-6}	6.33×10^{-8}
		2 LPCI Pumps Do Not Start Automatically and 1 Loop of RHR Automatically Starts in SPC Mode	7.60×10^{-6}	6.32×10^{-8}
	AC Independent Injection No Diverse Onsite AC Power		4.86×10^{-6}	3.73×10^{-8}
		LPCS Does Not Start Automatically Following LOOP	4.83×10^{-6}	3.73×10^{-8}
		2 LPCI Pumps Do Not Start Automatically Following LOOP	4.83×10^{-6}	3.73×10^{-8}
		1 Loop of RHR Automatically Starts in SPC Mode	4.86×10^{-6}	3.73×10^{-8}
		1 LPCS and 1 LPCI Pumps Do Not Start Automatically and 1 Loop of RHR Automatically Starts in SPC Mode	4.83×10^{-6}	3.73×10^{-8}
		2 LPCI Pumps Do Not Start Automatically and 1 Loop of RHR Automatically Starts in SPC Mode	4.83×10^{-6}	3.73×10^{-8}

Table C.3-3 Quantification Results (continued)

Plant Type	Configuration	Change	CDF	LERF
BWR 6	Diverse Onsite AC Power		2.63×10^{-6}	5.29×10^{-8}
		LPCS Does Not Start Automatically Following LOOP	2.63×10^{-6}	5.29×10^{-8}
		2 LPCI Pumps Do Not Start Automatically Following LOOP	2.63×10^{-6}	5.29×10^{-8}
		1 Loop of RHR Automatically Starts in SPC Mode	2.60×10^{-6}	5.29×10^{-8}
		1 LPCS and 1 LPCI Pumps Do Not Start Automatically and 1 Loop of RHR Automatically Starts in SPC Mode	2.59×10^{-6}	5.29×10^{-8}
		2 LPCI Pumps Do Not Start Automatically and 1 Loop of RHR Automatically Starts in SPC Mode	2.59×10^{-6}	5.29×10^{-8}

Table C.3-3 Quantification Results (continued)

Plant Type	Configuration	Change	CDF	LERF
BWR 4 with LPCI Loop-Select	No AC Independent Injection No Diverse Onsite AC Power	[No plants in this category]		
BWR 4 with LPCI Loop-Select	AC Independent Injection No Diverse Onsite AC Power		6.71×10^{-6}	4.45×10^{-8}
		LPCS Does Not Start Automatically Following LOOP	6.69×10^{-6}	4.45×10^{-8}
		2 LPCI Pumps Do Not Start Automatically Following LOOP	6.69×10^{-6}	4.45×10^{-8}
		1 Loop of RHR Automatically Starts in SPC Mode	6.71×10^{-6}	4.45×10^{-8}
		2 LPCI Pumps Do Not Start Automatically and 1 Loop of RHR Automatically Starts in SPC Mode	6.69×10^{-6}	4.45×10^{-8}
		Eliminate Loop Select	6.71×10^{-6}	4.45×10^{-8}
		Eliminate Loop Select and LPCS Does Not Start Automatically Following LOOP	6.69×10^{-6}	4.46×10^{-8}
		Eliminate Loop Select and 2 LPCI Pumps Do Not Start Automatically Following LOOP	6.69×10^{-6}	4.45×10^{-8}
		Eliminate Loop Select and 1 Loop of RHR Automatically Starts in SPC Mode	6.71×10^{-6}	4.46×10^{-8}
		Eliminate Loop Select and 2 LPCI Pumps Do Not Start Automatically and 1 Loop of RHR Automatically Starts in SPC Mode	6.69×10^{-6}	4.46×10^{-8}

Table C.3-3 Quantification Results (continued)

Plant Type	Configuration	Change	CDF	LERF
BWR 4 with LPCI Loop-Select	Diverse Onsite AC Power		4.55×10^{-6}	6.51×10^{-8}
		LPCS Does Not Start Automatically Following LOOP	4.55×10^{-6}	6.52×10^{-8}
		2 LPCI Pumps Do Not Start Automatically Following LOOP	4.55×10^{-6}	6.51×10^{-8}
		1 Loop of RHR Automatically Starts in SPC Mode	4.52×10^{-6}	6.52×10^{-8}
		2 LPCI Pumps Do Not Start Automatically and 1 Loop of RHR Automatically Starts in SPC Mode	5.51×10^{-6}	6.52×10^{-8}
		Eliminate Loop Select	4.55×10^{-6}	6.52×10^{-8}
		Eliminate Loop Select and LPCS Does Not Start Automatically Following LOOP	4.55×10^{-6}	6.52×10^{-8}
		Eliminate Loop Select and 2 LPCI Pumps Do Not Start Automatically Following LOOP	4.55×10^{-6}	6.52×10^{-8}
		Eliminate Loop Select and 1 Loop of RHR Automatically Starts in SPC Mode	4.52×10^{-6}	6.52×10^{-8}
		Eliminate Loop Select and 2 LPCI Pumps Do Not Start Automatically and 1 Loop of RHR Automatically Starts in SPC Mode	4.52×10^{-6}	6.52×10^{-8}

Table C.3-4 Plant Damage State Definitions

PDS	Conditional Probability of LERF	Description	Binning Rules
ATSLC	5.32×10^{-2}	Failure to scram accident with subsequent reactivity control. Inadequate coolant injection results in core damage in approximately an hour with the containment intact.	Transient, support, or LOCA initiator; scram failure; RPT successful; early or late SLC successful; level controlled to TAF; ADS inhibited; no RPV overfill; no RPV injection
ATWS	1.31×10^{-1}	Failure to scram accident with subsequent reactivity control failure.	A initiator; scram failure - or - Transient, support, or LOCA initiator; scram failure; RPT failure - or - Transient, support, or LOCA initiator; scram failure; RPT successful; early or late SLC successful; RPV level overfilled and boron diluted - or - Transient, support, or LOCA initiator; scram failure; RPT successful; early and late SLC failure - or - Transient, support, or LOCA initiator; scram failure; RPT successful; early or late SLC successful; level not controlled to TAF - or - Transient, support, or LOCA initiator; scram failure; RPT successful; early or late SLC successful; failure to inhibit ADS
DT-SBHP	3.94×10^{-4}	Station blackout with initial HPCI or RCIC injection. Battery depletion occurs in approximately 4 hours with core damage occurring at about 5 hours with the RPV at high pressure and the containment intact.	LOSP; emergency AC fails; no SORV; HPCI or RCIC success
DT-SBLP	1.84×10^{-4}	Station blackout with a stuck open relief valve (or the operators depressurize the plant prior to core damage) and initial HPCI or RCIC injection. Battery depletion occurs in approximately 4 hours with core damage occurring at about 5 hours with the RPV at low pressure and the containment intact.	LOSP; emergency AC fails; SORV; HPCI or RCIC success - or - T3C initiator; coincident LOSP; emergency AC fails; HPCI or RCIC success - or - LOSP; emergency AC fails; SORV; HPCI or RCIC success; depressuriztion before core damage
LOCA	9.06×10^{-3}	Loss of coolant accident with inadequate coolant injection. Core damage occurs in approximately an hour with the containment intact.	A, S1, or S2 initiator, or S3 w/leak unisolated; scram successful; no injection

Table C.3-4 Plant Damage State Definitions (continued)

PDS	Conditional Probability of LERF	Description	Binning Rules
PSS	6.21×10^{-1}	LOCA, or SORV, with failure of the pressure suppression system. Containment fails rapidly, failing injection systems due to harsh environmental effects. Core damage occurs in about an hour and a half with the RPV at low pressure.	Transient initiator; SORV; PSS failure - or - LOCA initiator; PSS failure
ST-SBHP	8.35×10^{-2}	Station blackout with no injection. Core damage occurs in approximately an hour with the RPV at high pressure and the containment intact.	LOSP; emergency AC fails; no SORV; no HP injection
ST-SBLP	2.77×10^{-2}	Station blackout with a stuck open relief valve and no injection. Core damage occurs in approximately an hour with the RPV at low pressure and the containment intact.	LOSP: emergency AC fails; SORV; no HP injection - or - T3C initiator; coincident LOSP; emergency AC fails; no HP injection
TPUV	0.0	Transient with a stuck open relief valve and loss of all coolant injection. Core damage occurs in approximately an hour with the RPV at low pressure and the containment intact.	Transient initiator; scram successful; SORV; no HP injection; no LP injection - or - T3C initiator; scram successful; no HP injection; no LP injection
TPUX	2.00×10^{-6}	Transient with a stuck open relief valve, loss of coolant injection and failure to depressurize the RPV to allow low pressure injection. Core damage occurs in approximately an hour with the RPV still at high pressure and the containment intact.	Transient or support initiator, or S3 w/leak isolated; scram successful; SORV; all HP injection fails; timely depressurization fails
TQUV	3.47×10^{-2}	Transient with loss of all coolant injection. Core damage occurs in approximately an hour with the RPV at low pressure and the containment intact.	Transient or support initiator, or S3 w/leak isolated; scram successful; no SORV; all HP injection fails; timely depressurization successful; all LP injection fails
TQUX	0.0	Transient with loss of coolant injection and failure to depressurize the RPV to allow low pressure injection. Core damage occurs in approximately an hour with the RPV at high pressure and the containment intact.	Transient or support initiator, or S3 w/leak isolated; scram successful; no SORV; all HP injection fails; timely depressurization fails

Table C.3-4 Plant Damage State Definitions (continued)

PDS	Conditional Probability of LERF	Description	Binning Rules
TWDT	0.0	Transient with initial ECCS injection but loss of all containment heat removal methods. Containment heat-up causes loss of injection sources due to loss of NPSH at about 4 hours. Core damage occurs at approximately 12.5 hours with containment intact.	Transient, support, or LOCA initiator; scram successful; initial HPCI or RCIC injection; SPC failure; depressurization failure - or - Transient, support, or LOCA initiator; scram successful; initial HPCI or RCIC injection; SPC failure; depressurization successful; SDC failure; LPCI or LPCS successful; containment vent failure - or - Transient, support, or LOCA initiator; scram successful; no initial HP injection; timely depressurization successful; initial LPCI or LPCS injection; SPC failure; SDC failure; containment vent failure - or - Transient, support, or LOCA initiator; scram successful; initial HPCI or RCIC injection; SPC failure; depressurization successful; SDC failure; all LP injection failure - or - Transient w/SORV initiator; scram successful; initial HPCI or RCIC injection; SPC failure; SDC failure; LP injection failure due to suppression pool heat up
TPWLT	0.0	Transient with SORV (or LOCA initiator), with initial successful injection but loss of all containment heat removal methods. Alignment of external low pressure injection sources continues core cooling until containment failure at approximately 27 hrs.	Transient w/SORV, or LOCA initiator; scram successful; initial HPCI or RCIC injection; SPC failure; depressurization successful; SDC failure; vent successful; all LP injection failure

Table C.3-4 Plant Damage State Definitions (continued)

PDS	Conditional Probability of LERF	Description	Binning Rules
TWLT	0.0	Transient w/injection but loss of containment heat removal. Alignment of ext. LP injection delays core damage until SRVs re-close due to high containment pressure, failing injection. Core damage in >24 hrs with containment intact but at high pressure.	Transient or support (except TTEC or TREC) initiator, or S3 w/leak isolated; scram successful; no SORV; SPC failure; depressurization failure - or - Transient or support (except TTEC or TREC) initiator, or S3 w/leak isolated; scram successful; no SORV; SPC failure; depressurization successful; SDC failure; containment vent failure - or - Transient, support, or LOCA initiator; scram successful; no SORV; initial HPCI or RCIC injection; SPC failure; depressurization successful; SDC failure; condensate or crosstie injection successful; containment vent failure

Table C.5-1 Combined CDF Offsets for BWR 3/4 Plant Configurations

Configuration	ECCS Change[1]	ECCS Offset	EDG Reliability Offset[2]	HRA Offset[3]	Both EDG and HRA Offset	LERF Offset (ECCS Change Only)	LERF Offset (All Changes)
No AC Independent Injection No Diverse Onsite AC Power			4.48×10^{-7}	3.09×10^{-7}	7.51×10^{-7}		1×10^{-9}
	LPCS	1.2×10^{-8}	4.58×10^{-7}	3.20×10^{-7}	7.61×10^{-7}	ε	1×10^{-9}
	LPCI	1.4×10^{-8}	4.59×10^{-7}	3.21×10^{-7}	7.62×10^{-7}	ε	1×10^{-9}
	RHR in SPC	4.1×10^{-8}	4.87×10^{-7}	3.45×10^{-7}	7.86×10^{-7}	ε	1×10^{-9}
	LPCI and RHR in SPC	5.3×10^{-8}	4.98×10^{-7}	3.57×10^{-7}	7.97×10^{-7}	ε	1×10^{-9}
AC Independent Injection No Diverse Onsite AC Power			3.48×10^{-7}	4.00×10^{-7}	7.37×10^{-7}	ε	ε
	LPCS	3.0×10^{-8}	3.75×10^{-7}	4.28×10^{-7}	7.64×10^{-7}	ε	ε
	LPCI	3.0×10^{-8}	3.76×10^{-7}	4.28×10^{-7}	7.64×10^{-7}	ε	ε
	RHR in SPC	6.0×10^{-8}	3.52×10^{-7}	4.03×10^{-7}	7.40×10^{-7}	ε	ε
	LPCI and RHR in SPC	3.3×10^{-8}	3.79×10^{-7}	4.30×10^{-7}	7.66×10^{-7}	ε	ε
Diverse Onsite AC Power			5.2×10^{-8}	2.27×10^{-7}	2.80×10^{-7}	ε	ε
	LPCS	2×10^{-9}	5.5×10^{-8}	2.30×10^{-7}	2.82×10^{-7}	ε	ε
	LPCI	3×10^{-9}	5.6×10^{-8}	2.31×10^{-7}	2.84×10^{-7}	ε	ε
	RHR in SPC	3.6×10^{-8}	8.8×10^{-8}	2.63×10^{-7}	3.15×10^{-7}	ε	ε
	LPCI and RHR in SPC	3.9×10^{-8}	9.2×10^{-8}	2.67×10^{-7}	3.19×10^{-7}	ε	ε

1) LPCS - LPCS not started automatically following LOOP
LPCI - 2 of 4 LPCI pumps not started automatically following LOOP
RHR in SPC - One Loop of RHR started automatically in SPC mode
2) 10% reduction in failure rates and unavailability applied
3) 10% reduction in selected human error rates applied

Table C.5-2 Combined CDF Offsets for BWR 5/6 Plant Configurations

Configuration	ECCS Change[1]	ECCS Offset	EDG Reliability Offset[2]	HRA Offset[3]	Both EDG and HRA Offset	LERF Offset (ECCS Change Only)	LERF Offset (All Changes)
No AC Independent Injection No Diverse Onsite AC Power			4.72×10^{-7}	2.13×10^{-7}	6.79×10^{-7}		1×10^{-9}
	LPCS	1.3×10^{-8}	4.83×10^{-7}	2.24×10^{-7}	6.90×10^{-7}	ε	1×10^{-9}
	LPCI	1.3×10^{-8}	4.83×10^{-7}	2.25×10^{-7}	6.91×10^{-7}	ε	1×10^{-9}
	RHR in SPC	3.7×10^{-8}	5.07×10^{-7}	2.44×10^{-7}	7.11×10^{-7}	ε	1×10^{-9}
	LPCS and RHR in SPC	4.8×10^{-8}	5.18×10^{-7}	2.56×10^{-7}	7.21×10^{-7}	ε	1×10^{-9}
	LPCI and RHR in SPC	4.9×10^{-8}	5.19×10^{-7}	2.56×10^{-7}	7.22×10^{-7}	ε	1×10^{-9}
AC Independent Injection No Diverse Onsite AC Power			3.37×10^{-7}	3.03×10^{-7}	6.41×10^{-7}	ε	ε
	LPCS	3.3×10^{-8}	3.70×10^{-7}	3.36×10^{-7}	6.74×10^{-7}	ε	ε
	LPCI	3.3×10^{-8}	3.71×10^{-7}	3.37×10^{-7}	6.74×10^{-7}	ε	ε
	RHR in SPC	2×10^{-9}	3.39×10^{-7}	3.05×10^{-7}	6.43×10^{-7}	ε	ε
	LPCS and RHR in SPC	3.2×10^{-8}	3.70×10^{-7}	3.36×10^{-7}	6.73×10^{-7}	ε	ε
	LPCI and RHR in SPC	3.3×10^{-8}	3.70×10^{-7}	3.36×10^{-7}	6.74×10^{-7}	ε	ε
Diverse Onsite AC Power			5.3×10^{-8}	1.32×10^{-7}	1.86×10^{-7}	ε	ε
	LPCS	3×10^{-9}	5.7×10^{-8}	1.35×10^{-7}	1.89×10^{-7}	ε	ε
	LPCI	4×10^{-9}	5.8×10^{-8}	1.36×10^{-7}	1.90×10^{-7}	ε	ε
	RHR in SPC	3.5×10^{-8}	8.9×10^{-8}	1.67×10^{-7}	2.21×10^{-7}	ε	ε
	LPCS and RHR in SPC	3.9×10^{-8}	9.2×10^{-8}	1.70×10^{-7}	2.24×10^{-7}	ε	ε
	LPCI and RHR in SPC	3.9×10^{-8}	9.3×10^{-8}	1.71×10^{-7}	2.25×10^{-7}	ε	ε

1) LPCS - 1 LPCS and 1 LPCI not started automatically following LOOP
LPCI - 2 LPCI pumps not started automatically following LOOP
RHR in SPC - One Loop of RHR started automatically in SPC mode
2) 10% reduction in failure rates and unavailability applied
3) 10% reduction in selected human error rates applied

Table C.5-3 Combined CDF Offsets for BWR 3/4 with LPCI Loop-Select Plant Configurations

Configuration[4]	ECCS Change[1]	ECCS Offset	EDG Reliability Offset[2]	HRA Offset[3]	Both EDG and HRA Offset	LERF Offset (ECCS Change Only)	LERF Offset (All Changes)
AC Independent Injection No Diverse Onsite AC Power			$3.41x10^{-7}$	$3.10x10^{-7}$	$6.51x10^{-7}$	ε	ε
	LPCS	$2.0x10^{-8}$	$3.61x10^{-7}$	$3.30x10^{-7}$	$6.71x10^{-7}$	ε	ε
	LPCI	$2.1x10^{-8}$	$3.62x10^{-7}$	$3.31x10^{-7}$	$6.72x10^{-7}$	ε	ε
	RHR in SPC	$3x10^{-9}$	$3.44x10^{-7}$	$3.13x10^{-7}$	$6.54x10^{-7}$	ε	ε
	LPCI and RHR in SPC	$2.1x10^{-8}$	$3.62x10^{-7}$	$3.31x10^{-7}$	$6.72x10^{-7}$	ε	ε
	Loop Select	$(1x10^{-9})$	$3.39x10^{-7}$	$3.07x10^{-7}$	$6.43x10^{-7}$	ε	ε
	LPCS and Loop Select	$1.7x10^{-8}$	$3.58x10^{-7}$	$3.27x10^{-7}$	$6.69x10^{-7}$	ε	ε
	LPCI and Loop Select	$1.8x10^{-8}$	$3.59x10^{-7}$	$3.28x10^{-7}$	$6.69x10^{-7}$	ε	ε
	RHR in SPC and Loop Select	$(2x10^{-8})$	$3.39x10^{-7}$	$3.08x10^{-7}$	$6.49x10^{-7}$	ε	ε
	LPCI and RHR in SPC and Loop Select	$1.6x10^{-8}$	$3.57x10^{-7}$	$3.26x10^{-7}$	$6.67x10^{-7}$	ε	ε
Diverse Onsite AC Power			$4.9x10^{-8}$	$2.22x10^{-7}$	$2.71x10^{-7}$	ε	ε
	LPCS	$2x10^{-9}$	$5.1x10^{-8}$	$2.24x10^{-7}$	$2.72x10^{-7}$	ε	ε
	LPCI	$3x10^{-9}$	$5.2x10^{-8}$	$2.25x10^{-7}$	$2.74x10^{-7}$	ε	ε
	RHR in SPC	$3.5x10^{-8}$	$8.4x10^{-8}$	$2.57x10^{-7}$	$3.06x10^{-7}$	ε	ε
	LPCI and RHR in SPC	$3.8x10^{-8}$	$8.7x10^{-8}$	$2.60x10^{-7}$	$3.09x10^{-7}$	ε	ε
	Loop Select	$(2x10^{-9})$	$4.7x10^{-8}$	$2.19x10^{-7}$	$2.68x10^{-7}$	ε	ε
	LPCS and Loop Select	ε	$4.8x10^{-8}$	$2.21x10^{-7}$	$2.70x10^{-7}$	ε	ε
	LPCI and Loop Select	$1x10^{-9}$	$5.0x10^{-8}$	$2.23x10^{-7}$	$2.71x10^{-7}$	ε	ε
	RHR in SPC and Loop Select	$3.0x10^{-8}$	$7.9x10^{-8}$	$2.52x10^{-7}$	$3.01x10^{-7}$	ε	ε
	LPCI and RHR in SPC and Loop Select	$3.3x10^{-8}$	$8.2x10^{-8}$	$2.55x10^{-7}$	$3.04x10^{-7}$	ε	ε

Table C.5-3 Combined CDF Offsets for BWR 3/4 with LPCI Loop-Select Plant Configurations (continued)

1) LPCS - LPCS not started automatically following LOOP
   LPCI - 2 of 4 LPCI pumps not started automatically following LOOP
   RHR in SPC - One Loop of RHR started automatically in SPC mode
   Loop Select - Eliminate LPCI Loop Select Logic
2) 10% reduction in failure rates and unavailability applied
3) 10% reduction in selected human error rates applied
4) No Loop-Select plants are in the base BWR 3/4 configuration

Table C.6-1 Model Assumptions and Attributes

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
Four high pressure injection systems: Feedwater, HPCI, RCIC, and CRD	Event Sequences	The generic model uses four systems that are available at most BWR plants. The BWR 6 version of the model explicitly addresses plants that have a HPCS instead of a HPCI. Plants that have an isolation condenser rather than RCIC would have similar results to the generic model with the exception of the Small LOCA sequences. This difference has a negligible affect on the changes described in this report.	
Steam-driven Feedwater	Feedwater	Plants that have a motor-driven Feedwater system will have a lower CCDP for transients with offsite power available and isolation of the main steam lines. These sequences have negligible contribution for the changes described in this report.	
BOP support systems are required for Feedwater	Feedwater	This is standard for all BWRs.	
Feedwater requires Condensate and Main Condenser cooling to operate	Feedwater	This is standard for all BWRs.	
HPCI injects outside the shroud	HPCI	This is standard for all BWRs. In the generic model, it injects through the feedwater spargers, but this specific location is not important.	
Except for room cooling, HPCI is AC independent	HPCI	HPCI would not be available in SBO sequences if it is AC dependent.	X
Room cooling is required for HPCI to run for more than 2 hours	HPCI	This requirement makes HPCI less capable for SBO sequences, but as described in the next assumption, there is a backup. Because long-term cooling of the suppression pool is required for long-term HPCI operation, this assumption has a negligible affect on the model results.	

Table C.6-1 Model Assumptions and Attributes (continued)

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
Alternate, AC independent, room cooling is available for HPCI	HPCI	This action is only credited in the SBO model to extend the time frame for restoration of power. Plants that do not have this capability, but require room cooling for HPCI, would have a LOOP CCDP that is approximately 3 times higher than the generic model. This will not affect the conclusions of this report.	
HPCI initial suction is from CST	HPCI	This is standard BWR design. Some plants have a dedicated tank for HPCI and/or RCIC, while others use a dedicated volume in the main CST. This distinction is not important for the generic model.	
SPC is required for long-term HPCI injection	HPCI	It is possible for the operators to revert the suction of HPCI back to the CST in case the suppression pool water is not suitable for continued injection. Other factors, however, would cause limitations for continued HPCI injection that have similar failure modes to SPC (e.g. room cooling). This conservative assumption does not affect the results of the changes described in this report.	
HPCI can be used to depressurize the RPV	Event Sequences	This is standard BWR design, but the specific method of operation will vary from plant to plant. The mode of operation is not important for this report, and the assumption is applicable as long as the plant will depressurize to the point where low pressure systems can inject prior to HPCI losing its motive power.	X
RCIC injects outside the shroud	RCIC	This is standard for all BWRs. In the generic model, it injects through the feedwater spargers, but this specific location is not important.	
RCIC is AC independent	RCIC	This is standard BWR design.	
RCIC does not require room cooling	RCIC	Requiring room cooling would make RCIC less effective during SBO sequences.	X
RCIC initial suction is from CST	RCIC	This is standard BWR design. Some plants have a dedicated tank for HPCI and/or RCIC, while others use a dedicated volume in the main CST. This distinction is not important for the generic model.	

Table C.6-1 Model Assumptions and Attributes (continued)

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
SPC is required for long-term RCIC injection	RCIC	It is possible for the operators to revert the suction of RCIC back to the CST in case the suppression pool water is not suitable for continued injection. Other containment factors, however, would cause limitations for continued RCIC, in particular high RCIC steam line discharge pressure would cause isolation of the system. This conservative assumption does not affect the results of the changes described in this report.	
Operation of RCIC will not depressurize the RPV	Event Sequences	This is a conservative assumption that makes the event trees more manageable. The RPV will eventually depressurize while operating on RCIC, however the competing requirements of containment control make more precise modeling difficult. This assumption has little affect on the model results.	
CRD can be used for long-term cooling in transients	Event Sequences	Many BWR PRAs do not take credit for this system because of its low flow rate. The generic model assumes that as long as another high pressure injection system has operated for more than 12 hours, decay heat will be low enough so that CRD can provide enough flow. Evaluation of the generic model results indicates that CRD does not have a significant affect on the model (RAW = 1.0).	
CRD can be powered by onsite power	CRD	The generic model assumes that CRD can be powered by the EDGs, but they must be started manually. Evaluation of the generic model results indicates that CRD does not have a significant affect on the model (RAW = 1.0).	
Four low pressure injection systems: Condensate, LPCS, LPCI, Service Water	Event Sequences	The generic model uses four systems that are available at most BWR plants. The potential for the most difference is in the service water injection. Each plant has this configured in a different way. The configuration of service water injection will be discussed in other assumptions.	X
Condensate has 3 pumps	Condensate	Review of the model results indicates that the number of Condensate pumps is not important.	

Table C.6-1 Model Assumptions and Attributes (continued)

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
Condensate requires offsite power	Condensate	This is standard BWR design.	
Condensate requires Main Condenser cooling to operate	Condensate	This is standard BWR design.	
Condensate injects through the Feedwater injection lines	Condensate	This is standard BWR design.	
LPCS is a 2 pump system	LPCS	In the generic model, LPCS is modeled as having two independent loops, each with one pump. Some BWRs have a configuration in which each loop has a main pump and a booster pump. The effect of this configuration would be an increase in the failure rate of each loop. Either configuration gives essentially the same results for the changes described in this report. The BWR 6 design has a single LPCS loop. This is explicitly covered in the BWR 6 version of the generic model. In this model, one division of AC power is used for LPCS and one LPCI loop. The other division powers two LPCI loops.	X
LPCS requires SPC for long-term cooling	LPCS	It is assumed that LPCS is not able to pump saturated water from the suppression pool. Failure modes include loss of NPSH and pump seal failure at very high temperatures. If a plant can demonstrate that LPCS can pump suppression pool water at any temperature and pressure expected in a severe accident, then the CST suction option (discussed in other assumptions) is not necessary.	

Table C.6-1 Model Assumptions and Attributes (continued)

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
LPCS can take a suction from the CST for long-term cooling	LPCS	In the generic model, both LPCS and LPCI have the capability of taking its suction from the CST or some other water source that is external to the containment. As long as the plant has one system that has this capability, the generic model results are applicable. Alternatively, if LPCS can pump suppression pool water at any temperature and pressure expected in a severe accident, the CST suction option is not necessary.	X
LPCS requires room cooling	LPCS	In the generic model, LPCS requires room cooling to operate for long-term cooling. In LOOP scenarios, the operators can establish an alternate, AC independent, means of room cooling for LPCS. A review of the generic model results shows that the ability to provide the alternate cooling is not important to the model results.	
LPCS does not require motor or seal cooling	LPCS	Motor cooling is provided by HVAC, so a dedicated system is not necessary. For plants that have such a system, it would be cooled by the same system (RBCCW) as HVAC, so no model differences would be observed. Seal cooling is only necessary to extend the life of the seals and is not required for the mission time in the PRA. Once again, this cooling is provided by the same system as HVAC, so no differences would be observed.	

Table C.6-1 Model Assumptions and Attributes (continued)

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
LPCI electrical configuration	LPCI	The base model is patterned after the "LPCI Mod" configuration, in which each loop of LPCI has one pump powered by each electrical division. The injection valve is powered by DC electrical. This arrangement ensures that no single active failure can disable the LPCI function. Two additional configurations are provided in the generic model. One for the LPCI Loop-Select plants and one for BWR 6 designs. The base and additional configurations cover all BWR designs. In the LPCI Loop-Select configuration, both pumps in each loop are powered by the same electrical division. The inject valves are powered by DC in the generic model, but this nearly equivalent to a swing bus configuration. In the BWR 6 configuration, two LPCI pumps are powered by one electrical division. The other is powered by the same electrical division as the LPCS pump.	
LPCI cross-tie valve	LPCI	The base model assumes that the LPCI cross-tie valve is closed for short-term injection. The LPCI Loop-Select configuration model assumes that the LPCI cross-tie valve is open in all scenarios. The BWR 6 configuration does not contain a cross-tie valve. The combination of these configurations covers all BWR designs.	

Table C.6-1 Model Assumptions and Attributes (continued)

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
LPCI does not require room cooling	LPCI	The generic model makes the assumption that RHR room cooling can be provided by natural circulation of air within the reactor building. A sensitivity analysis was performed to determine the effect of requiring room cooling. It was determined that the conclusions drawn in this report are not sensitive to this assumption.	
LPCI does not require motor or seal cooling	LPCI	Motor cooling is provided by the environment, so a dedicated system is not necessary. For plants that have such a system, it would be cooled by RBCCW, which is in turn cooled by service water. Because service water is a large part of long-term heat removal, it is judged that the impact of this requirement would be very small. Seal cooling is only necessary to extend the life of the seals and is not required for the mission time in the PRA. Once again, this cooling is provided by RBCCW, so the impact (if any) would be very small.	
LPCI requires SPC for long-term cooling	LPCI	It is assumed that LPCI is not able to pump saturated water from the suppression pool. Failure modes include loss of NPSH and pump seal failure at very high temperatures. If a plant can demonstrate that LPCI can pump suppression pool water at any temperature and pressure expected in a severe accident, then the CST suction option (discussed in other assumptions) is not necessary.	

Table C.6-1 Model Assumptions and Attributes (continued)

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
LPCI can take a suction from the CST for long-term cooling	LPCI	In the generic model, both LPCS and LPCI have the capability of taking its suction from the CST or some other water source that is external to the containment. As long as the plant has one system that has this capability, the generic model results are applicable. Alternatively, if LPCI can pump suppression pool water at any temperature and pressure expected in a severe accident, the CST suction option is not necessary.	X
1 LPCI pump is sufficient for both injection and SPC	LPCI	It is assumed that if the only low pressure pump available is one RHR pump, it can be switched alternately between LPCI and SPC modes, or by passing LPCI flow through the RHR heat exchangers. The swap is not explicitly included in the model, but the failure modes associated with the swap are included in the initiation logic for each of the modes. The BWR 6 configuration accounts for the one loop of LPCI that does not have an in-line heat exchanger. If that loop were the only one available, it could not be used for both functions.	X
Service Water can be used for early injection in some sequences	Service Water	The only event tree in the generic model that excludes service water as an early injection system is LBLOCA. It is assumed that there is sufficient time in all other sequences to align this system. A sensitivity analysis was performed to determine the importance of this assumption. The results indicate that the conclusions drawn in this report are not influenced by this assumption.	
Pumps used for service water injection can be powered by onsite power	Service Water	The generic model assumes that the pumps used for service water injection can be powered by onsite power, thus the function is available in LOOP sequences. A sensitivity analysis was performed to determine the importance of this assumption.	X
The reactor can be depressurized using: PCS, SRVs, or HPCI	Event Sequences	The PCS and SRV methods of depressurization are standard BWR design. The HPCI method is discussed in other assumptions.	

Table C.6-1 Model Assumptions and Attributes (continued)

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
PCS requires Condensate	PCS	For convenience, the PCS system is modeled such that the steam flow path from the reactor to the condenser and the liquid flow path from the condenser back to the reactor are included in the same fault tree. The configuration assumption utilized here is that these functions are coupled. For example, steaming to the condenser via the main steam lines and injection back to the vessel using CRD would not be considered successful. This assumption is typical of BWR PRAs. It is judged that a more complex model would not significantly alter the results of this report. There are several dependencies between the main steam path and the condensate pumps that would limit the effectiveness of more complex modeling.	
PCS automatically controls pressure	PCS	This is standard BWR design.	
MSIV isolation signals	Event Sequences	MSIVs isolate on several signals: low water level (LPCI initiation), low steam line pressure, high steam tunnel temperature, high main steam line flow, and low condenser vacuum. They are also assumed to isolate on a loss of offsite power (although this is redundant to low condenser vacuum for PRA purposes). This makes PCS unavailable for MBLOCA (low level), LBLOCA (low level), LOOP (several signals), and Loss of Service Water (high steam tunnel temperature) sequences. It can be available for all other event trees. This is standard BWR design. Plants should confirm that PCS can be used in all other event trees.	X
SRVs discharge steam to the suppression pool	Event Sequences	This is standard BWR design.	
SRVs are DC powered	Depressurization	This is standard BWR design.	

Table C.6-1 Model Assumptions and Attributes (continued)

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
ADS logic is assumed to be disabled in non-ATWS sequences	Depressurization	The generic model assumes that the operators, per EPG implementing procedures, will inhibit ADS logic. Past studies by the BWROG have found that the CDF effect of this procedure is plant specific. A sensitivity analysis was performed to investigate this affect on the changes recommended in this report. The results show that this assumption does not impact the conclusions of this report.	
Containment control can be provided by 2 systems: RHR and containment vent	Event Sequences	The RHR system is the standard design for containment heat removal in BWRs. It is assumed that all BWRs have a means of performing containment venting. Other assumptions will deal with the specific attributes of the containment vent.	X
RHR system has two loops	RHR	This is standard for all BWR designs.	
Each RHR loop has two pumps	RHR	This is the standard design for BWR models up through BWR 4. The BWR 6 model has only one RHR pump per loop of RHR.	
Each RHR loop has one heat exchanger	RHR	This is standard for all BWR designs.	
RHR must be manually initiated in SPC mode	RHR	This is standard for all BWR designs.	
RHR in SPC mode can be powered by onsite power	RHR	This is standard for all BWR designs.	
Containment spray valves can be powered by onsite power	RHR	This configuration is typical of all BWR designs. The changes in this report do not rely on the power source of these valves because of the nearly complete dependence of CSS failure modes with the failure modes of the SPC mode of RHR.	
SDC suction valves can be powered by onsite power	RHR	This configuration is typical of all BWR designs. The changes in this report do not rely on the power source of these valves because of the nearly complete dependence of SDC failure modes with the failure modes of the SPC mode of RHR.	

Table C.6-1 Model Assumptions and Attributes (continued)

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
Containment vent requires both AC and DC power	Vent	This is a conservative assumption because plants have many different configurations for the containment vent system(s). The important feature is that the plant has the ability and procedures available to perform containment venting.	X
RPS Reliability	RPS	In the generic model, ATWS sequences do not form a significant portion of the CDF. This is because of the high reliability of the RPS system. The generic model RPS is based on the system configuration documented in INEEL/EXT-98-00670, General Electric RPS Unavailability, 1984-1995. Manual SCRAM was included in the electrical portion of RPS reliability in the INEEL study. In the generic model, the manual SCRAM was modeled separately, and the reliability was adjusted appropriately.	X
Electrical RPS is backed up by ARI and manual SCRAM	RPS	This is standard BWR design.	
Mechanical RPS does not have a backup	RPS	This assumption is consistent with the way that the mechanical RPS reliability was defined.	
SLC injects boron through a single penetration	SLC	The manner in which SLC injects does not affect the results of the generic model.	
One SLC pump is sufficient if main steam is available	SLC	Success of SLC is highly dependent on the amount of energy that is deposited into the containment. It is assumed that if a substantial amount of heat is removed to the main condenser, then a lower injection rate of SLC is possible. The specifics of the SLC system do not affect the conclusions of this report as long as the reliability of RPS is consistent with the INEEL study.	

Table C.6-1 Model Assumptions and Attributes (continued)

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
SLC can be powered by onsite AC power	SLC	This is standard BWR design.	
SRVs do not require power or pneumatics to operate in safety mode	Main Steam	This is standard BWR design.	
The generic model has a single service water system	Service Water	There are many different service water arrangements at BWRs. The range from a single common system to several systems dedicated for different purposes. The generic model uses a configuration that provides a high degree of intra-system dependence, and a high degree of dependence for the supported systems. In the PRA, this amounts to a limiting configuration. The conclusions of this report are not sensitive to changes in the reliability of the service water system, so long as the system does not fail altogether. A sensitivity analysis was performed to confirm this. The specific service water configuration at a plant that is different from the generic model would not affect the conclusions of this report.	
The generic model incorporates service water booster pumps	Service Water	The generic model relies on service water booster pumps for RHR heat exchanger cooling and for service water injection. A review of the model results indicates that the conclusions of this report will not be altered for plants that do not have (or need) booster pumps for these functions.	

Table C.6-1 Model Assumptions and Attributes (continued)

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
The generic model uses 2 diesel generators	AC Power	There are a varying number of diesel generators per unit at BWRs. The survey of plant specific information requested this information, and it was analyzed in the PRA benchmark (Section C.3.5). The results of the model weakly influenced by the number of diesel generators dedicated to each unit. Because two per unit is a limiting configuration, and because more diesel generators per unit would have a multiplying effect on the CDF decrease due to DG reliability, it is judged that the conclusions of this report apply to plants with any number of dedicated diesel generators per unit.	
EDGs require service water	AC Power	This is standard design for large diesel generators.	
EDGs require room ventilation only	AC Power	The diesel configuration in the generic model assumes that room cooling is not required, but ventilation is. Because the EDGs already require service water for engine cooling, the assumption about room cooling is irrelevant.	
Offsite power configuration	AC Power	The offsite power connection in the generic model (described in Section C.2.1.7) is fairly complex. The specific configuration would vary from plant to plant, but all plants incorporate similar failure modes. Section C.3.5 discusses this aspect of the generic model as it applies to other plant configurations.	
There are two DC sub-systems	DC Power	This is standard configuration.	
125 V DC and 250 V DC are independent	DC Power	The generic model assumes that the 125 V and 250 V DC systems are independent. Some plants use a single battery set to provide both functions. A sensitivity analysis was performed to address this feature.	
Each DC system has a swing charger	DC Power	In the generic model, each DC division has a dedicated charger. There is also a 125 V and 250 V swing charger that may be used to power either division, but not both.	X

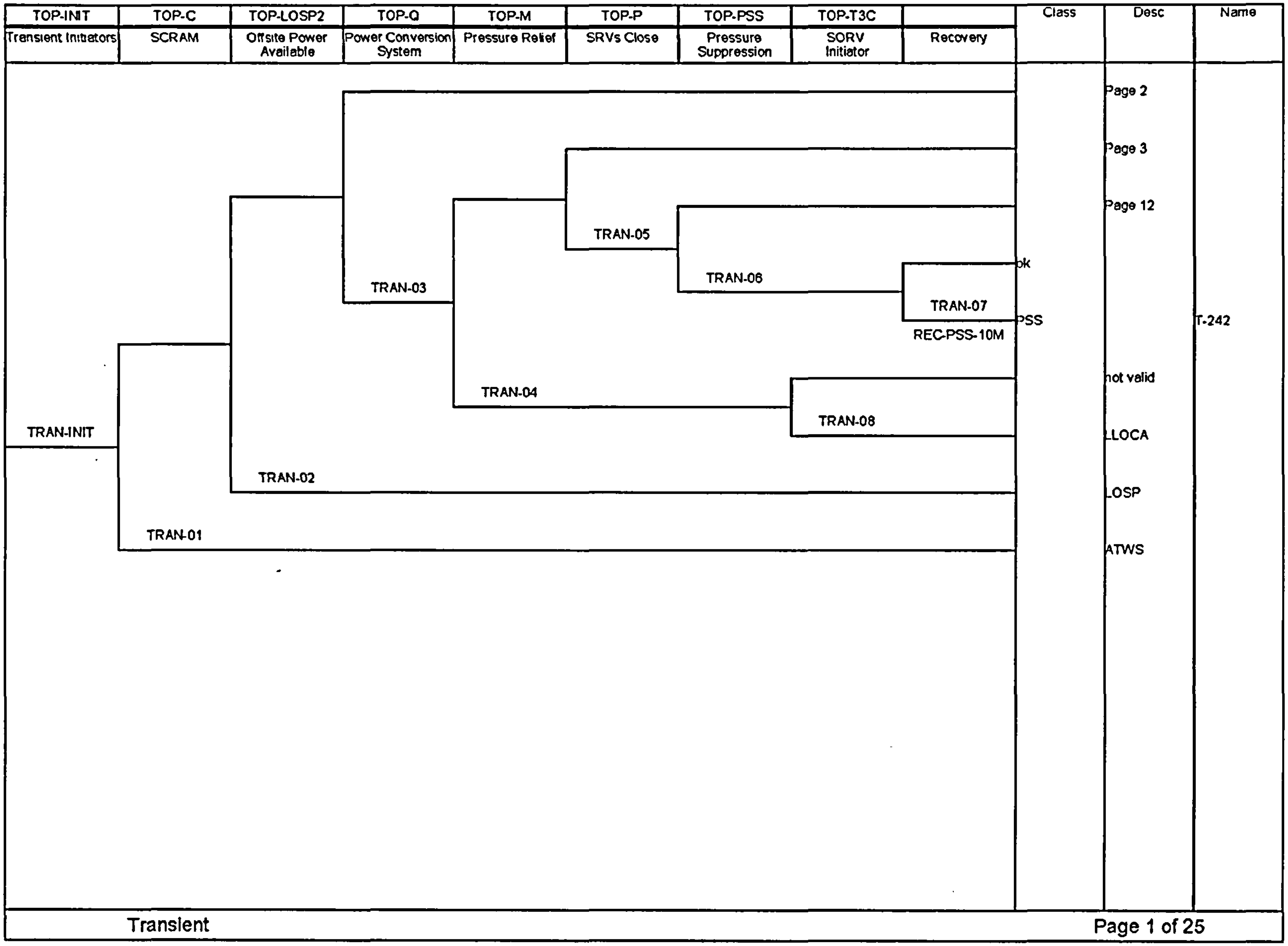
Table C.6-1 Model Assumptions and Attributes (continued)

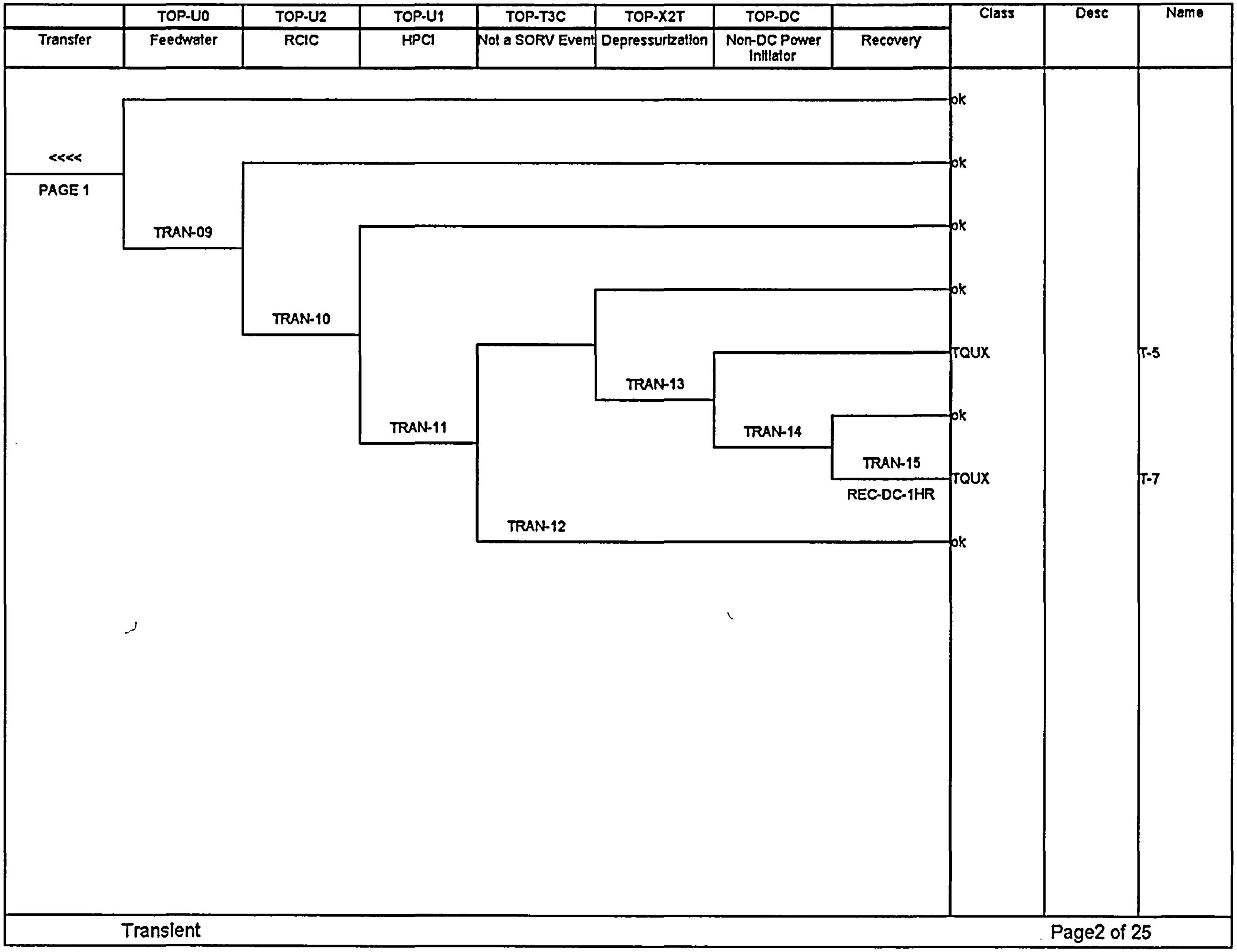
Assumption or Attribute	**Context**	**Assessment**	**Requires Plant-Specific Validation**
Batteries are rated for 4 hours	DC Power	The generic model uses a 4 hour rating for 125 V batteries. This translates to a 6 hour offsite power recovery factor for SBO sequences that rely on DC powered injection systems. Section C.3 discusses the impact that different battery ratings have on the analysis.	
Success Criteria	Event Sequences	Plants need to ensure that the success criteria listed in table C.5-1 are applicable to their plant. If a particular system or subsystem is already described in this table of attributes and assumptions and this table indicates that the attribute or assumption does not affect the results, then the associated success criterion need not apply to that plant.	X
Initiating Event Frequencies	Initiating Events	The plant needs to ensure that its pipe break frequency is consistent with the available industry analyses. It is expected that plants can use their existing ISI programs to make this assessment. It is also important that plants have taken the actions necessary to ensure that their local grid is stable and that it is unlikely that a start of the ECCS system (either real or spurious) will disturb the grid to the point of losing offsite power.	X
Changes are Not Detrimental for Fire Initiated Events	External Events	Fire events that directly cause a loss of offsite power may have synergistic effects with the changes in this report that involve the reconfiguration of the LPCI or LPCS systems. If a plant has an area where a fire will cause a loss of offsite power and will disable a portion of either the LPCI or LPCS systems (other than the main control room), a plant specific fire risk analysis must be performed if the configuration of the LPCI or LPCS systems is being changed.	X

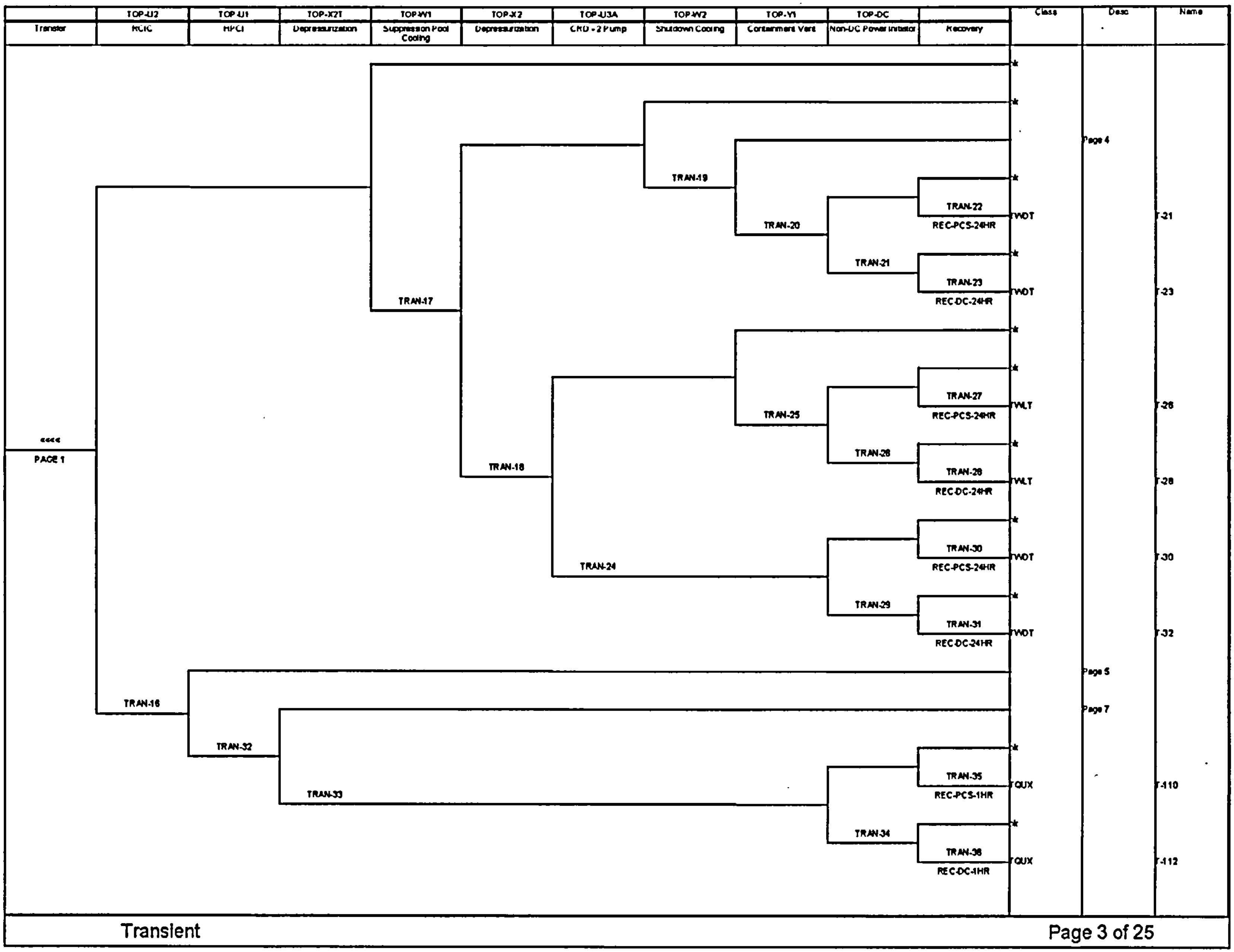
Table C.6-1 Model Assumptions and Attributes (continued)

Assumption or Attribute	Context	Assessment	Requires Plant-Specific Validation
Changes are Not Detrimental for Flood Initiated Events	External Events	Flood events that directly cause a loss of offsite power may have synergistic effects with the changes in this report that involve the reconfiguration of the LPCI or LPCS systems. If a plant has an area where a flood will cause a loss of offsite power and will disable a portion of either the LPCI or LPCS systems, a plant specific flood risk analysis must be performed if the configuration of the LPCI or LPCS systems is being changed.	X
Changes are Not Detrimental for Seismic Events	External Events	Seismic events of extremely large magnitude can cause a rupture in the primary system. If the diesel generators are not as seismically rugged as the plant recirculation loops, a plant specific seismic analysis needs to be performed to confirm the applicability of the Option 3 changes described in this report.	X

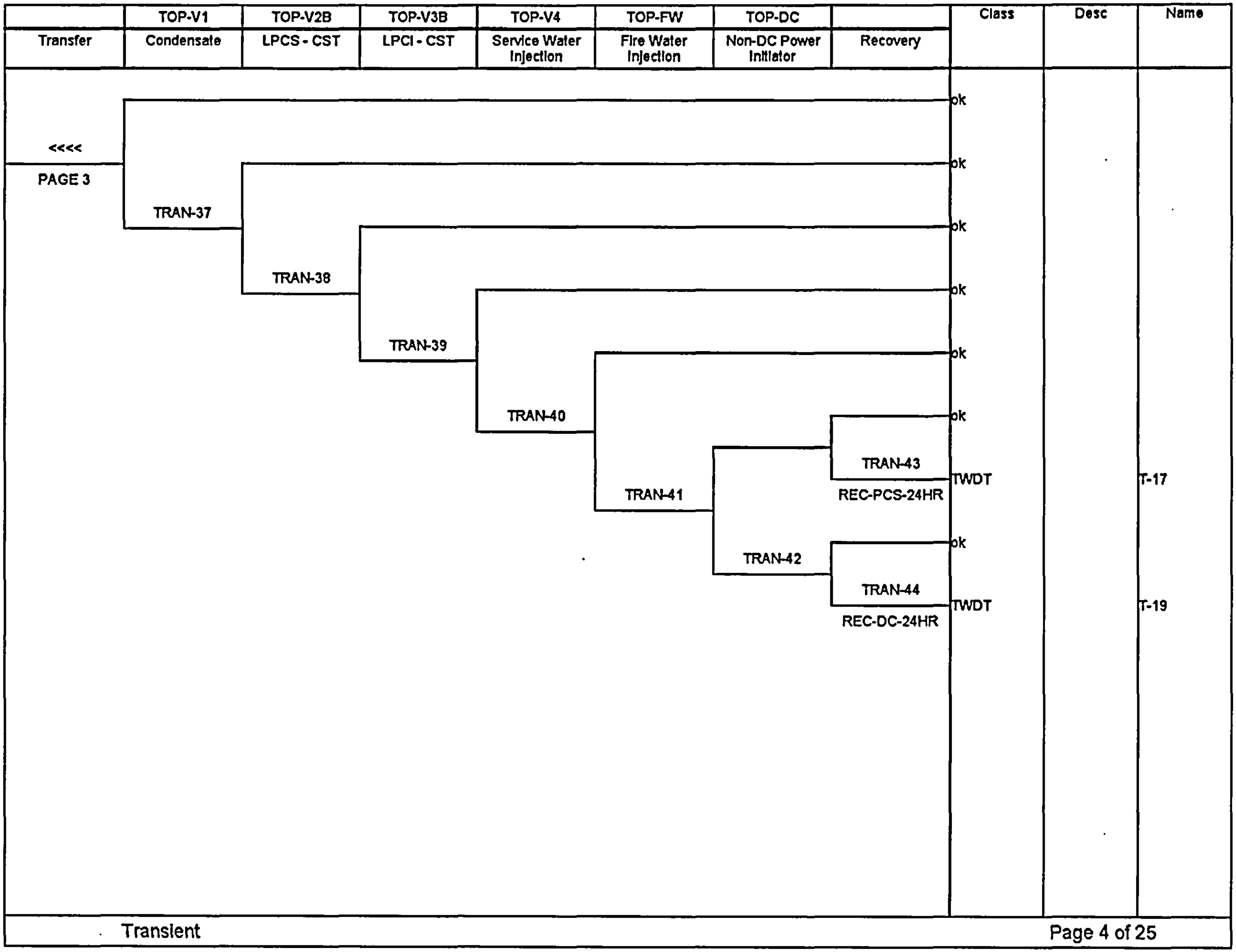
Figure C.3-1 Transient Event Tree



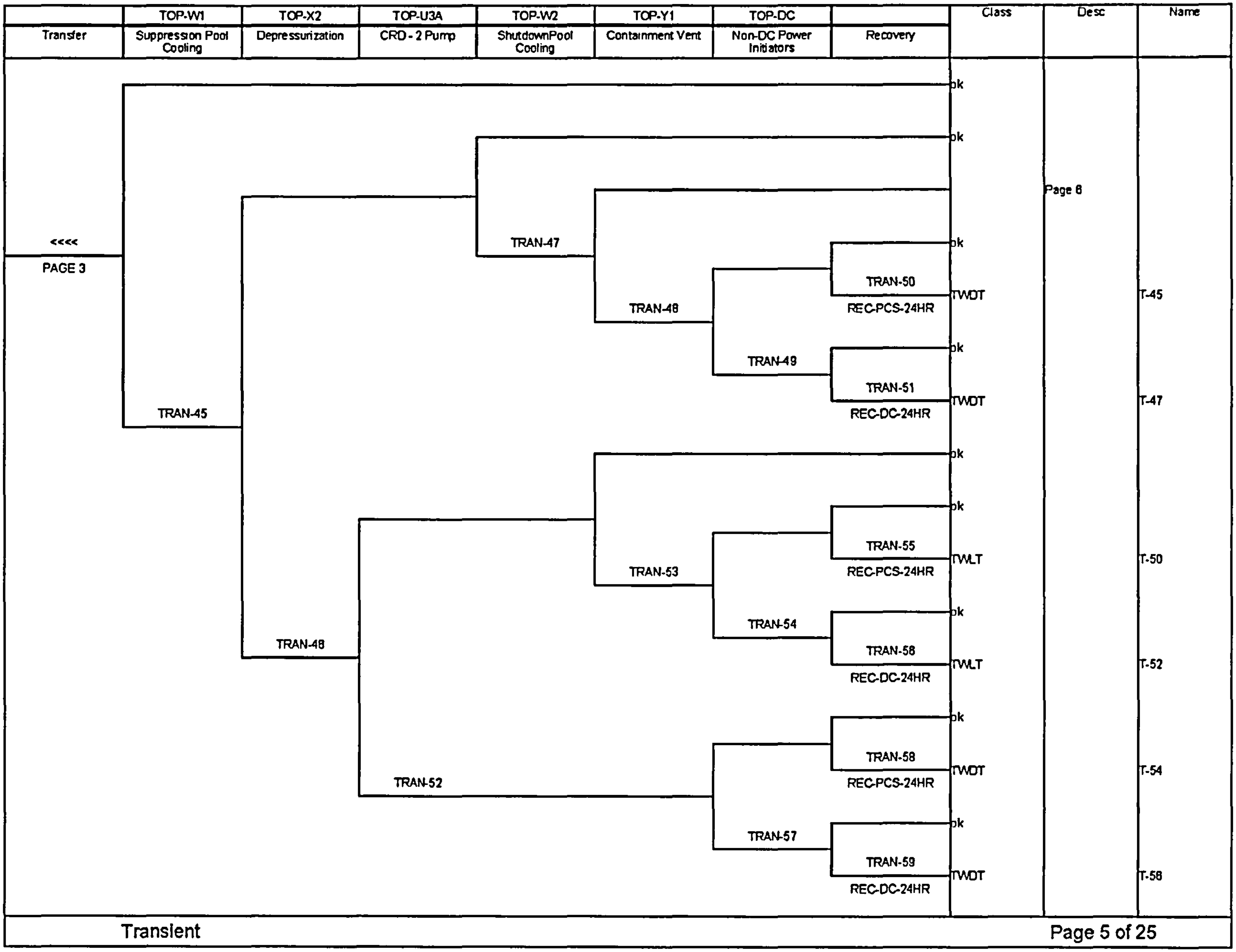

Transfer
TOP-U0
Feedwater
TOP-U2
RCIC
TOP-U1
HPCI
TOP-T3C
Not a SORV Event
TOP-X2T
Depressurization
TOP-DC
Non-DC Power Initiator
Recovery
Class
Desc
Name
<<<<
PAGE 1
TRAN-09
TRAN-10
TRAN-11
TRAN-12
TRAN-13
TRAN-14
TRAN-15
REC-DC-1HR
ok
ok
ok
ok
TQUX
T-5
ok
TQUX
T-7
ok
Transient
Page2 of 25

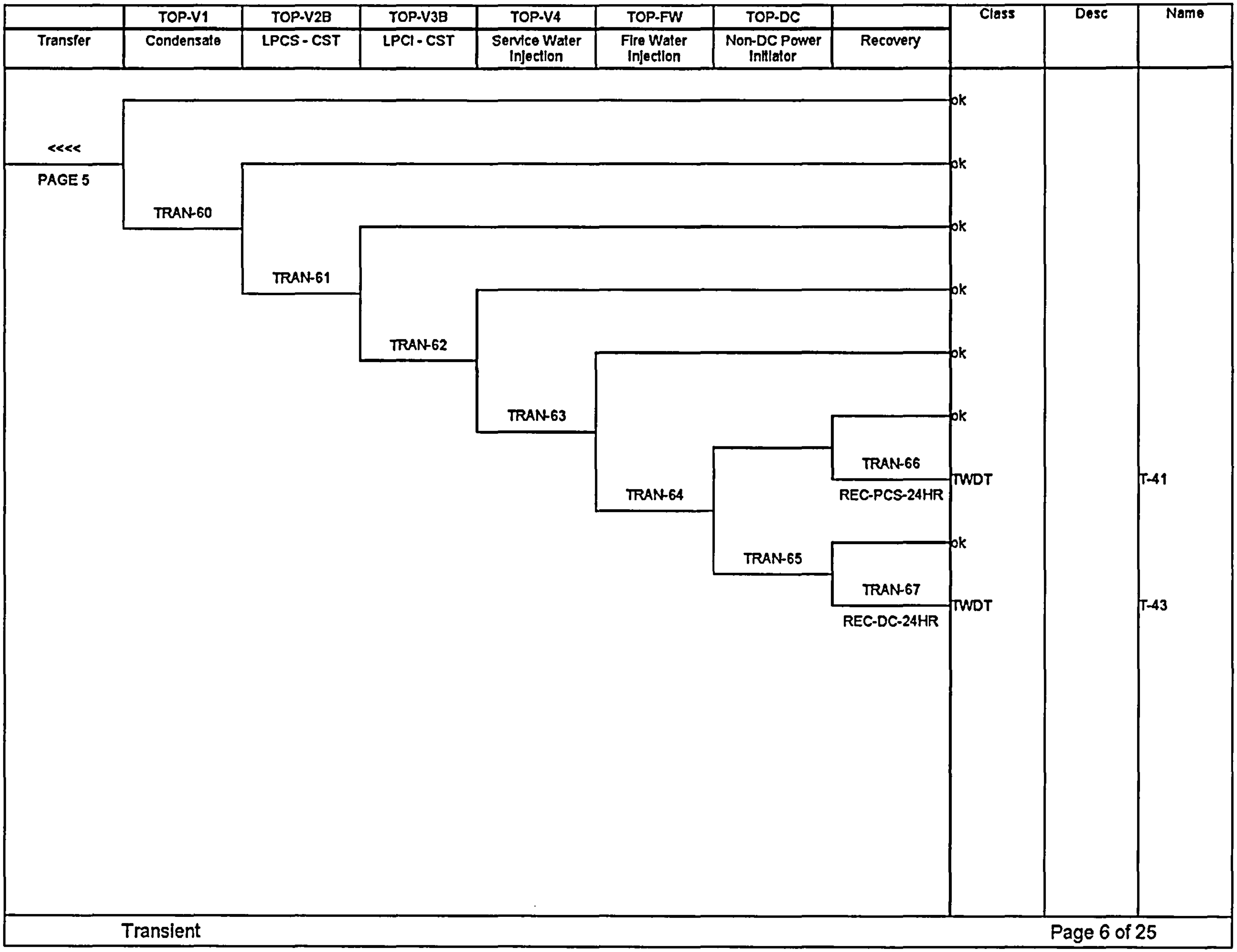


TOP-U2
TOP-U1
TOP-X2T
TOP-W1
TOP-X2
TOP-U3A
TOP-W2
TOP-Y1
TOP-DC
Class
Desc
Name
Transfer
RCIC
HPCI
Depressurization
Suppression Pool Cooling
Depressurization
CRD - 2 Pump
Shutdown Cooling
Containment Vent
Non-DC Power Initiator
Recovery
PAGE 1
TRAN-16
TRAN-17
TRAN-18
TRAN-19
TRAN-20
TRAN-21
TRAN-22
REC-PCS-24HR
TRAN-23
REC-DC-24HR
TRAN-24
TRAN-25
TRAN-26
TRAN-27
TRAN-28
TRAN-29
TRAN-30
TRAN-31
TRAN-32
TRAN-33
TRAN-34
TRAN-35
REC-PCS-1HR
TRAN-36
REC-DC-1HR
TWDT
TWLT
TQUX
Page 4
Page 5
Page 7
T-21
T-23
T-26
T-28
T-30
T-32
T-110
T-112
Transient
Page 3 of 25



TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-DC
Class
Desc
Name
Transfer
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Non-DC Power Initiator
Recovery
<<<<
PAGE 3
TRAN-37
TRAN-38
TRAN-39
TRAN-40
TRAN-41
TRAN-42
TRAN-43
REC-PCS-24HR
TRAN-44
REC-DC-24HR
ok
ok
ok
ok
ok
ok
TWDT
T-17
ok
TWDT
T-19
Transient
Page 4 of 25


TOP-W1
TOP-X2
TOP-U3A
TOP-W2
TOP-Y1
TOP-DC
Class
Desc
Name
Transfer
Suppression Pool Cooling
Depressurization
CRD - 2 Pump
ShutdownPool Cooling
Containment Vent
Non-DC Power Initiators
Recovery
<<<<
PAGE 3
TRAN-45
TRAN-46
TRAN-47
TRAN-48
TRAN-49
TRAN-50
REC-PCS-24HR
TRAN-51
REC-DC-24HR
TRAN-52
TRAN-53
TRAN-54
TRAN-55
REC-PCS-24HR
TRAN-56
REC-DC-24HR
TRAN-57
TRAN-58
REC-PCS-24HR
TRAN-59
REC-DC-24HR
ok
ok
Page 6
ok
TWDT
T-45
ok
TWDT
T-47
ok
ok
TWLT
T-50
ok
TWLT
T-52
ok
TWDT
T-54
ok
TWDT
T-56
Transient
Page 5 of 25


TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-DC
Class
Desc
Name
Transfer
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Non-DC Power Initiator
Recovery
<<<<
PAGE 5
TRAN-60
TRAN-61
TRAN-62
TRAN-63
TRAN-64
TRAN-65
TRAN-66
TRAN-67
REC-PCS-24HR
REC-DC-24HR
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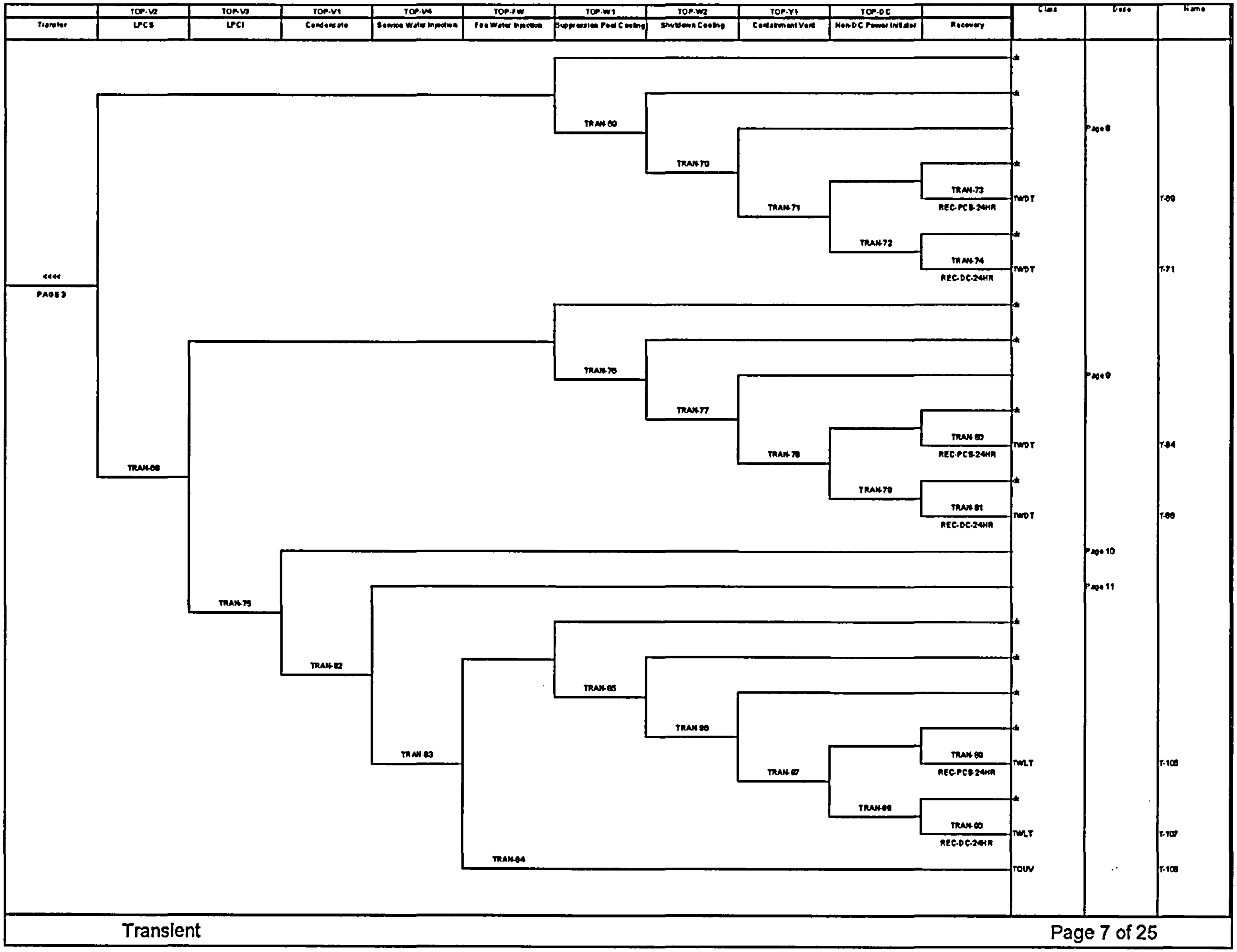
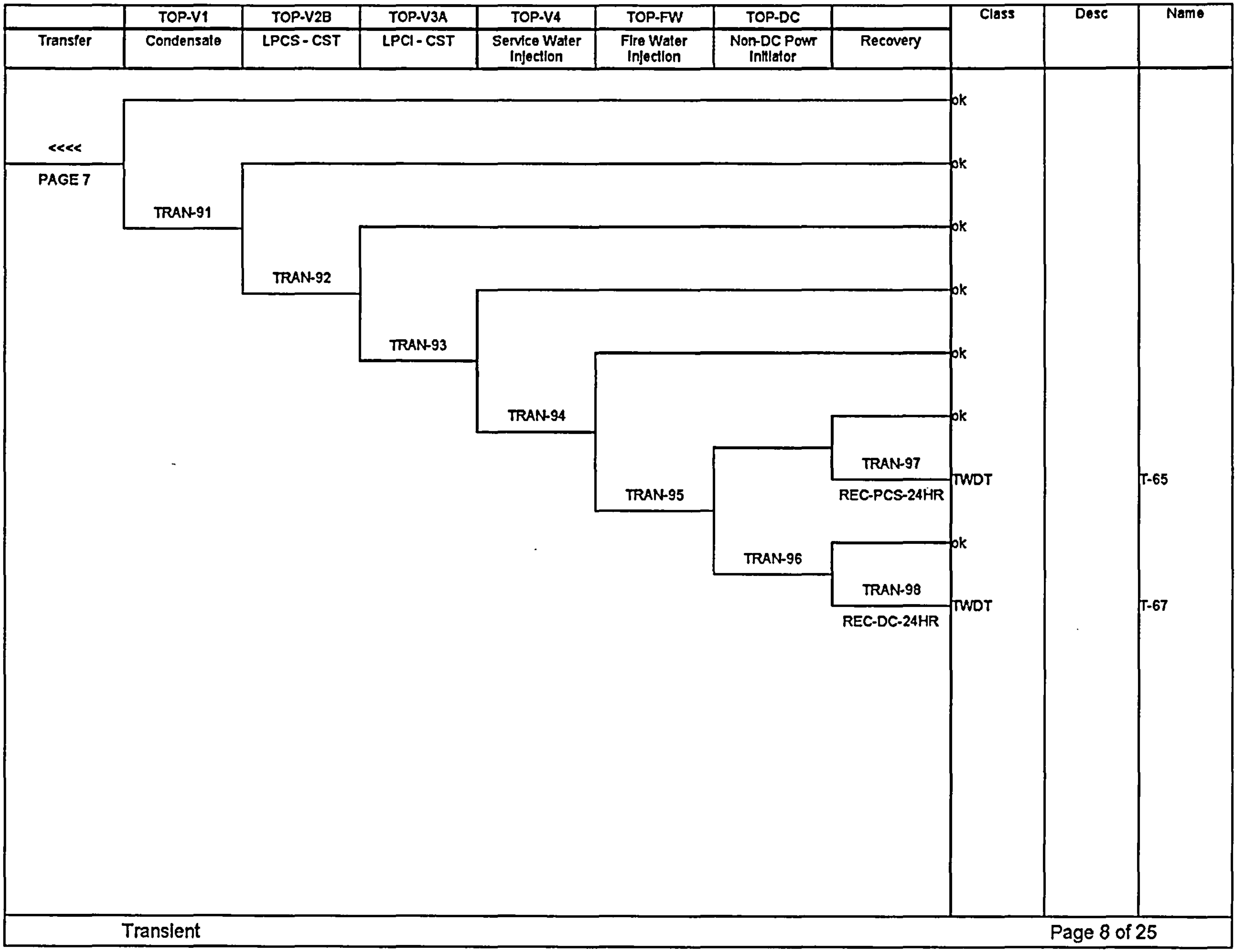
ok
ok
ok
ok
ok
TWDT
T-41
ok
TWDT
T-43
Transient
Page 6 of 25



TOP-V2
TOP-V3
TOP-V1
TOP-V4
TOP-FW
TOP-W1
TOP-W2
TOP-Y1
TOP-DC
Transfer
LPCS
LPCI
Condensate
Service Water Injection
Fire Water Injection
Suppression Pool Cooling
Shutdown Cooling
Containment Vent
Non-DC Power Initiator
Recovery
Class
Dose
Name
PAGE 3
TRAN-68
TRAN-69
TRAN-70
TRAN-71
TRAN-72
TRAN-73
TRAN-74
REC-PCS-24HR
REC-DC-24HR
TRAN-75
TRAN-76
TRAN-77
TRAN-78
TRAN-79
TRAN-80
TRAN-81
TRAN-82
TRAN-83
TRAN-84
TRAN-85
TRAN-86
TRAN-87
TRAN-88
TRAN-89
TRAN-90
Page 8
Page 9
Page 10
Page 11
TWDT
TWLT
TOUV
T-69
T-71
T-84
T-86
T-105
T-107
T-108
Transient
Page 7 of 25


TOP-V1
TOP-V2B
TOP-V3A
TOP-V4
TOP-FW
TOP-DC
Class
Desc
Name
Transfer
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Non-DC Powr Initiator
Recovery
<<<<
PAGE 7
TRAN-91
TRAN-92
TRAN-93
TRAN-94
TRAN-95
TRAN-96
TRAN-97
REC-PCS-24HR
TRAN-98
REC-DC-24HR
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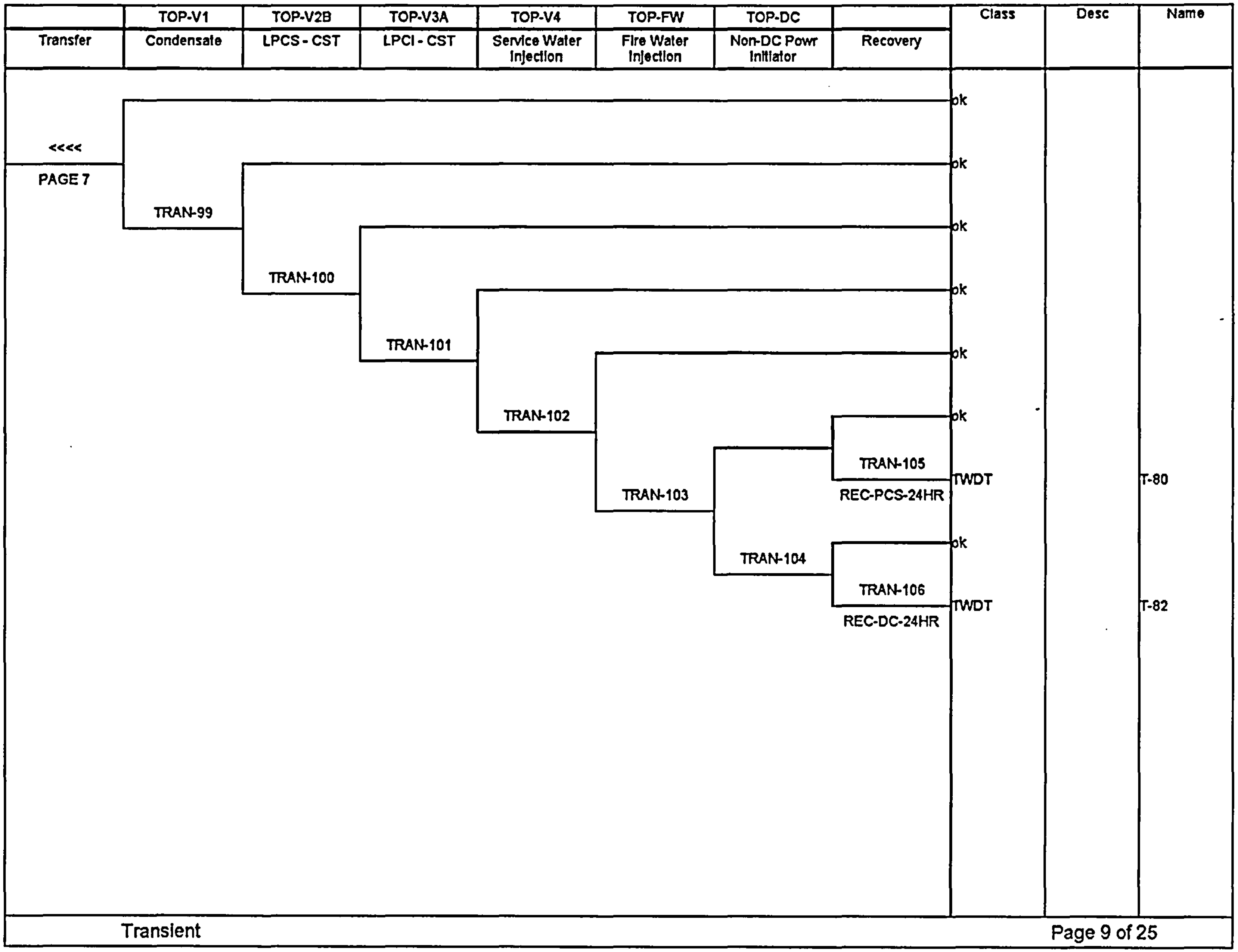
ok
ok
ok
ok
ok
TWDT
T-65
ok
TWDT
T-67
Transient
Page 8 of 25


Transfer
TOP-V1
Condensate
TOP-V2B
LPCS - CST
TOP-V3A
LPCI - CST
TOP-V4
Service Water Injection
TOP-FW
Fire Water Injection
TOP-DC
Non-DC Powr Initiator
Recovery
Class
Desc
Name
<<<<
PAGE 7
TRAN-99
TRAN-100
TRAN-101
TRAN-102
TRAN-103
TRAN-104
TRAN-105
REC-PCS-24HR
TRAN-106
REC-DC-24HR
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ok
ok
ok
ok
ok
TWDT
T-80
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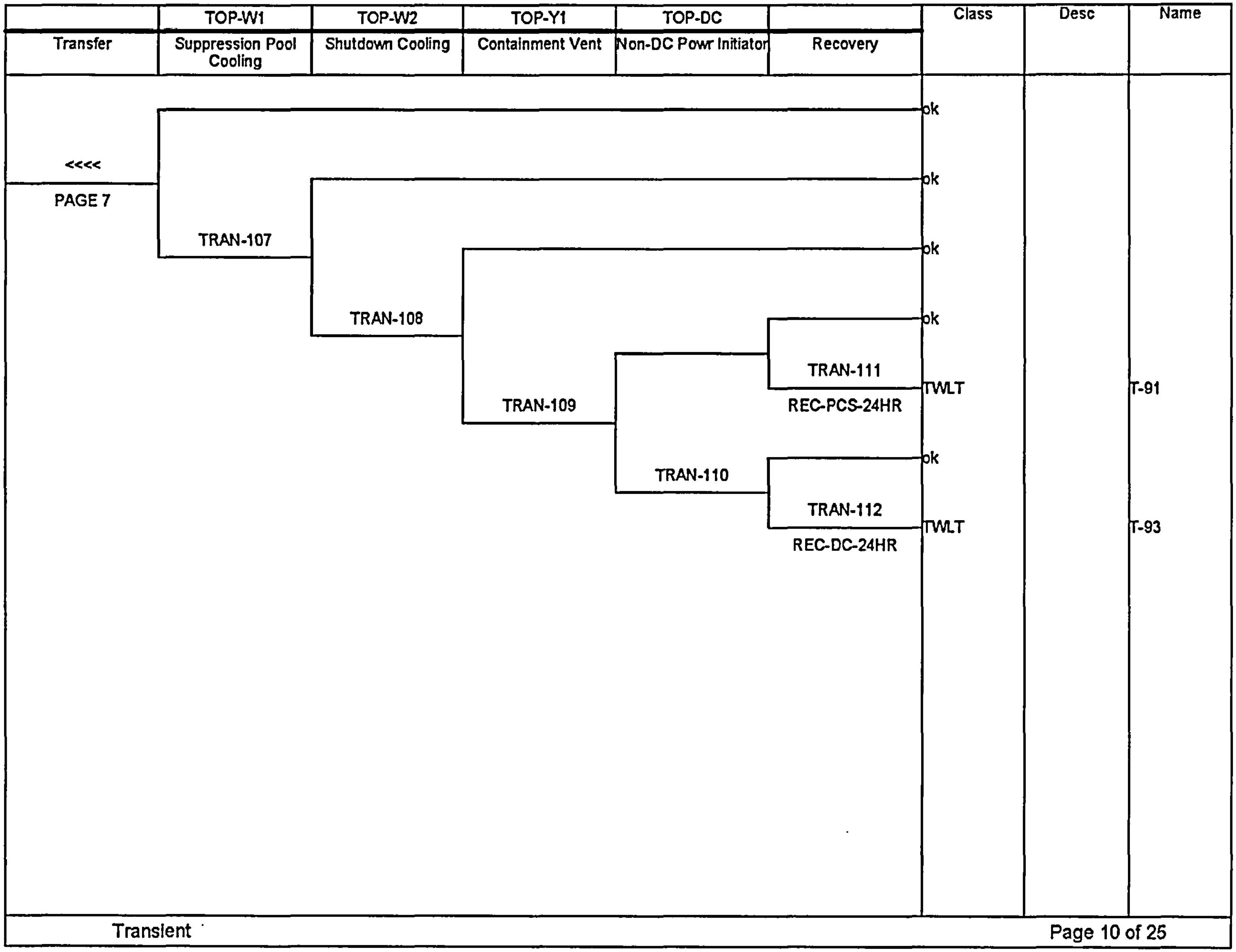
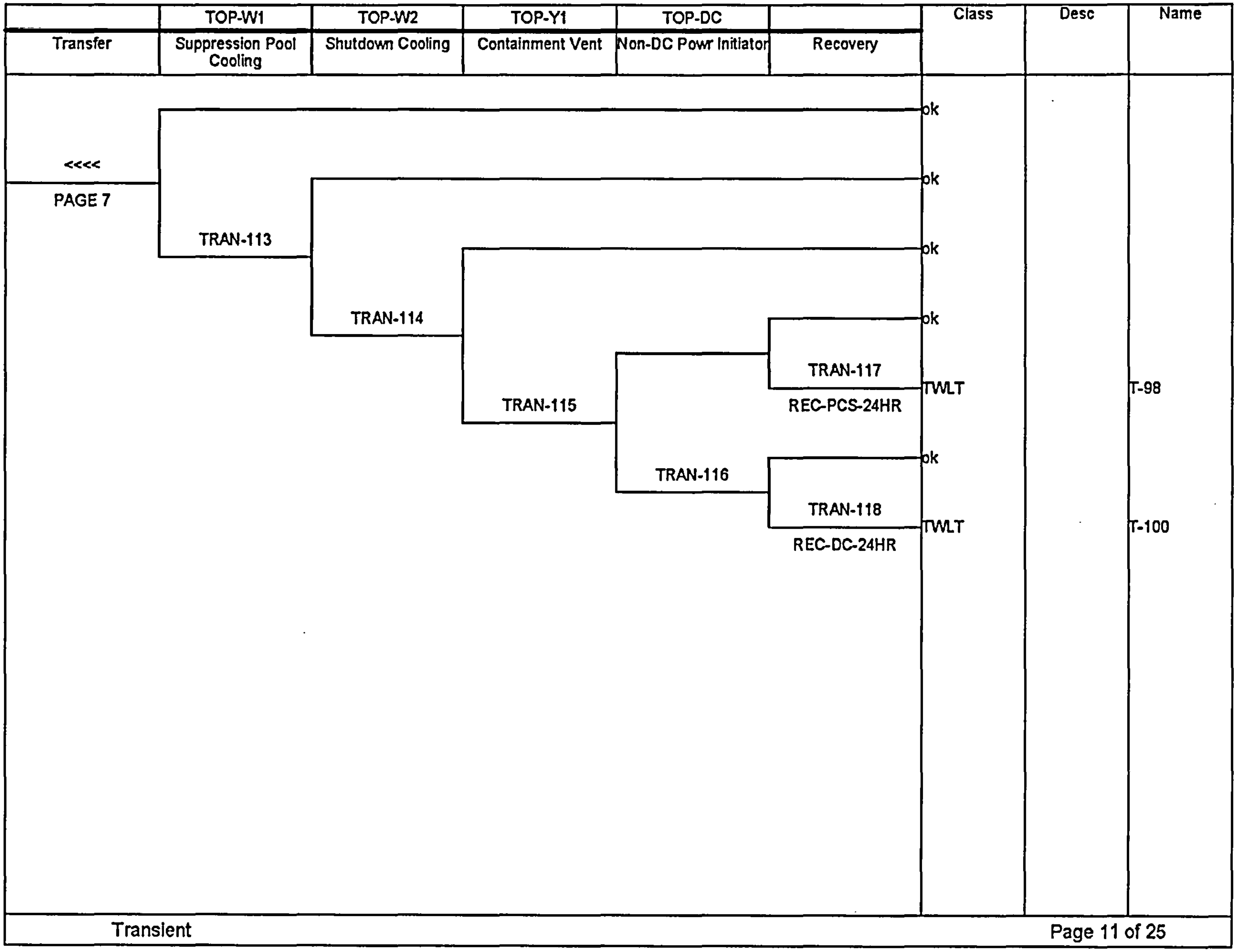
TWDT
T-82
Transient
Page 9 of 25


TOP-W1
TOP-W2
TOP-Y1
TOP-DC
Class
Desc
Name
Transfer
Suppression Pool Cooling
Shutdown Cooling
Containment Vent
Non-DC Powr Initiator
Recovery
<<<<
PAGE 7
TRAN-107
TRAN-108
TRAN-109
TRAN-110
TRAN-111
REC-PCS-24HR
TRAN-112
REC-DC-24HR
ok
ok
ok
ok
TWLT
T-91
ok
TWLT
T-93
Transient
Page 10 of 25


TOP-W1
TOP-W2
TOP-Y1
TOP-DC
Class
Desc
Name
Transfer
Suppression Pool Cooling
Shutdown Cooling
Containment Vent
Non-DC Powr Initiator
Recovery
<<<<
PAGE 7
TRAN-113
TRAN-114
TRAN-115
TRAN-116
TRAN-117
REC-PCS-24HR
TRAN-118
REC-DC-24HR
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ok
ok
ok
TWLT
T-98
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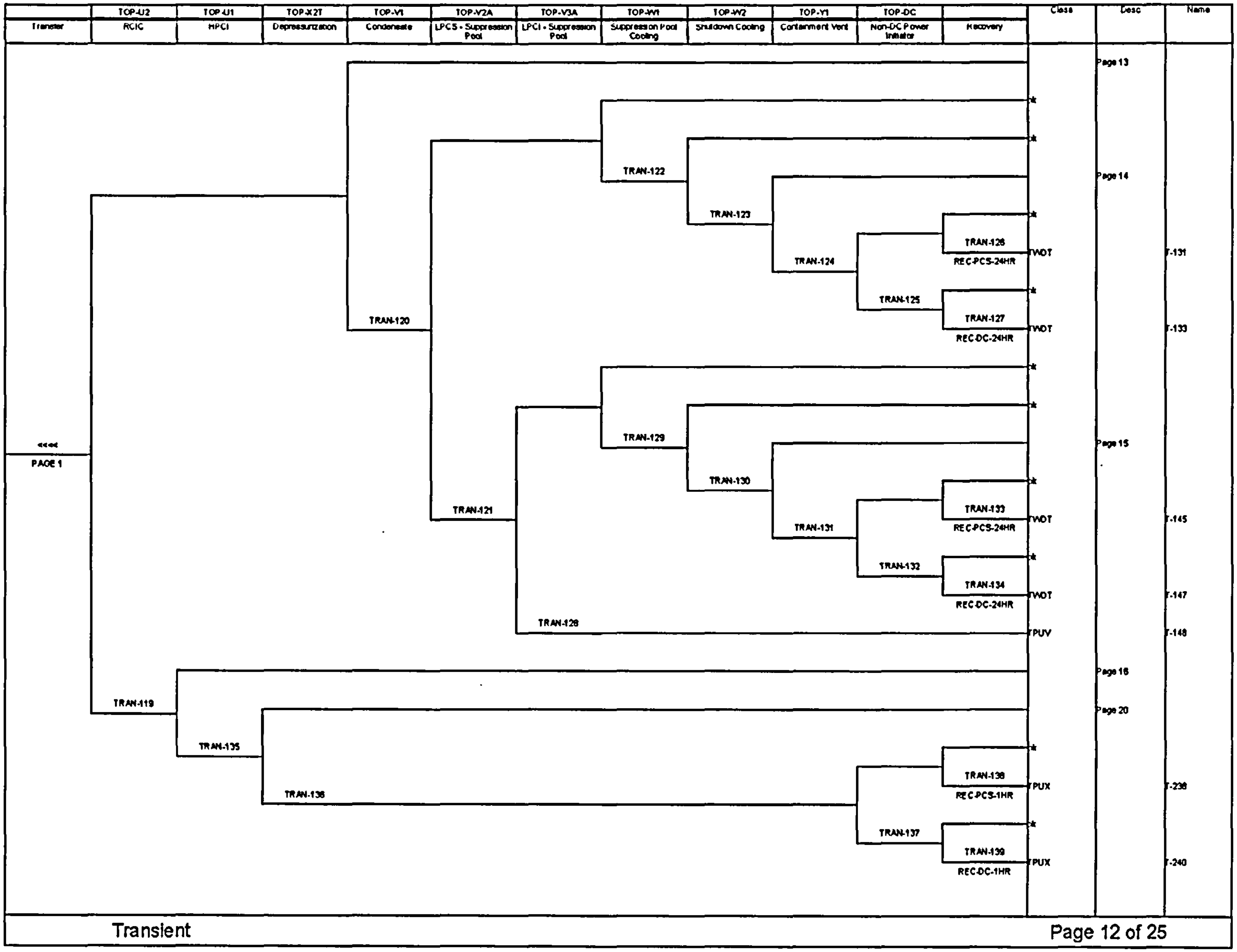
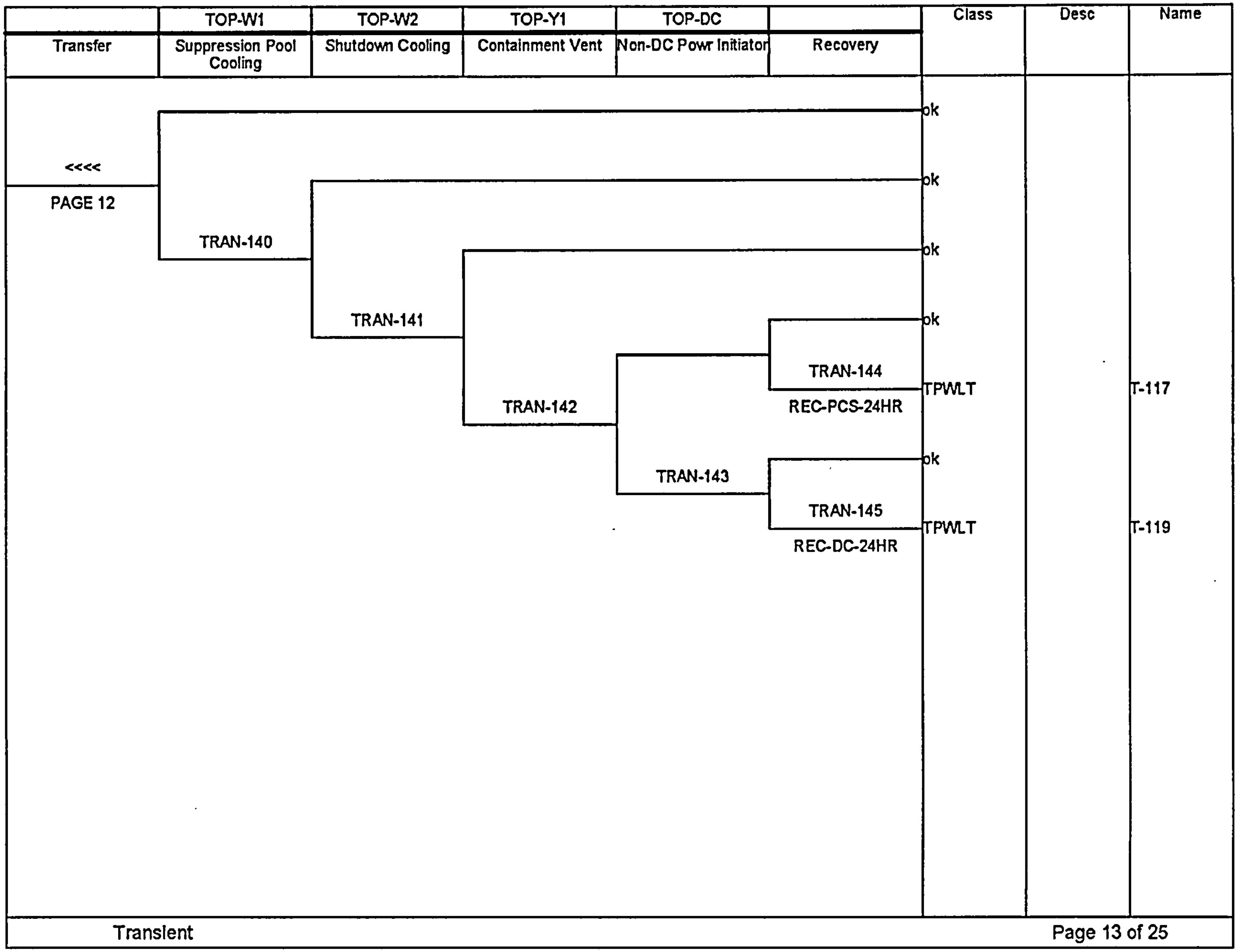
TWLT
T-100
Transient
Page 11 of 25


TOP-U2
RCIC
TOP-U1
HPCI
TOP-X2T
Depressurization
TOP-V1
Condensate
TOP-V2A
LPCS - Suppression Pool
TOP-V3A
LPCI - Suppression Pool
TOP-W1
Suppression Pool Cooling
TOP-W2
Shutdown Cooling
TOP-Y1
Containment Vent
TOP-DC
Non-DC Power Initiator
Recovery
Class
Desc
Name
Transfer
PAGE 1
TRAN-119
TRAN-120
TRAN-121
TRAN-122
TRAN-123
TRAN-124
TRAN-125
TRAN-126
REC-PCS-24HR
TRAN-127
REC-DC-24HR
TRAN-128
TRAN-129
TRAN-130
TRAN-131
TRAN-132
TRAN-133
REC-PCS-24HR
TRAN-134
REC-DC-24HR
TRAN-135
TRAN-136
TRAN-137
TRAN-138
REC-PCS-1HR
TRAN-139
REC-DC-1HR
TWOT
TPUV
TPUX
Page 13
Page 14
Page 15
Page 16
Page 20
T-131
T-133
T-145
T-147
T-148
T-238
T-240
Transient
Page 12 of 25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TOP-W2
TOP-Y1
TOP-DC
Class
Desc
Name
Transfer
Suppression Pool Cooling
Shutdown Cooling
Containment Vent
Non-DC Powr Initiator
Recovery
<<<<
PAGE 12
TRAN-140
TRAN-141
TRAN-142
TRAN-143
TRAN-144
REC-PCS-24HR
TRAN-145
REC-DC-24HR
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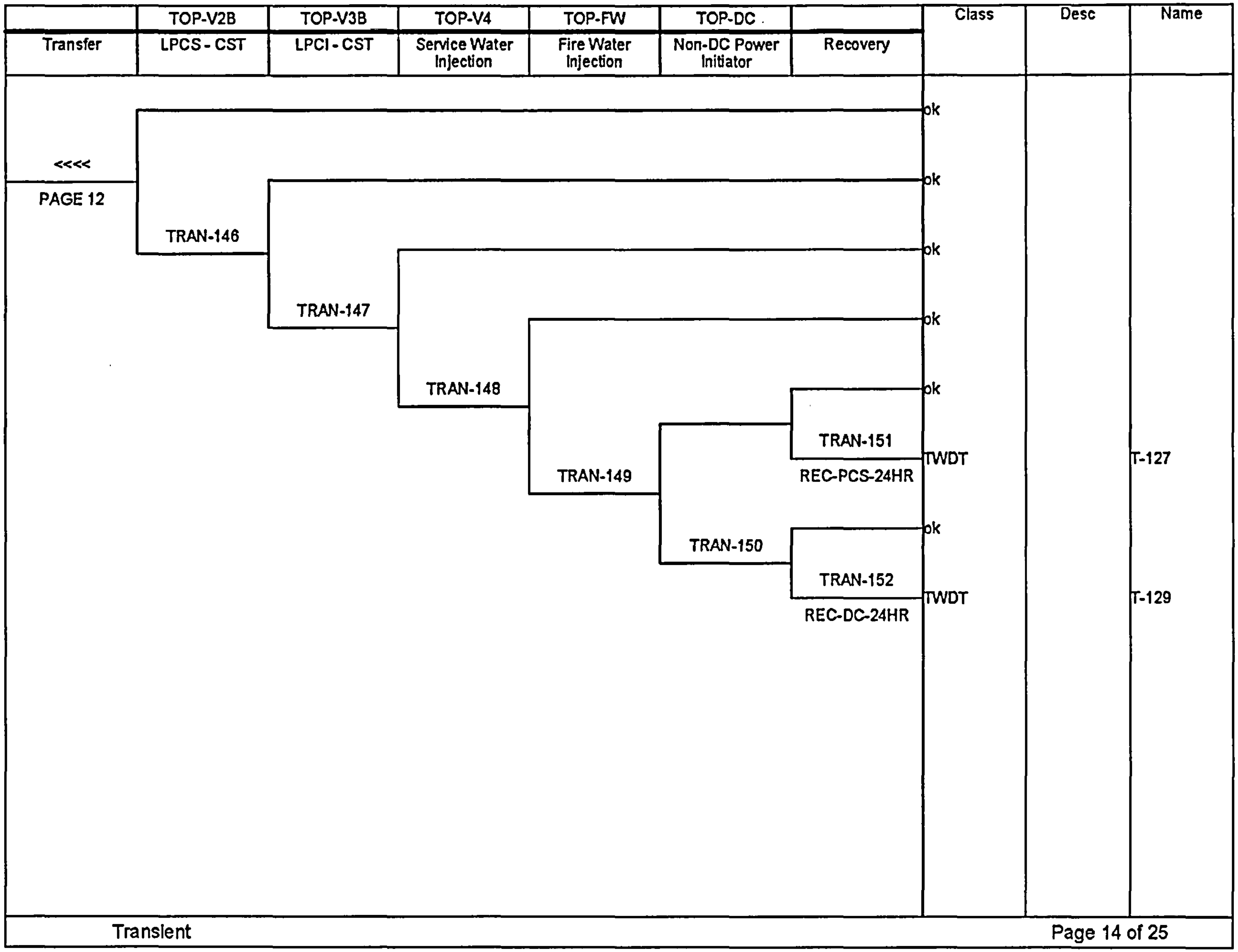
ok
ok
ok
TPWLT
T-117
ok
TPWLT
T-119
Transient
Page 13 of 25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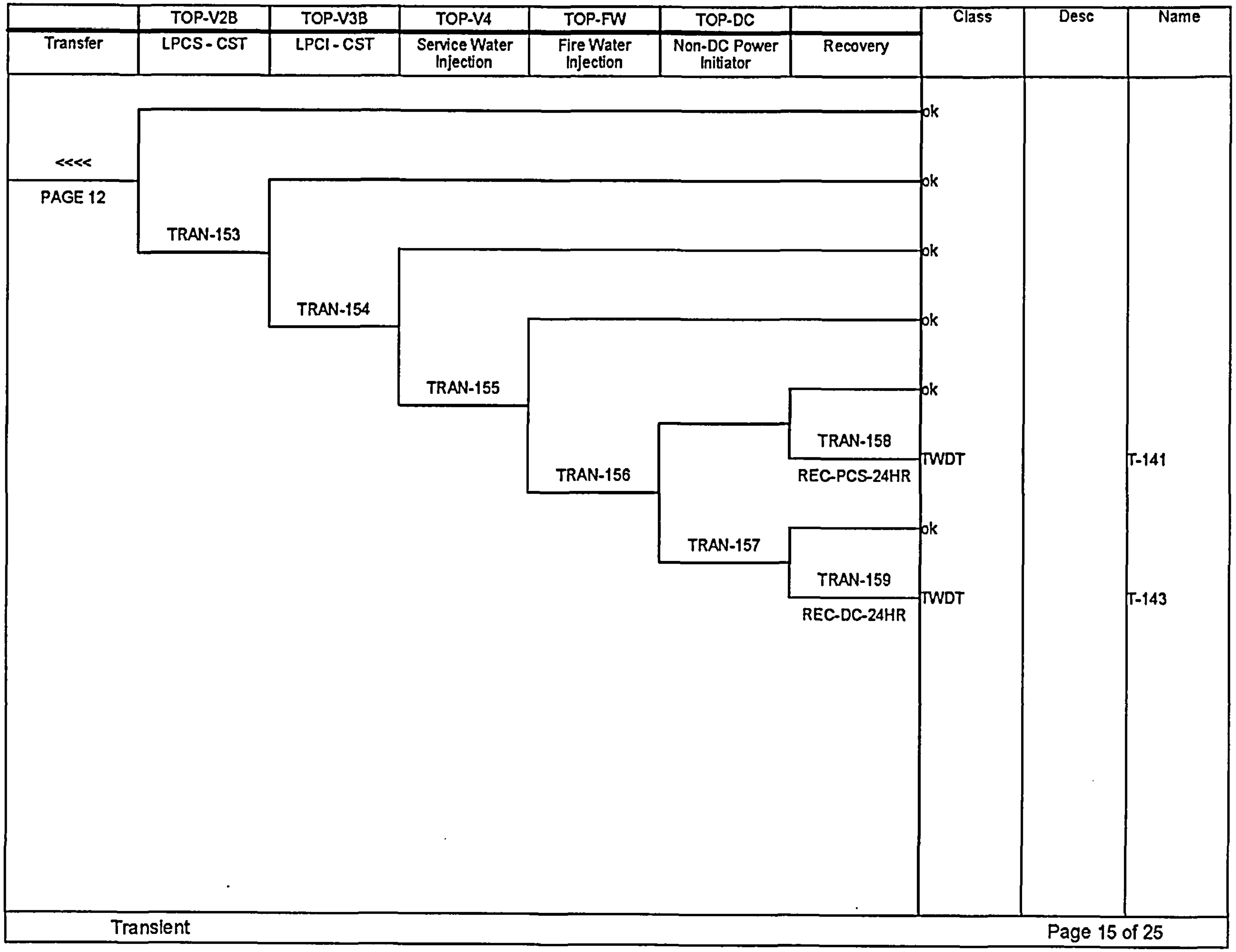
TOP-V3B
TOP-V4
TOP-FW
TOP-DC
Class
Desc
Name
Transfer
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Non-DC Power Initiator
Recovery
<<<<
PAGE 12
TRAN-146
TRAN-147
TRAN-148
TRAN-149
TRAN-150
TRAN-151
REC-PCS-24HR
TRAN-152
REC-DC-24HR
ok
ok
ok
ok
ok
TWDT
T-127
ok
TWDT
T-129
Transient
Page 14 of 25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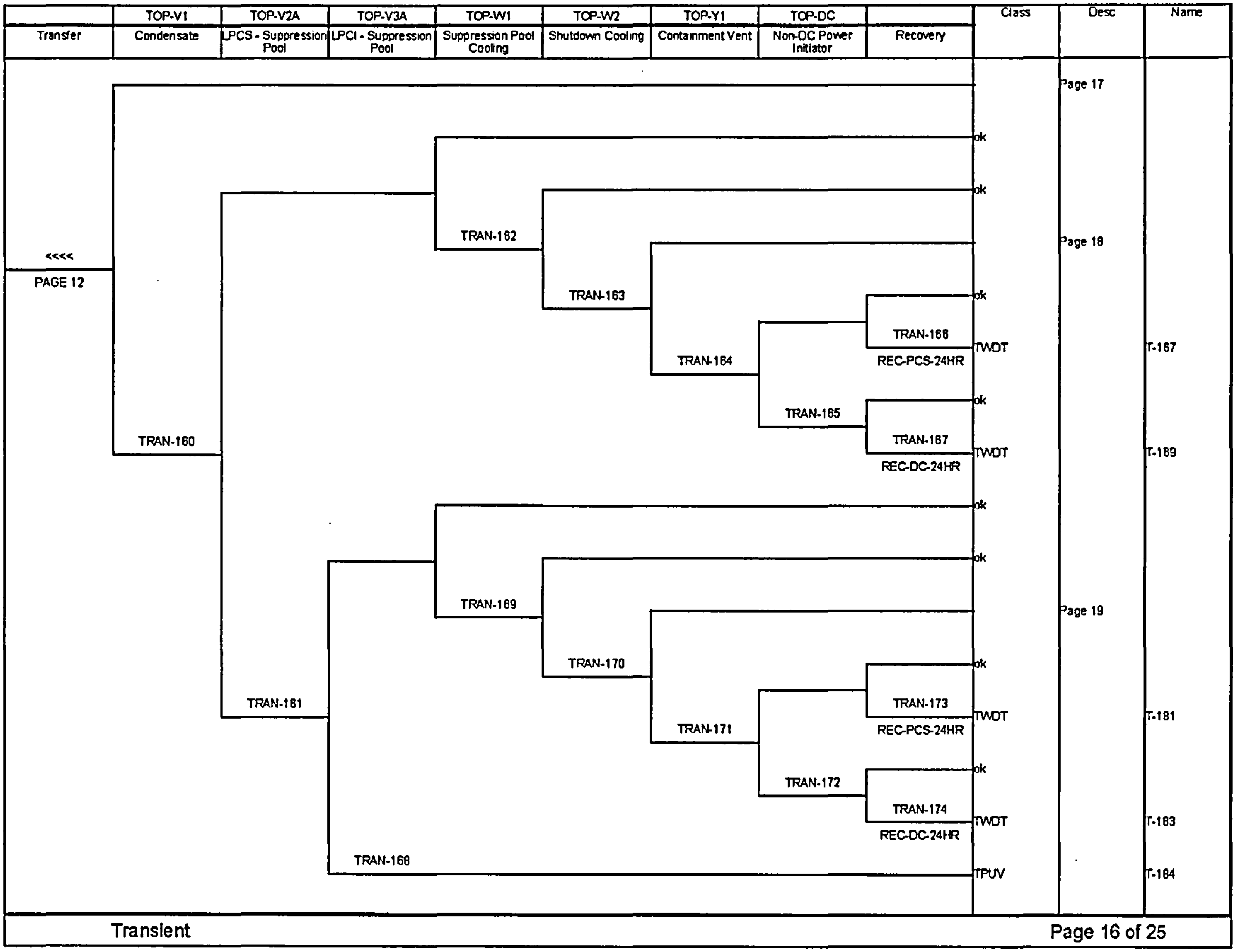
TOP-V3B
TOP-V4
TOP-FW
TOP-DC
Class
Desc
Name
Transfer
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Non-DC Power Initiator
Recovery
<<<<
PAGE 12
TRAN-153
TRAN-154
TRAN-155
TRAN-156
TRAN-157
TRAN-158
REC-PCS-24HR
TRAN-159
REC-DC-24HR
ok
ok
ok
ok
ok
TWDT
T-141
ok
TWDT
T-143
Transient
Page 15 of 25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TOP-V2A
TOP-V3A
TOP-W1
TOP-W2
TOP-Y1
TOP-DC
Class
Desc
Name
Transfer
Condensate
LPCS - Suppression Pool
LPCI - Suppression Pool
Suppression Pool Cooling
Shutdown Cooling
Containment Vent
Non-DC Power Initiator
Recovery
<<<<
PAGE 12
TRAN-160
TRAN-161
TRAN-162
TRAN-163
TRAN-164
TRAN-165
TRAN-166
REC-PCS-24HR
TRAN-167
REC-DC-24HR
TRAN-168
TRAN-169
TRAN-170
TRAN-171
TRAN-172
TRAN-173
REC-PCS-24HR
TRAN-174
REC-DC-24HR
Page 17
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ok
Page 18
ok
TWDT
T-167
ok
TWDT
T-169
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ok
Page 19
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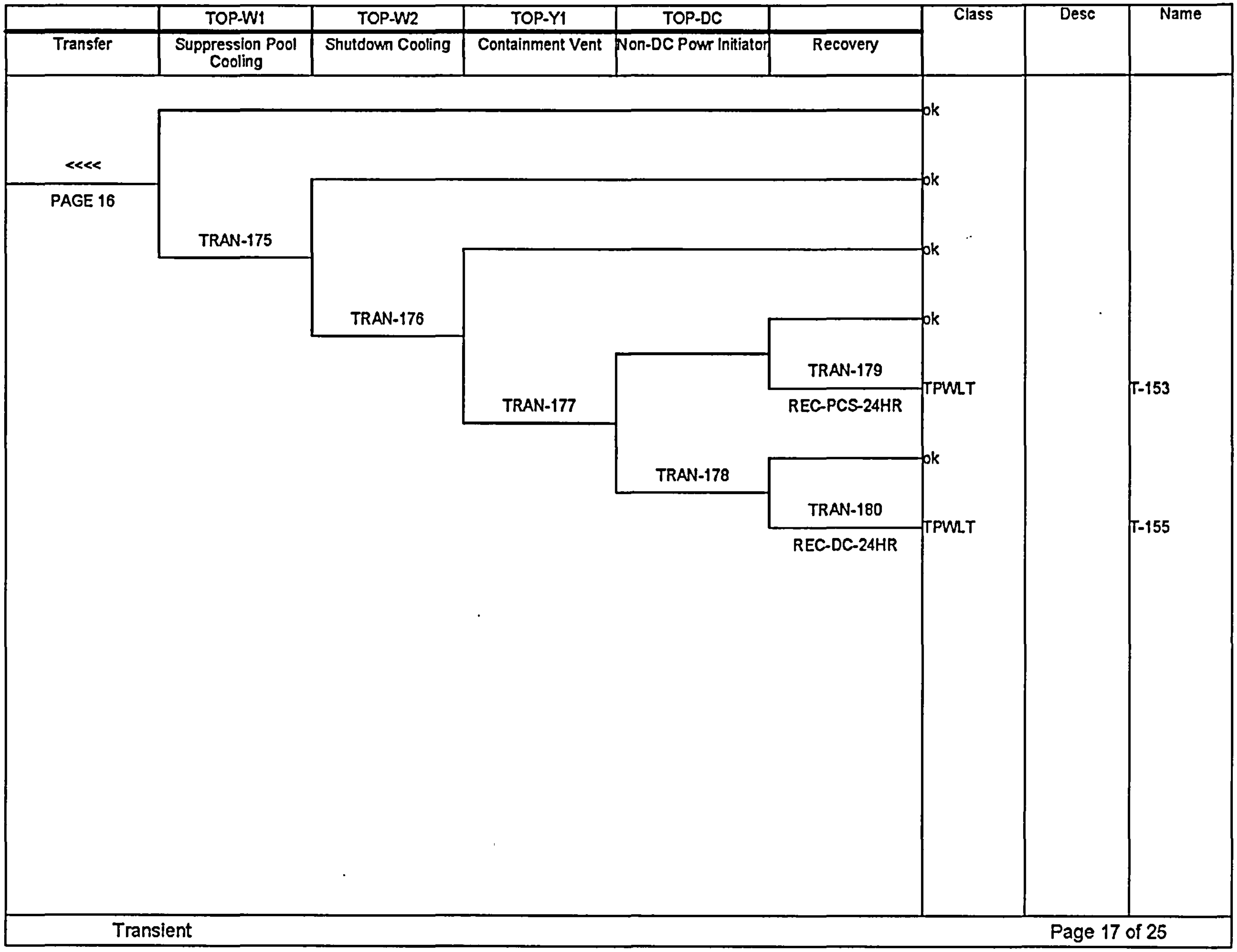
TWDT
T-181
ok
TWDT
T-183
TPUV
T-184
Transient
Page 16 of 25


TOP-W1
TOP-W2
TOP-Y1
TOP-DC
Class
Desc
Name
Transfer
Suppression Pool Cooling
Shutdown Cooling
Containment Vent
Non-DC Powr Initiator
Recovery
<<<<
PAGE 16
TRAN-175
TRAN-176
TRAN-177
TRAN-178
TRAN-179
REC-PCS-24HR
TRAN-180
REC-DC-24HR
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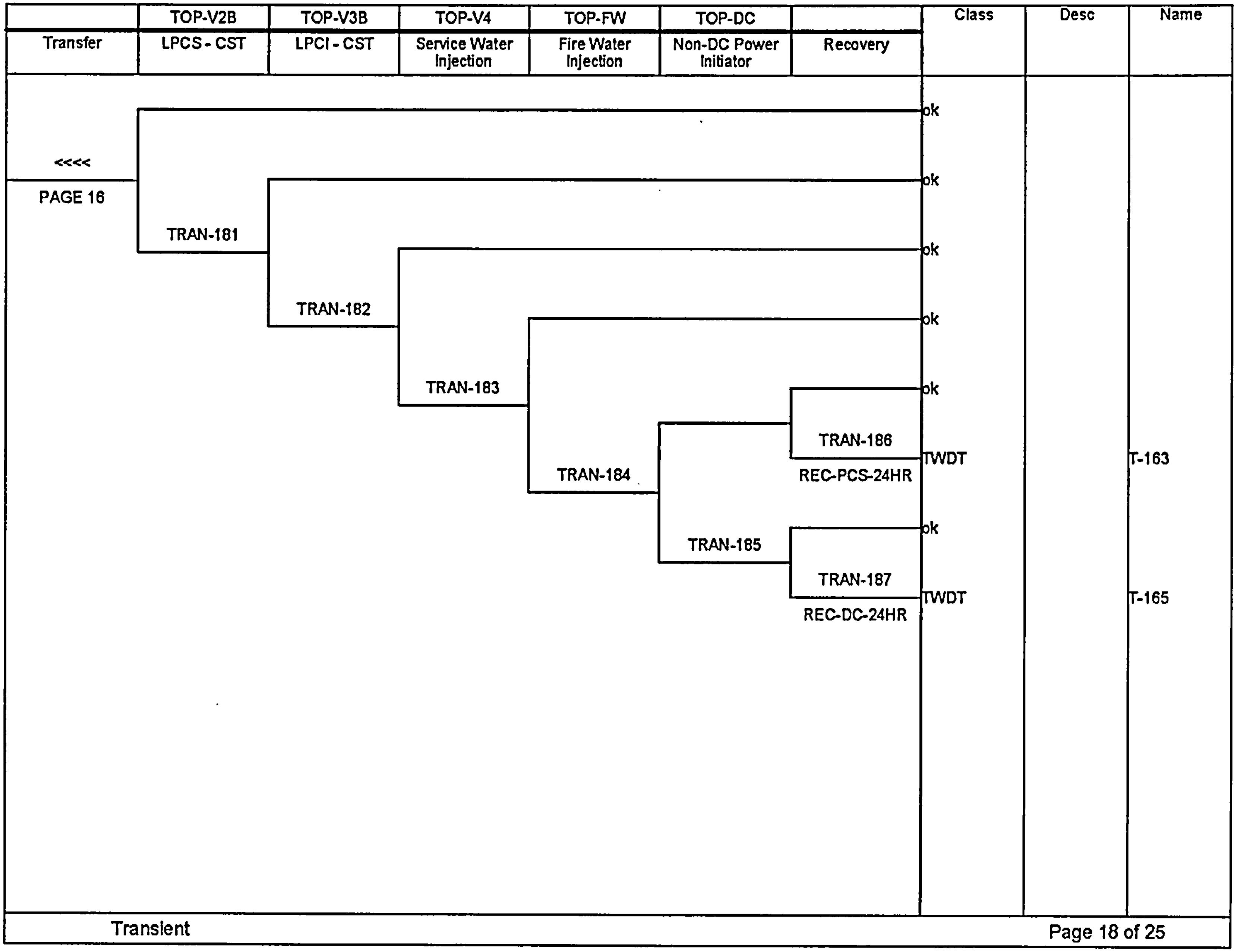
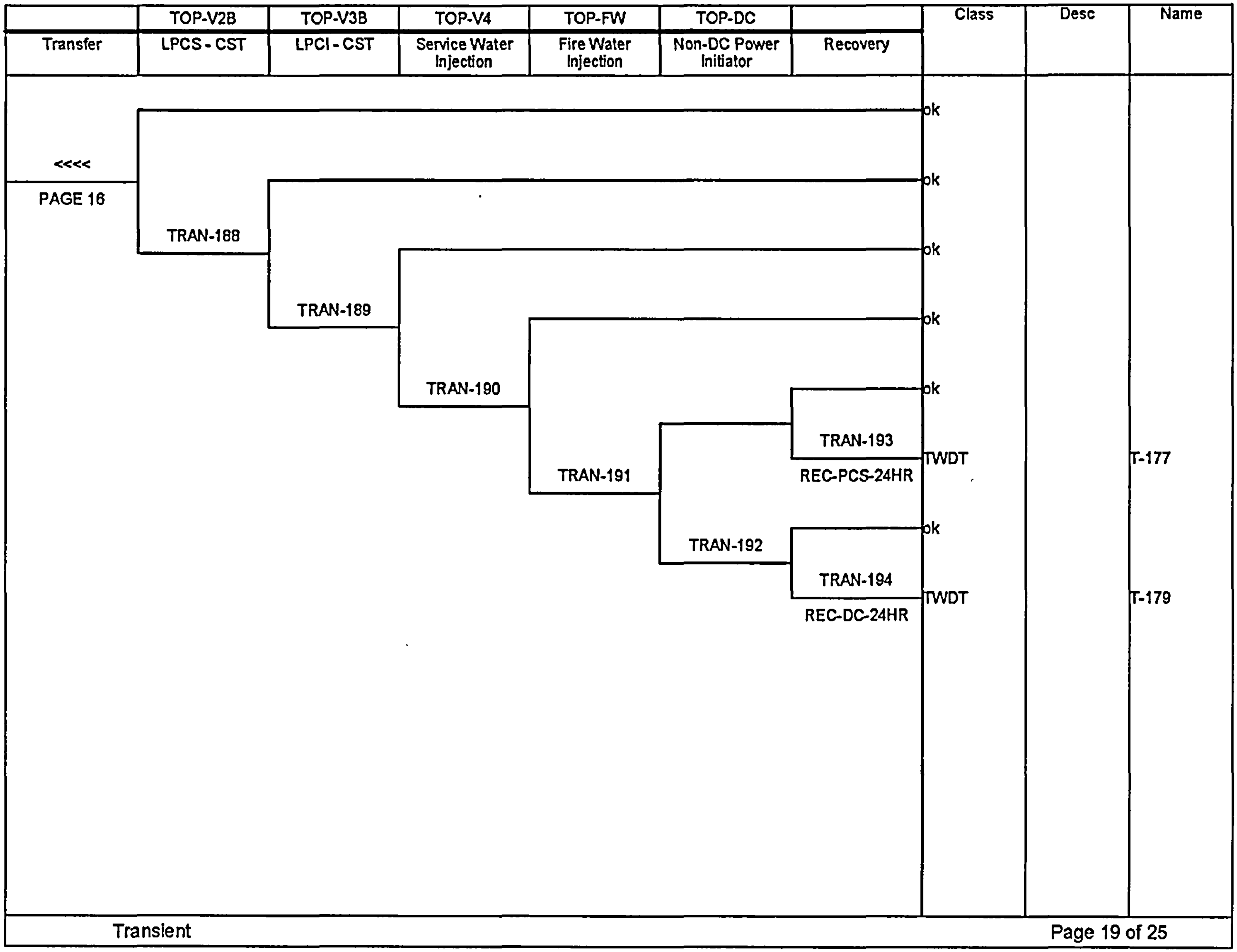
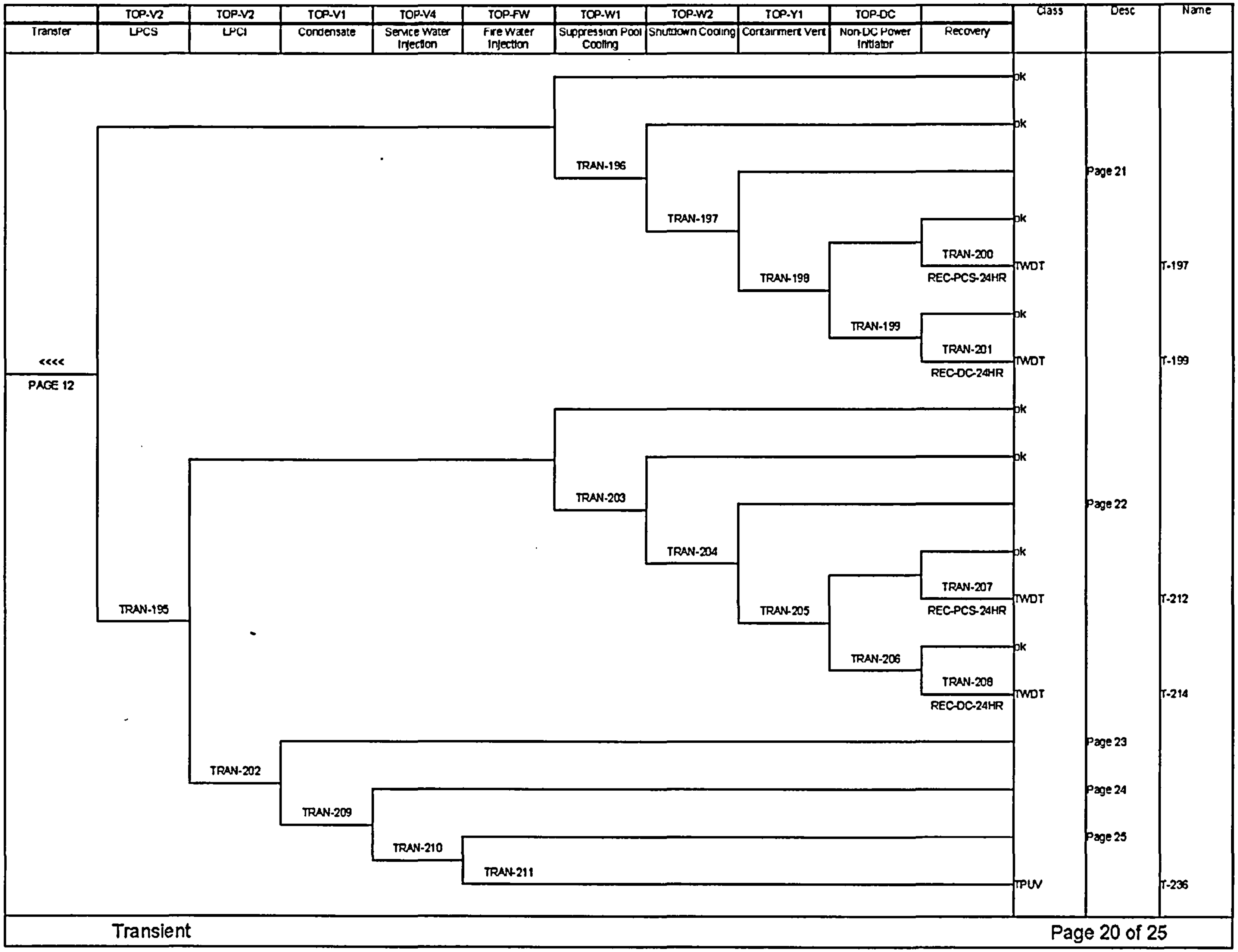
ok
ok
ok
TPWLT
T-153
ok
TPWLT
T-155
Transient
Page 17 of 25


TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-DC
Class
Desc
Name
Transfer
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Non-DC Power Initiator
Recovery
<<<<
PAGE 16
TRAN-181
TRAN-182
TRAN-183
TRAN-184
TRAN-185
TRAN-186
REC-PCS-24HR
TRAN-187
REC-DC-24HR
ok
ok
ok
ok
ok
TWDT
T-163
ok
TWDT
T-165
Transient
Page 18 of 25


TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-DC
Class
Desc
Name
Transfer
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Non-DC Power Initiator
Recovery
<<<<
PAGE 16
TRAN-188
TRAN-189
TRAN-190
TRAN-191
TRAN-192
TRAN-193
REC-PCS-24HR
TRAN-194
REC-DC-24HR
ok
ok
ok
ok
ok
TWDT
T-177
ok
TWDT
T-179
Transient
Page 19 of 25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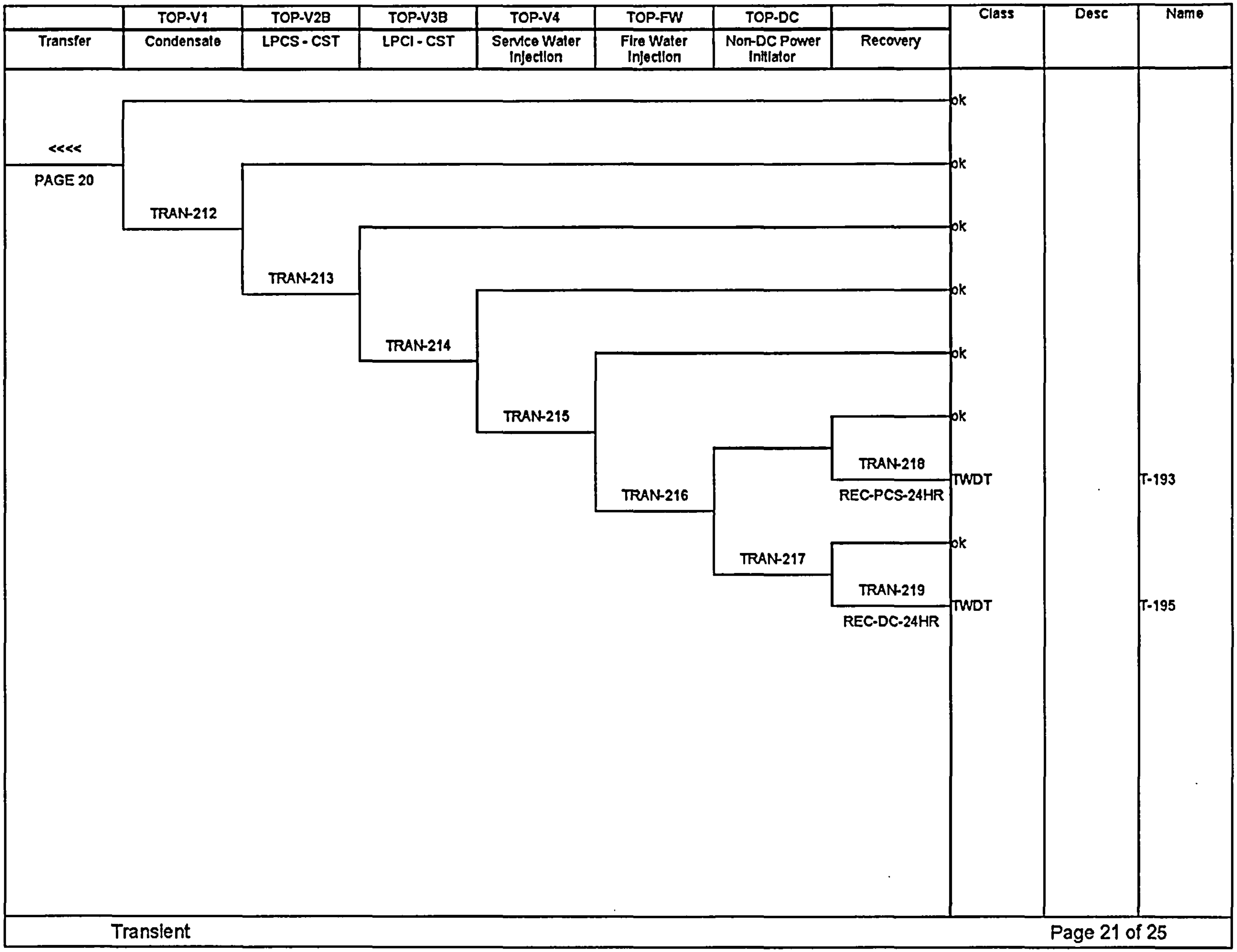
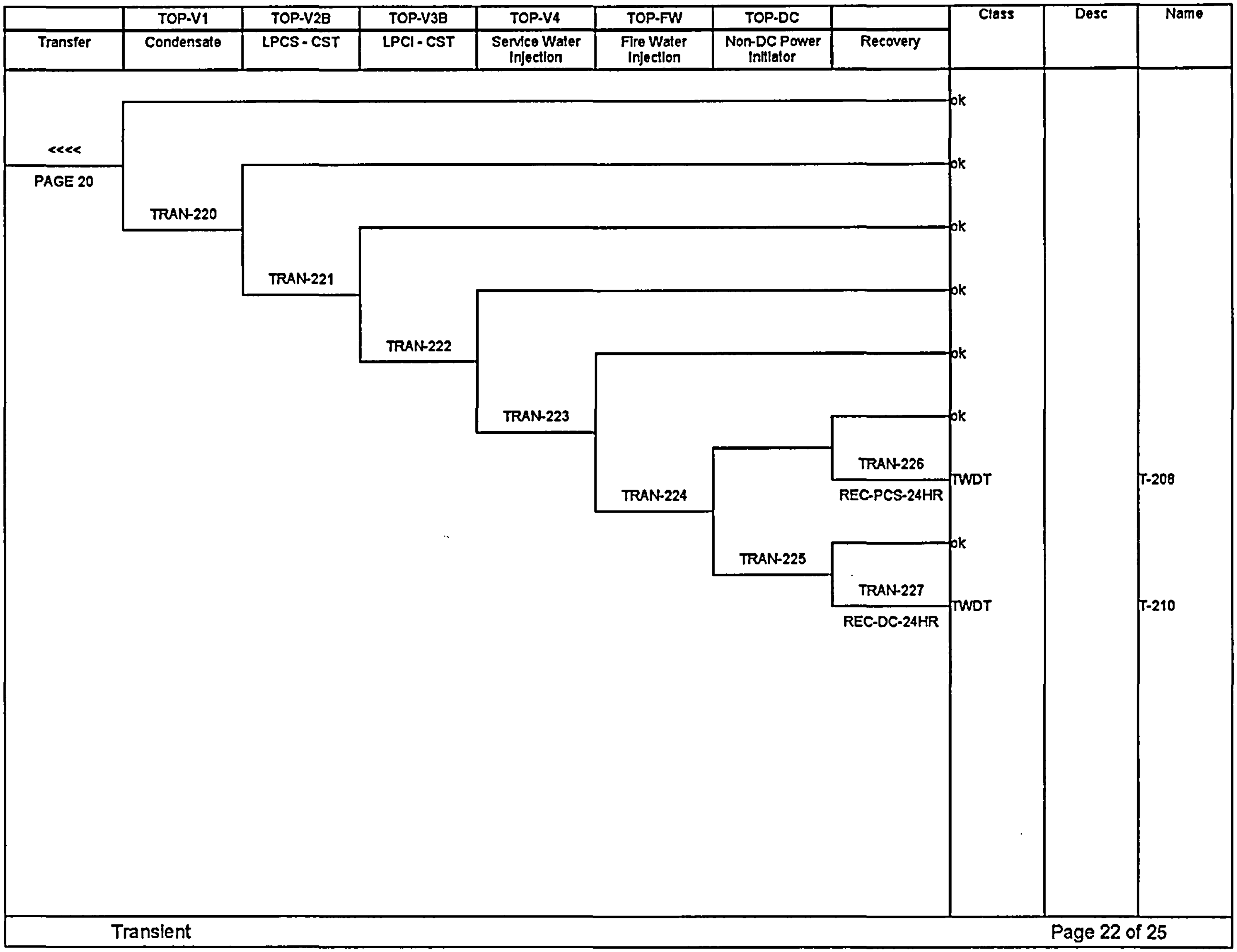
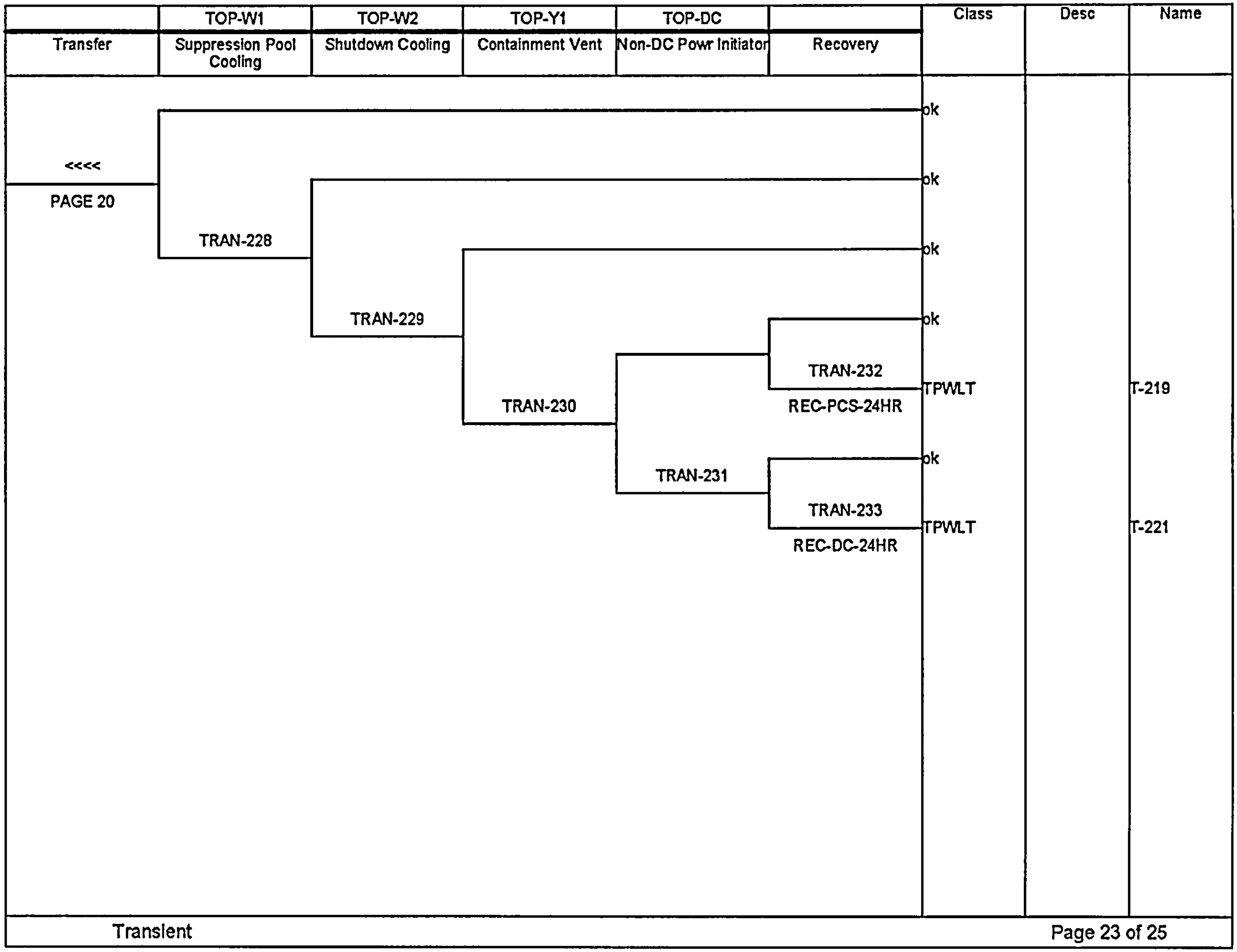
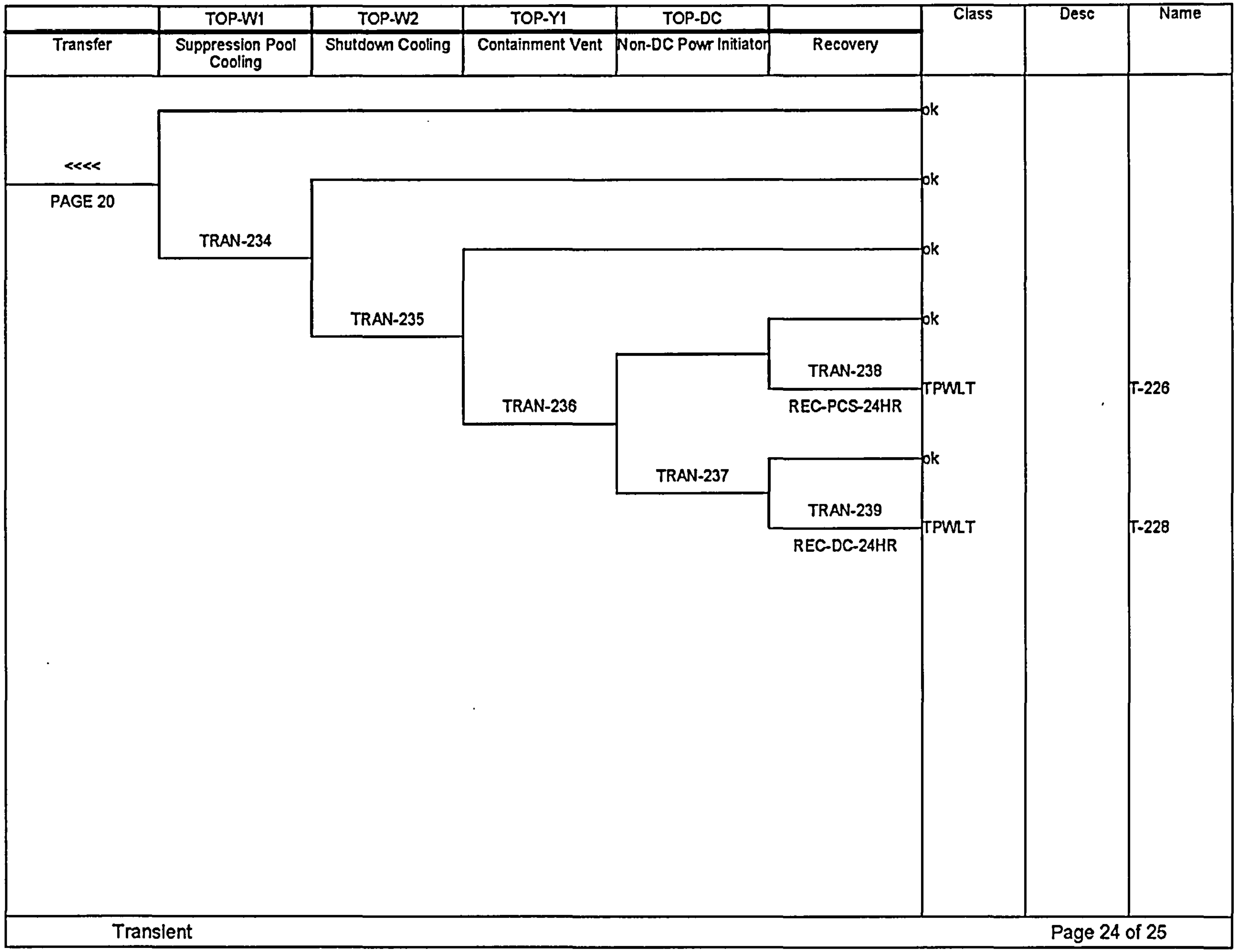
TOP-V2
TOP-V1
TOP-V4
TOP-FW
TOP-W1
TOP-W2
TOP-Y1
TOP-DC
Class
Desc
Name
Transfer
LPCS
LPCI
Condensate
Service Water Injection
Fire Water Injection
Suppression Pool Cooling
Shutdown Cooling
Containment Vent
Non-DC Power Initiator
Recovery
ok
ok
TRAN-196
Page 21
TRAN-197
ok
TRAN-200
TWDT
T-197
TRAN-198
REC-PCS-24HR
ok
TRAN-199
TRAN-201
TWDT
T-199
<<<<
PAGE 12
REC-DC-24HR
ok
ok
TRAN-203
Page 22
TRAN-204
ok
TRAN-207
TWDT
T-212
TRAN-195
TRAN-205
REC-PCS-24HR
ok
TRAN-206
TRAN-208
TWDT
T-214
REC-DC-24HR
Page 23
TRAN-202
Page 24
TRAN-209
Page 25
TRAN-210
TRAN-211
TPUV
T-236
Transient
Page 20 of 25


TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-DC
Class
Desc
Name
Transfer
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Non-DC Power Initiator
Recovery
ok
<<<<
PAGE 20
ok
TRAN-212
ok
TRAN-213
ok
TRAN-214
ok
TRAN-215
ok
TRAN-218
TWDT
T-193
TRAN-216
REC-PCS-24HR
ok
TRAN-217
TRAN-219
TWDT
T-195
REC-DC-24HR
Transient
Page 21 of 25


TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-DC
Class
Desc
Name
Transfer
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Non-DC Power Initiator
Recovery
<<<<
PAGE 20
TRAN-220
TRAN-221
TRAN-222
TRAN-223
TRAN-224
TRAN-225
TRAN-226
REC-PCS-24HR
TRAN-227
REC-DC-24HR
ok
ok
ok
ok
ok
ok
TWDT
T-208
ok
TWDT
T-210
Transient
Page 22 of 25


TOP-W1
TOP-W2
TOP-Y1
TOP-DC
Class
Desc
Name
Transfer
Suppression Pool Cooling
Shutdown Cooling
Containment Vent
Non-DC Powr Initiator
Recovery
ok
<<<<
PAGE 20
ok
TRAN-228
ok
TRAN-229
ok
TRAN-232
TPWLT
T-219
TRAN-230
REC-PCS-24HR
ok
TRAN-231
TRAN-233
TPWLT
T-221
REC-DC-24HR
Transient
Page 23 of 25


TOP-W1
TOP-W2
TOP-Y1
TOP-DC
Class
Desc
Name
Transfer
Suppression Pool Cooling
Shutdown Cooling
Containment Vent
Non-DC Powr Initiator
Recovery
<<<<
PAGE 20
TRAN-234
TRAN-235
TRAN-236
TRAN-237
TRAN-238
REC-PCS-24HR
TRAN-239
REC-DC-24HR
ok
ok
ok
ok
TPWLT
T-226
ok
TPWLT
T-228
Transient
Page 24 of 25

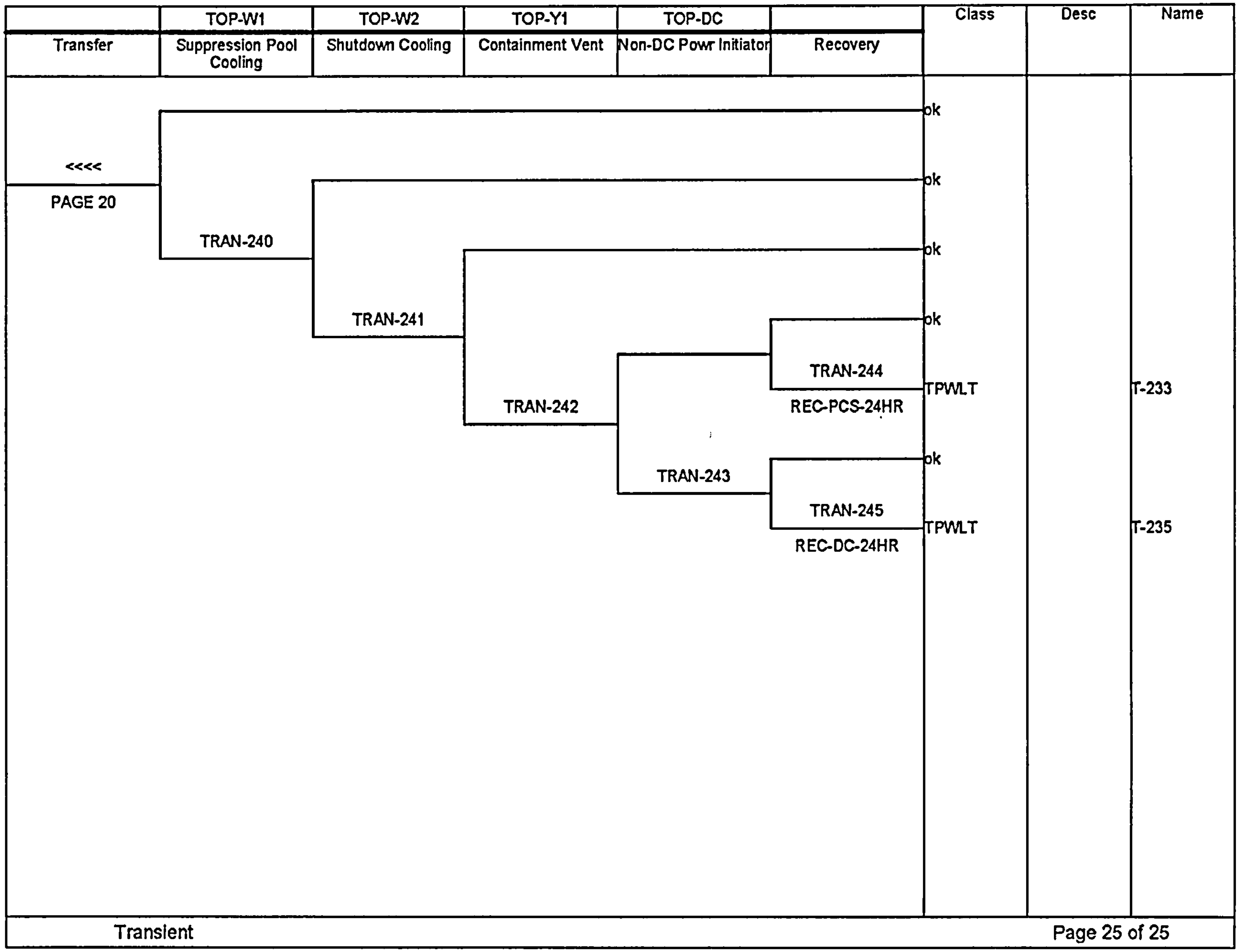
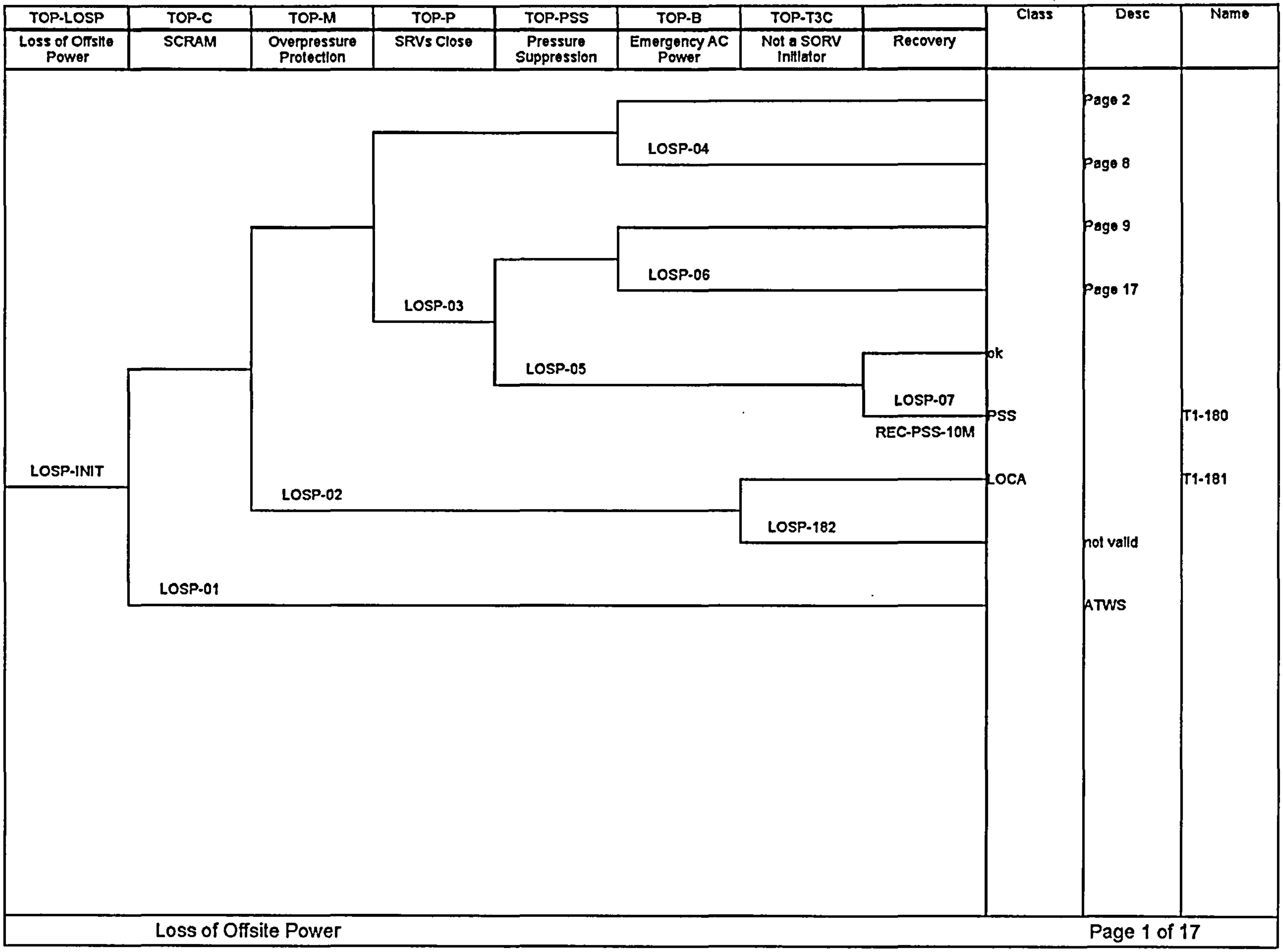
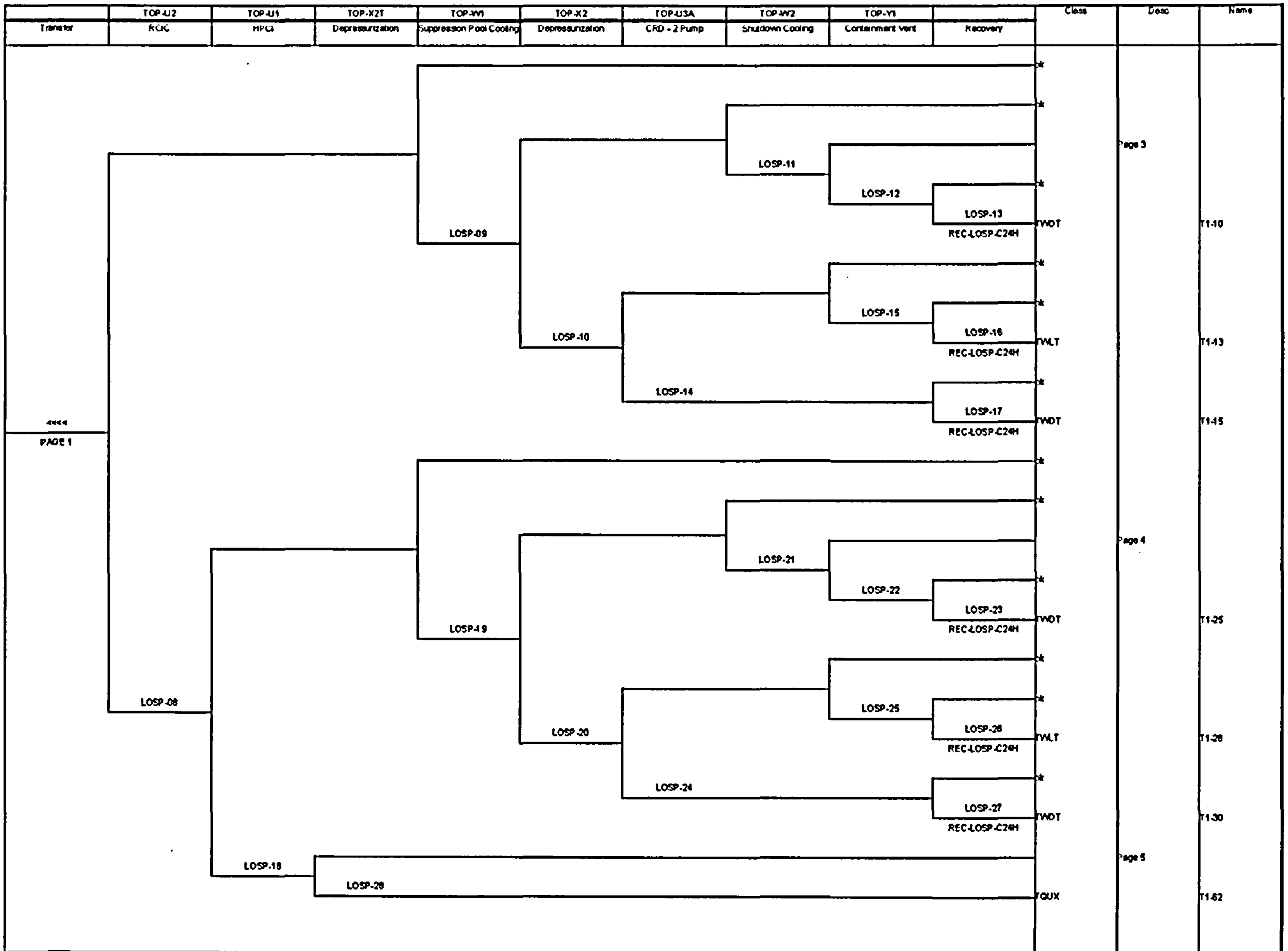

TOP-W1
TOP-W2
TOP-Y1
TOP-DC
Class
Desc
Name
Transfer
Suppression Pool Cooling
Shutdown Cooling
Containment Vent
Non-DC Powr Initiator
Recovery
<<<<
PAGE 20
TRAN-240
TRAN-241
TRAN-242
TRAN-243
TRAN-244
TRAN-245
REC-PCS-24HR
REC-DC-24HR
ok
ok
ok
ok
TPWLT
T-233
ok
TPWLT
T-235
Transient
Page 25 of 25

Figure C.3-2 Loss of Offsite Power Event Tree





Loss of Offsite Power

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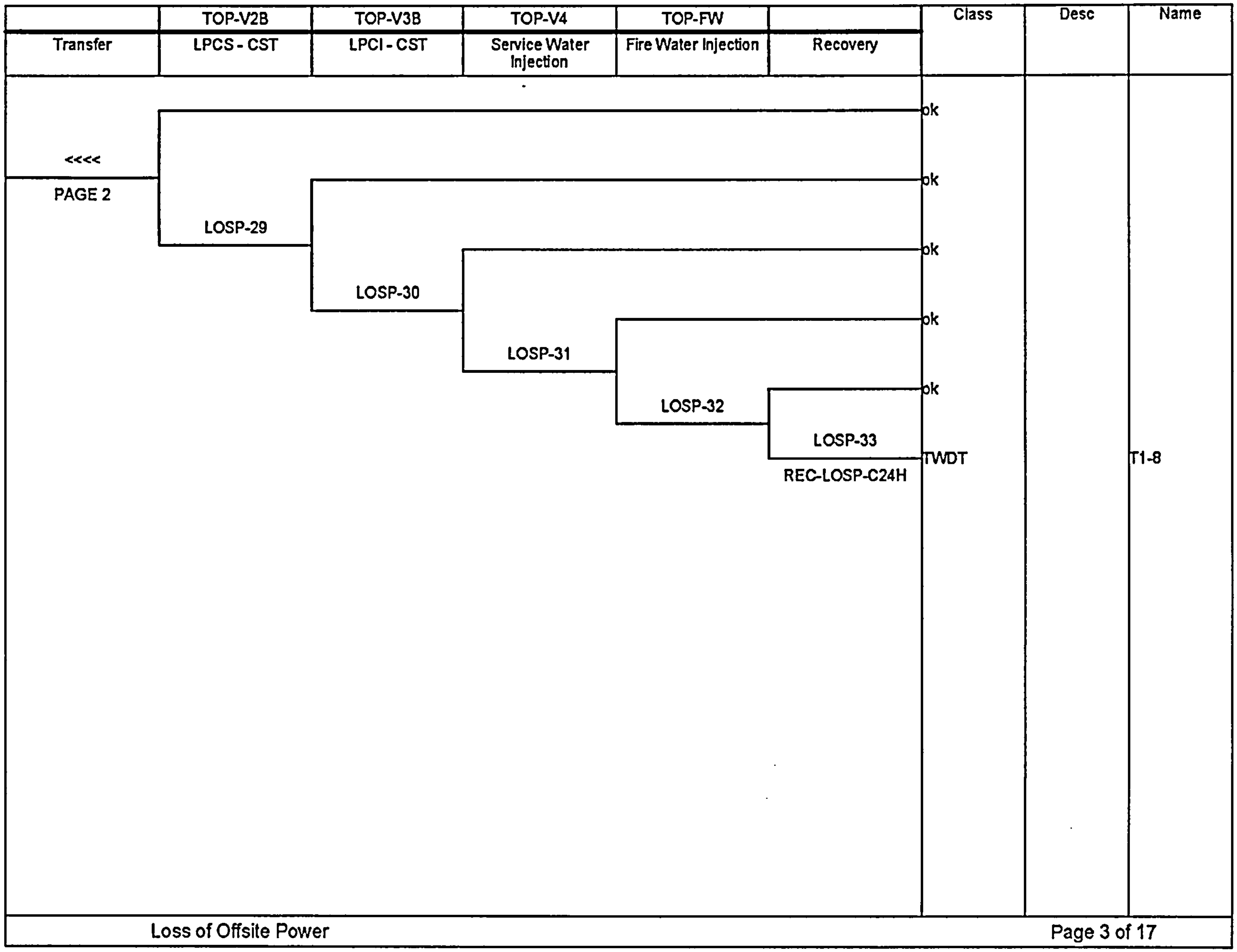
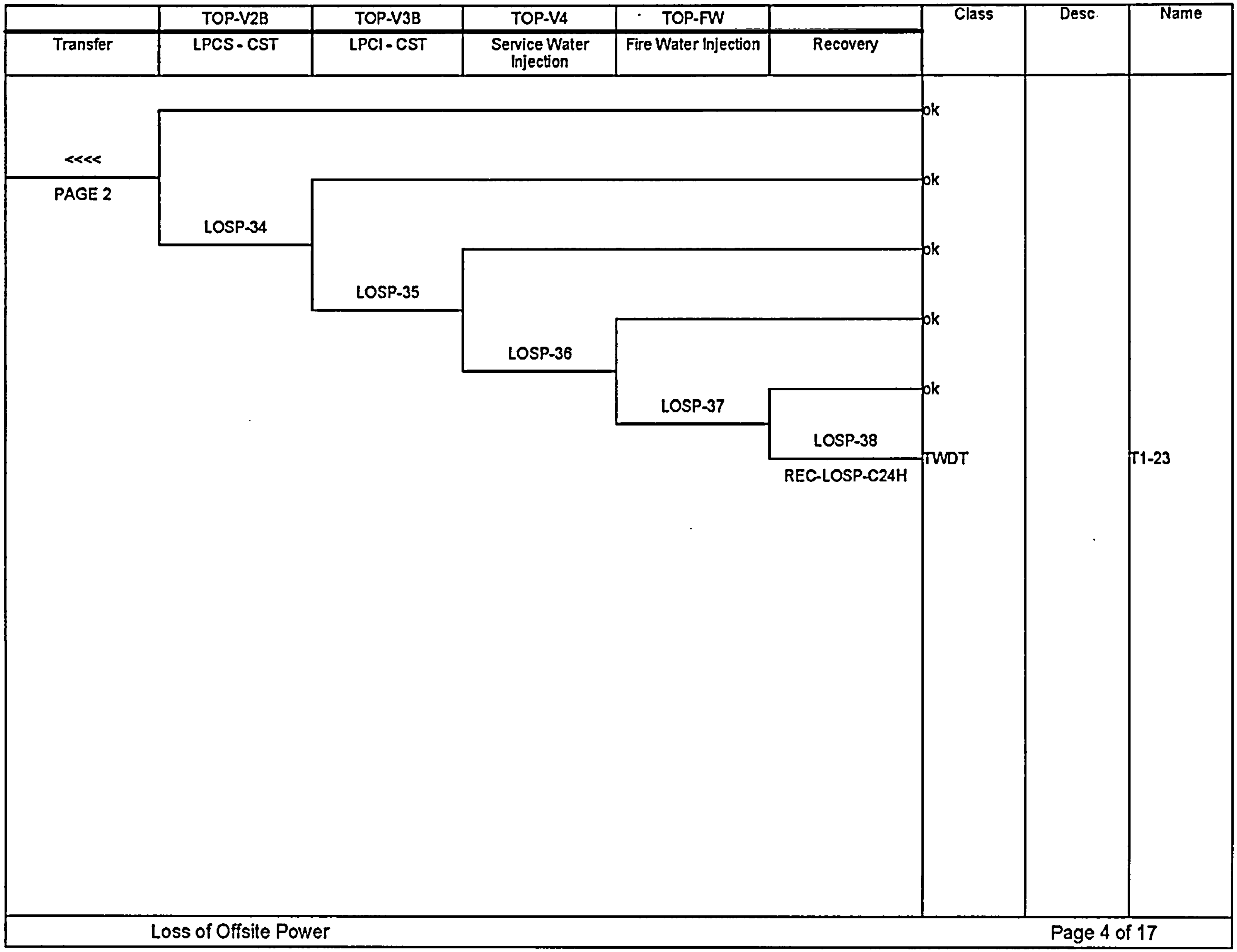
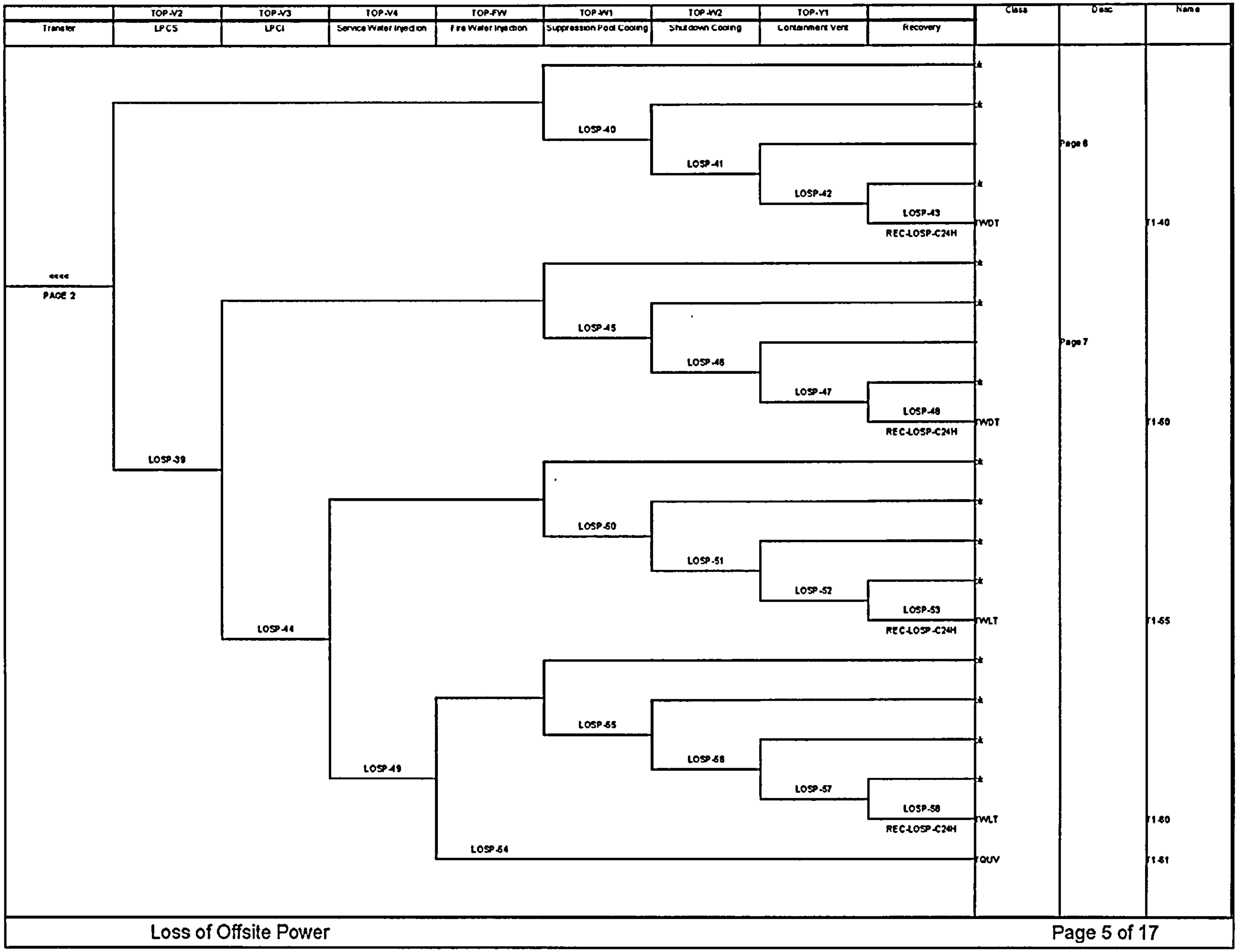

TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
Class
Desc
Name
Transfer
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Recovery
<<<<
PAGE 2
LOSP-29
LOSP-30
LOSP-31
LOSP-32
LOSP-33
REC-LOSP-C24H
ok
ok
ok
ok
ok
TWDT
T1-8
Loss of Offsite Power
Page 3 of 17


TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
Class
Desc.
Name
Transfer
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Recovery
<<<<
PAGE 2
LOSP-34
LOSP-35
LOSP-36
LOSP-37
LOSP-38
REC-LOSP-C24H
ok
ok
ok
ok
ok
TWDT
T1-23
Loss of Offsite Power
Page 4 of 17


Transfer
TOP-V2
LPCS
TOP-V3
LPCI
TOP-V4
Service Water Injection
TOP-FW
Fire Water Injection
TOP-W1
Suppression Pool Cooling
TOP-W2
Shutdown Cooling
TOP-Y1
Containment Vent
Recovery
Class
Desc
Name
<<<<
PAGE 2
LOSP-39
LOSP-40
LOSP-41
LOSP-42
LOSP-43
REC-LOSP-C24H
TWDT
T1-40
Page 6
LOSP-44
LOSP-45
LOSP-46
LOSP-47
LOSP-48
REC-LOSP-C24H
TWDT
T1-50
Page 7
LOSP-49
LOSP-50
LOSP-51
LOSP-52
LOSP-53
REC-LOSP-C24H
TWLT
T1-55
LOSP-54
LOSP-55
LOSP-56
LOSP-57
LOSP-58
REC-LOSP-C24H
TWLT
T1-60
TQUV
T1-61
Loss of Offsite Power
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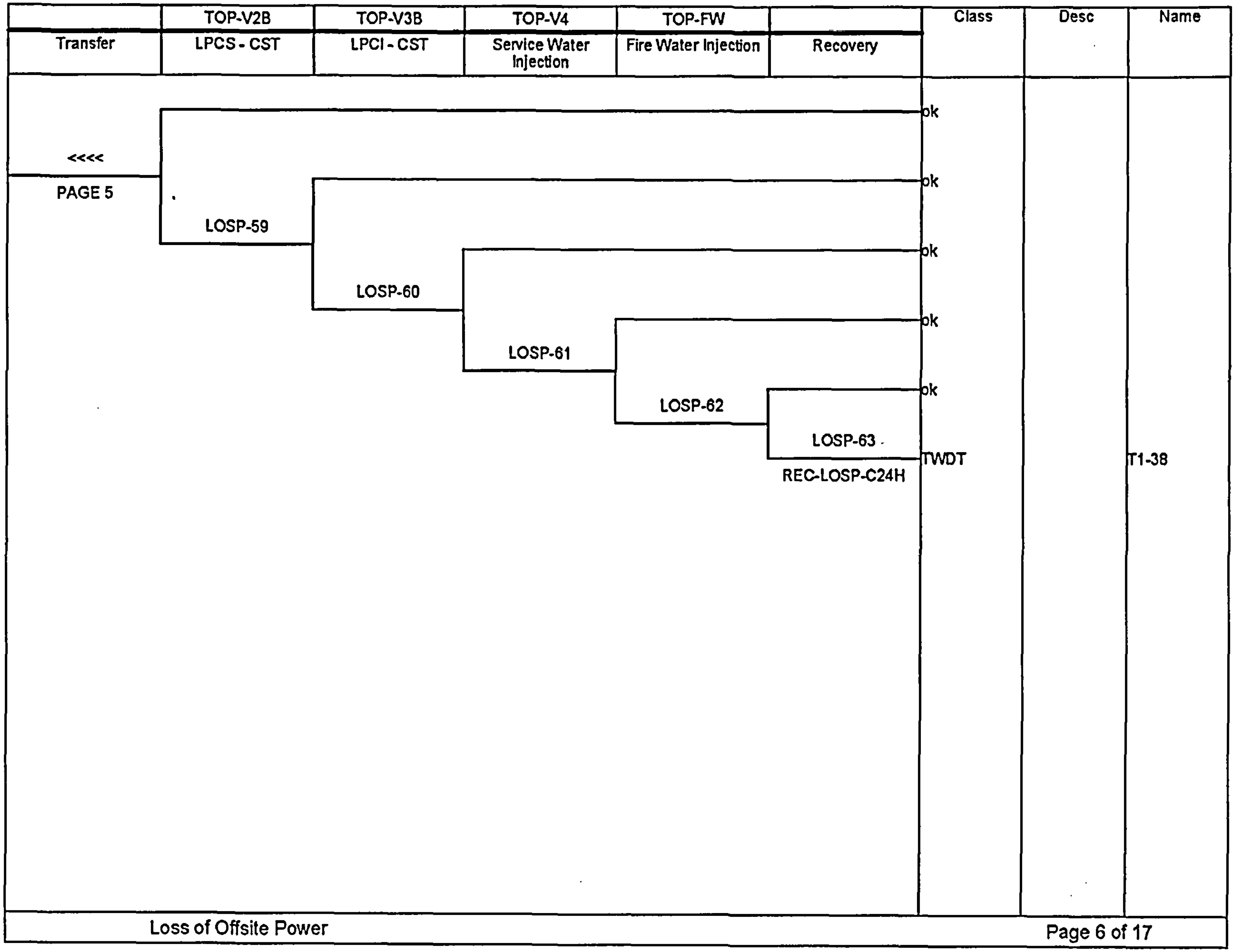
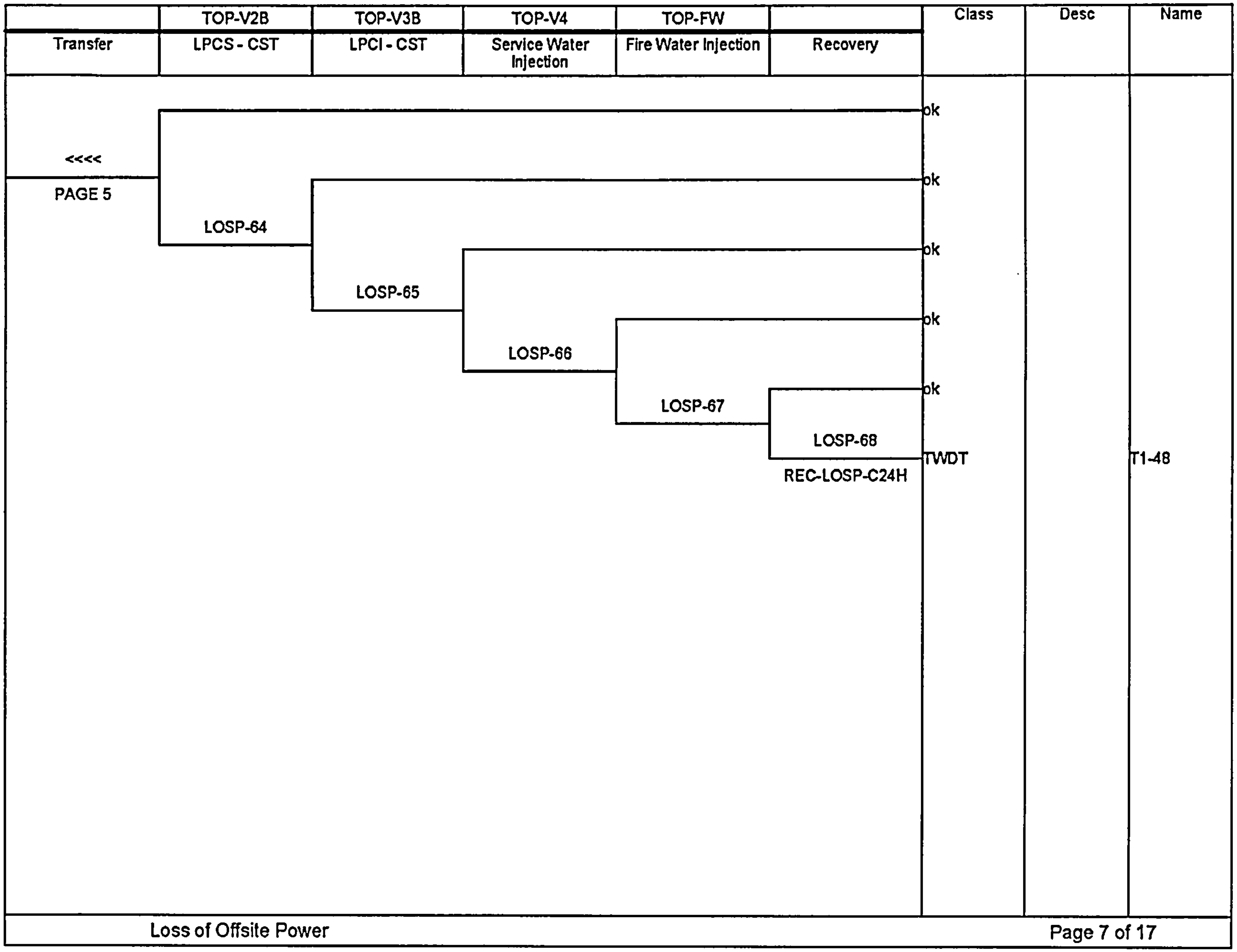
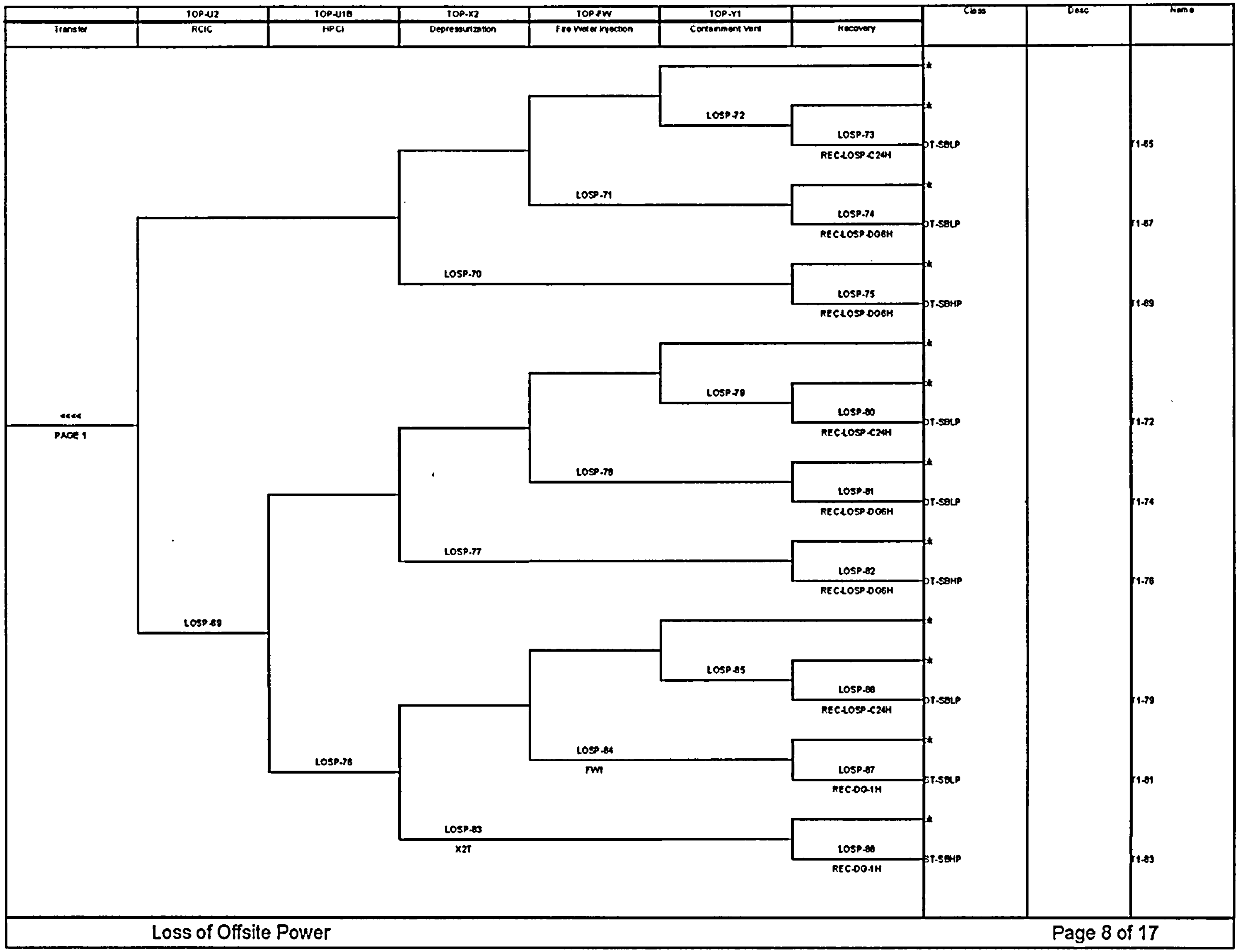
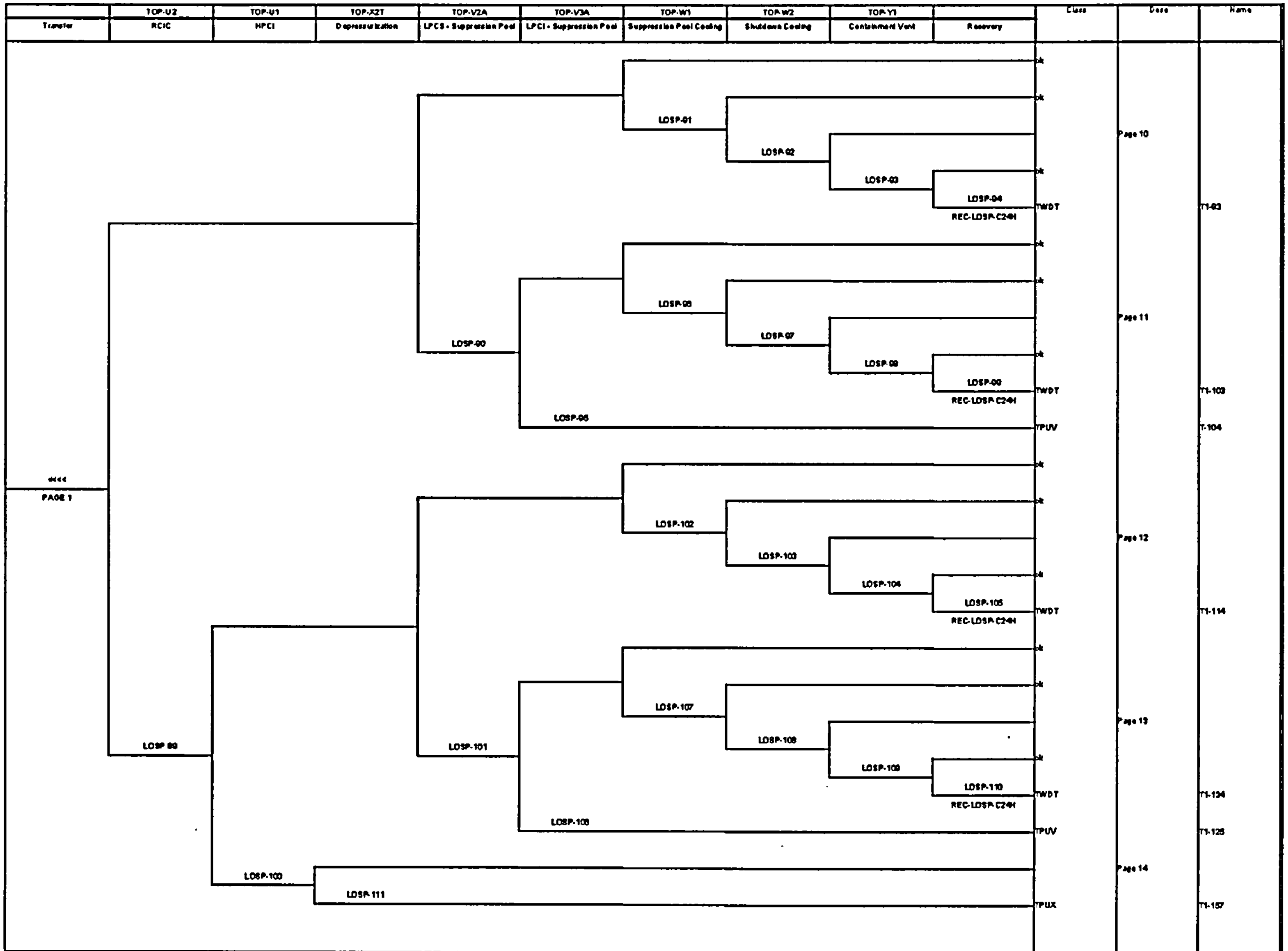

Transfer
TOP-V2B
LPCS - CST
TOP-V3B
LPCI - CST
TOP-V4
Service Water Injection
TOP-FW
Fire Water Injection
Recovery
Class
Desc
Name
<<<<
PAGE 5
LOSP-59
LOSP-60
LOSP-61
LOSP-62
LOSP-63
REC-LOSP-C24H
ok
ok
ok
ok
ok
TWDT
T1-38
Loss of Offsite Power
Page 6 of 17


TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
Class
Desc
Name
Transfer
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Recovery
<<<<
PAGE 5
LOSP-64
LOSP-65
LOSP-66
LOSP-67
LOSP-68
REC-LOSP-C24H
ok
ok
ok
ok
ok
TWDT
T1-48
Loss of Offsite Power
Page 7 of 17


TOP-U2
TOP-U1B
TOP-X2
TOP-FW
TOP-Y1
Class
Desc
Name
Transfer
RCIC
HPCI
Depressurization
Fire Water Injection
Containment Vent
Recovery
<<<<
PAGE 1
LOSP-69
LOSP-70
LOSP-71
LOSP-72
LOSP-73
REC-LOSP-C24H
LOSP-74
REC-LOSP-DG6H
LOSP-75
REC-LOSP-DG6H
LOSP-76
LOSP-77
LOSP-78
LOSP-79
LOSP-80
REC-LOSP-C24H
LOSP-81
REC-LOSP-DG6H
LOSP-82
REC-LOSP-DG6H
LOSP-83
X2T
LOSP-84
FWI
LOSP-85
LOSP-86
REC-LOSP-C24H
LOSP-87
REC-DG-1H
LOSP-88
REC-DG-1H
T-SBLP
T-SBLP
T-SBHP
T-SBLP
T-SBLP
T-SBHP
T-SBLP
T-SBLP
T-SBHP
1-65
1-67
1-69
1-72
1-74
1-76
1-79
1-81
1-83
Loss of Offsite Power
Page 8 of 17



Loss of Offsite Power

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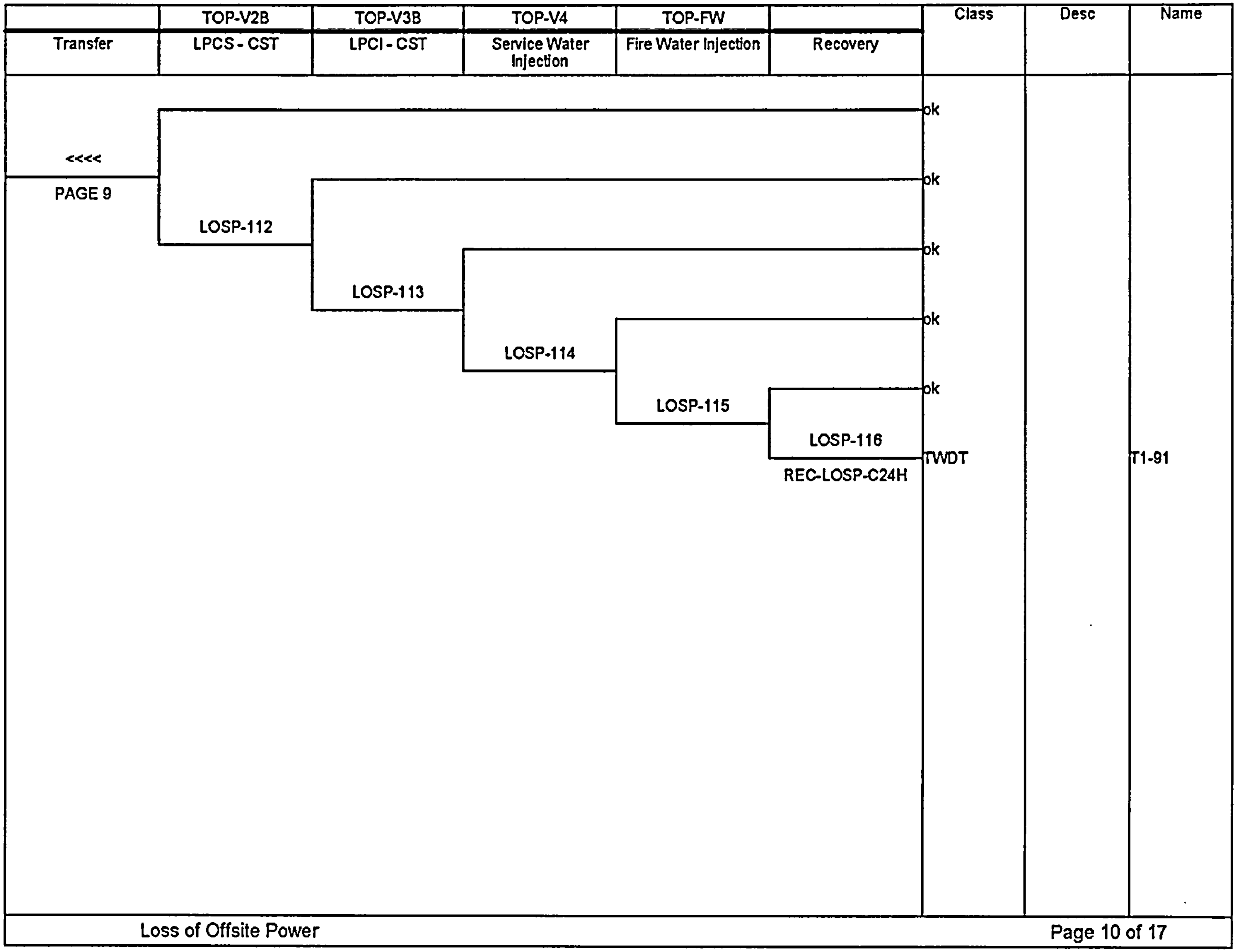
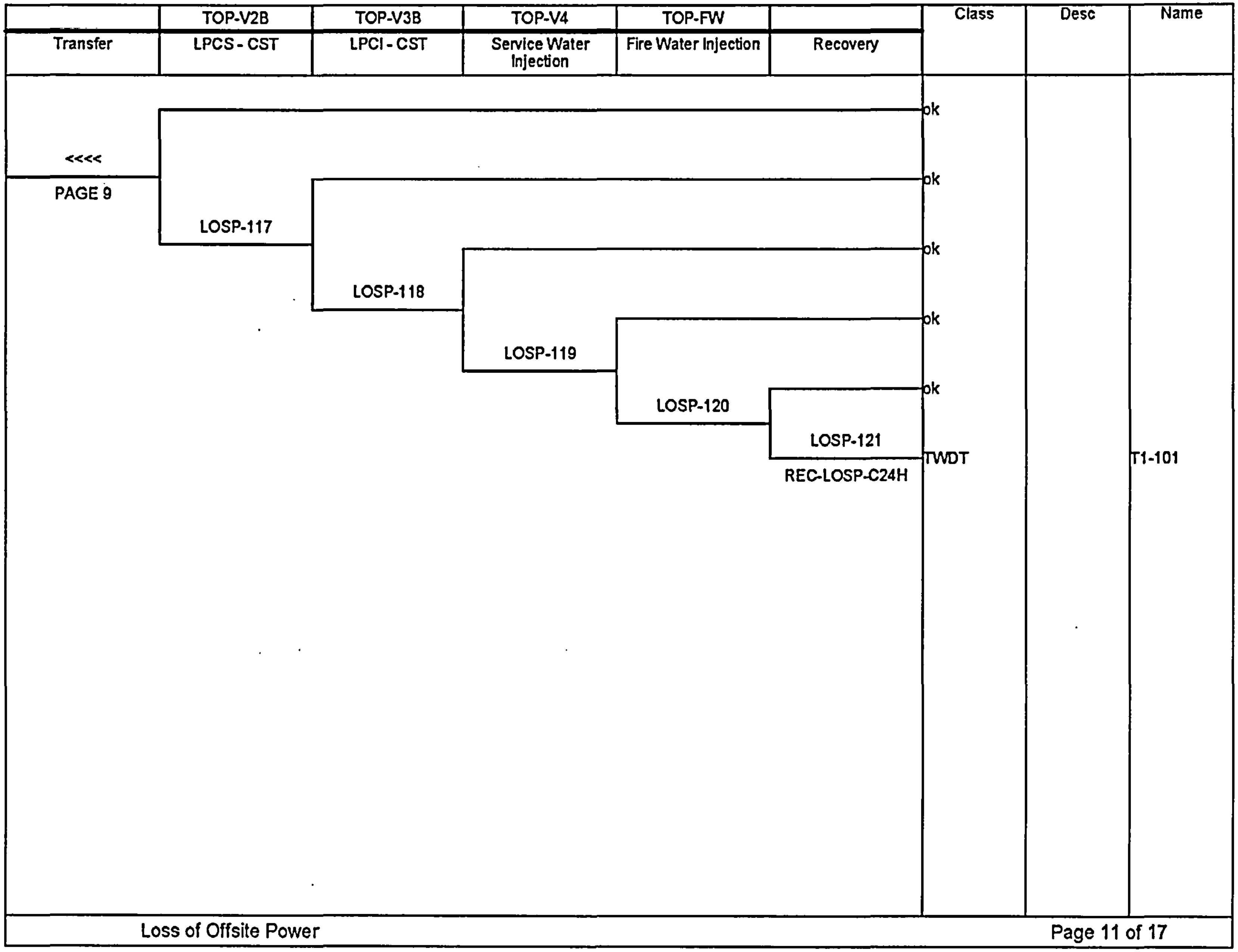
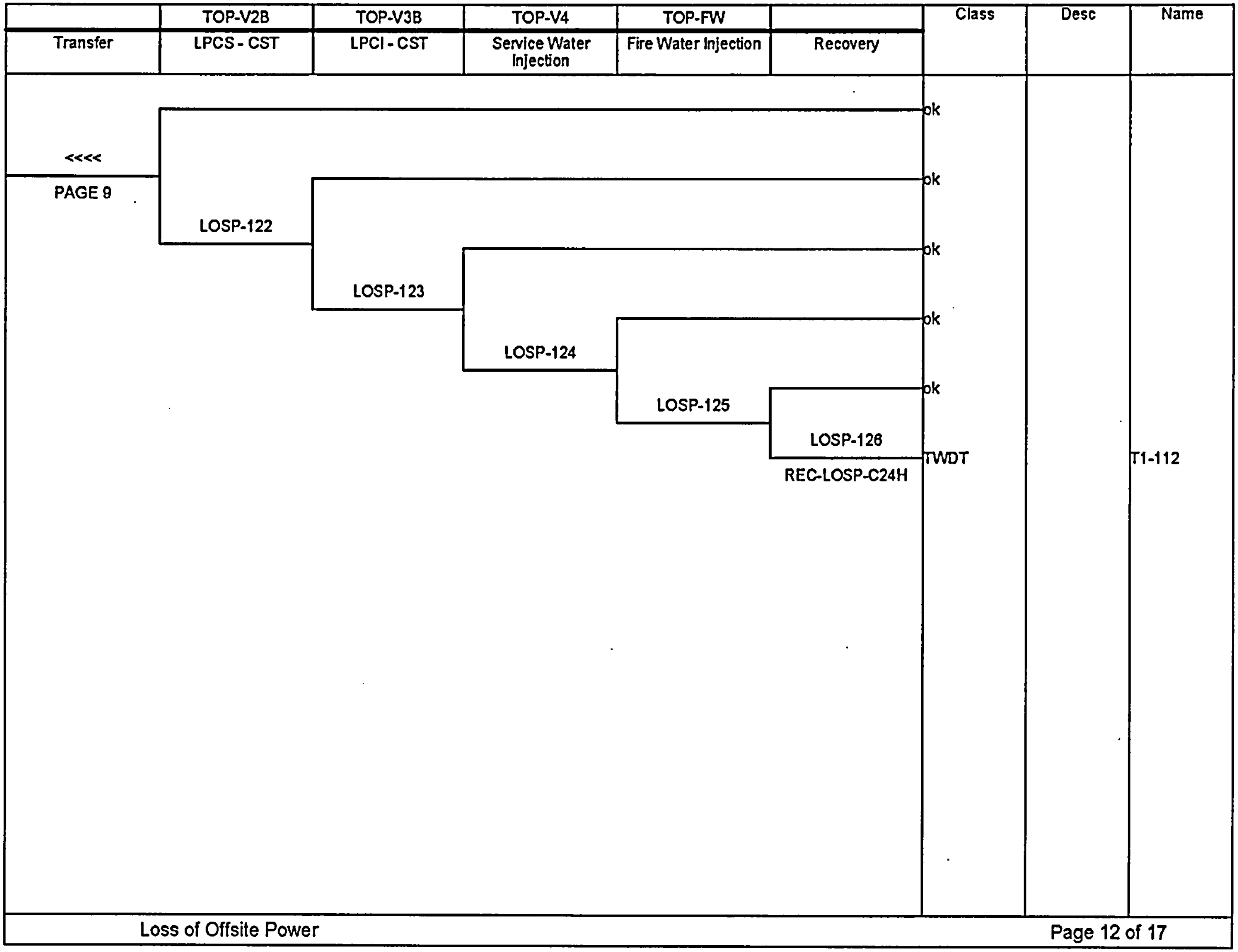
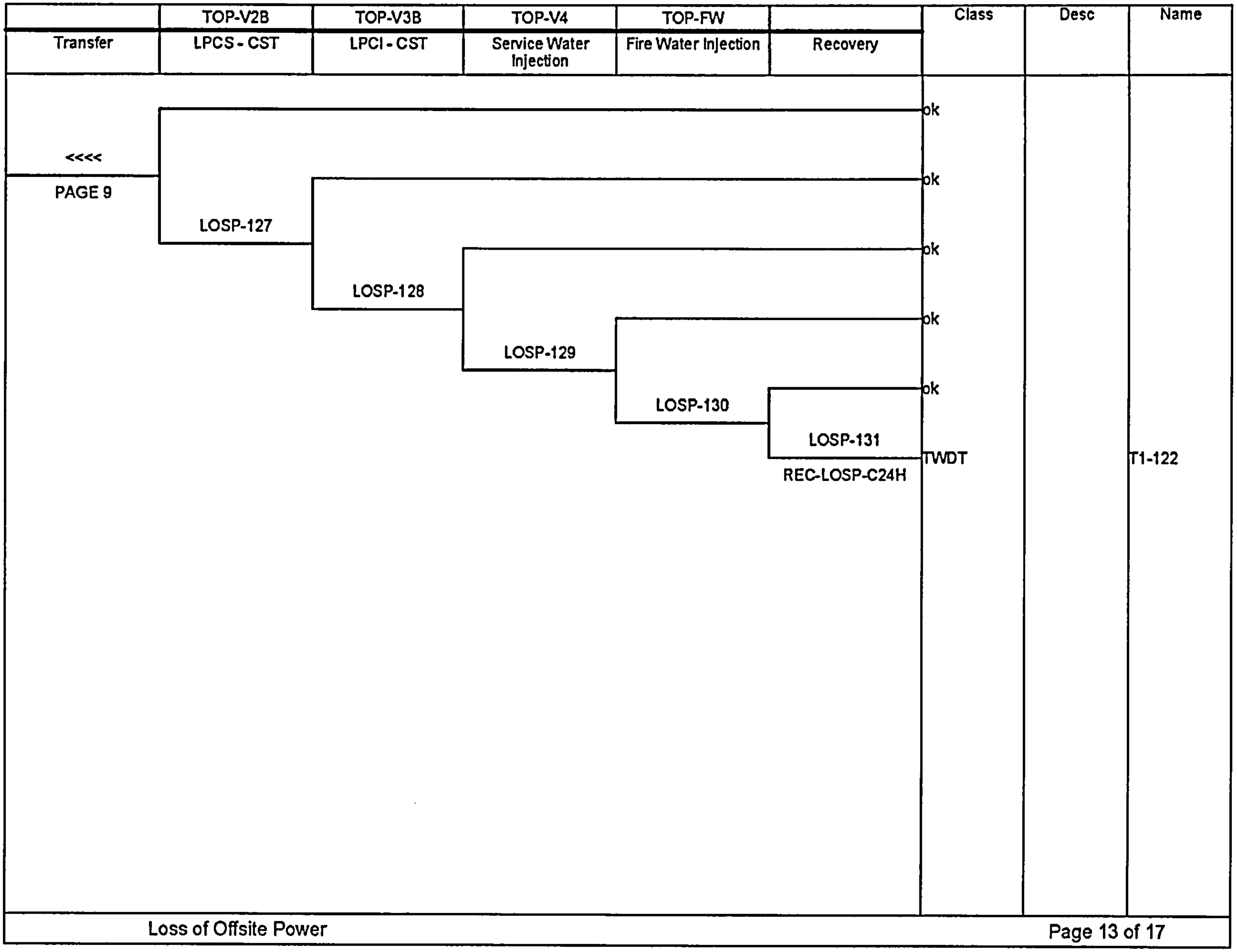
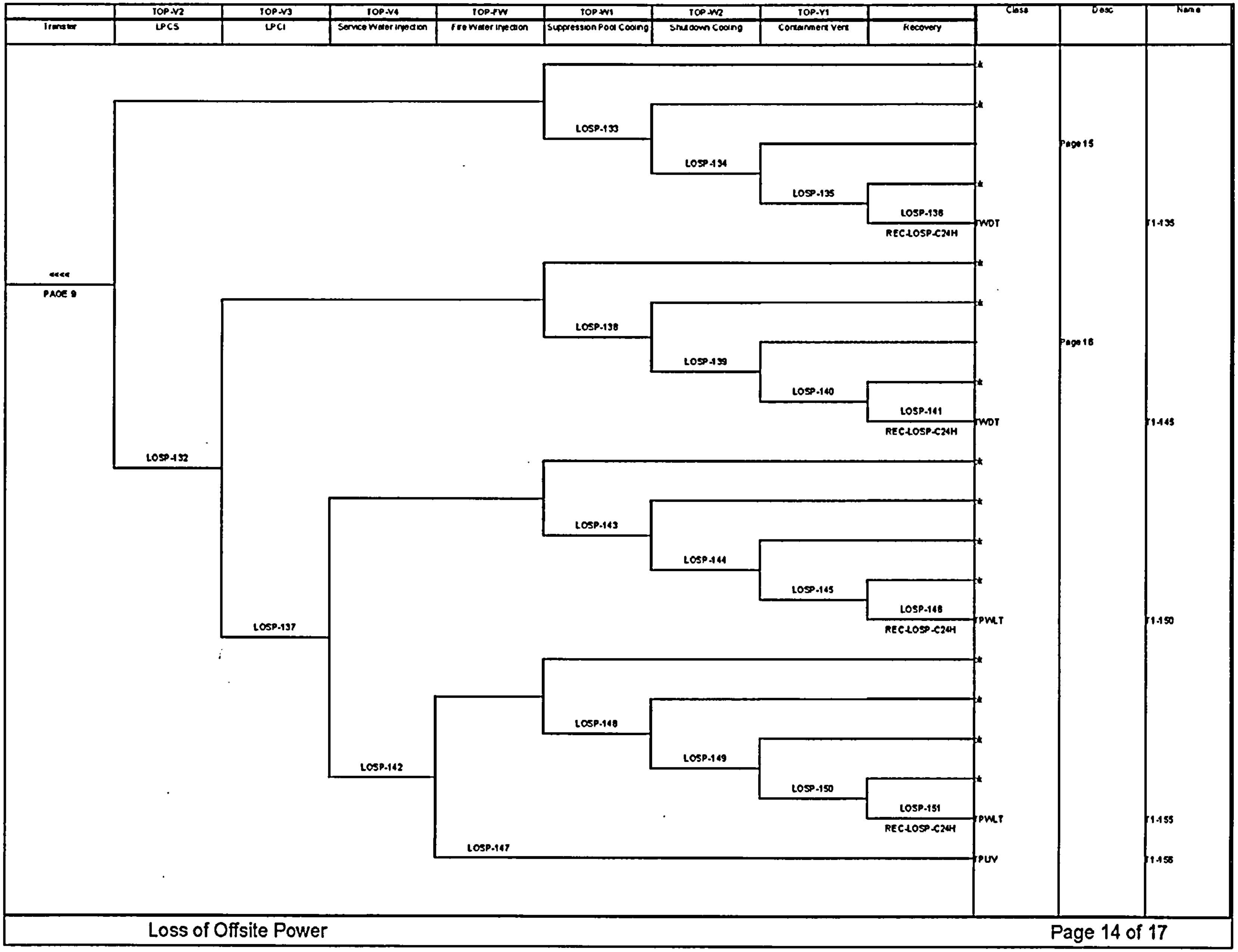

TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
Class
Desc
Name
Transfer
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Recovery
ok
<<<<
ok
PAGE 9
LOSP-112
ok
LOSP-113
ok
LOSP-114
ok
LOSP-115
LOSP-116
TWDT
T1-91
REC-LOSP-C24H
Loss of Offsite Power
Page 10 of 17


TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
Class
Desc
Name
Transfer
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Recovery
<<<<
PAGE 9
LOSP-117
LOSP-118
LOSP-119
LOSP-120
LOSP-121
REC-LOSP-C24H
ok
ok
ok
ok
ok
TWDT
T1-101
Loss of Offsite Power
Page 11 of 17


Transfer
TOP-V2B
LPCS - CST
TOP-V3B
LPCI - CST
TOP-V4
Service Water Injection
TOP-FW
Fire Water Injection
Recovery
Class
Desc
Name
<<<<
PAGE 9
LOSP-122
LOSP-123
LOSP-124
LOSP-125
LOSP-126
REC-LOSP-C24H
ok
ok
ok
ok
ok
TWDT
T1-112
Loss of Offsite Power
Page 12 of 17


TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
Class
Desc
Name
Transfer
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Recovery
ok
<<<<
PAGE 9
ok
LOSP-127
ok
LOSP-128
ok
LOSP-129
ok
LOSP-130
LOSP-131
TWDT
T1-122
REC-LOSP-C24H
Loss of Offsite Power
Page 13 of 17


TOP-V2
TOP-V3
TOP-V4
TOP-FW
TOP-W1
TOP-W2
TOP-Y1
Class
Desc
Name
Transfer
LPCS
LPCI
Service Water Injection
Fire Water Injection
Suppression Pool Cooling
Shutdown Cooling
Containment Vent
Recovery
<<<<
PAGE 9
LOSP-132
LOSP-133
LOSP-134
LOSP-135
LOSP-136
REC-LOSP-C24H
TWDT
Page 15
T1-135
LOSP-137
LOSP-138
LOSP-139
LOSP-140
LOSP-141
REC-LOSP-C24H
TWDT
Page 16
T1-145
LOSP-142
LOSP-143
LOSP-144
LOSP-145
LOSP-146
REC-LOSP-C24H
TPWLT
T1-150
LOSP-147
LOSP-148
LOSP-149
LOSP-150
LOSP-151
REC-LOSP-C24H
TPWLT
T1-155
TPUV
T1-156
Loss of Offsite Power
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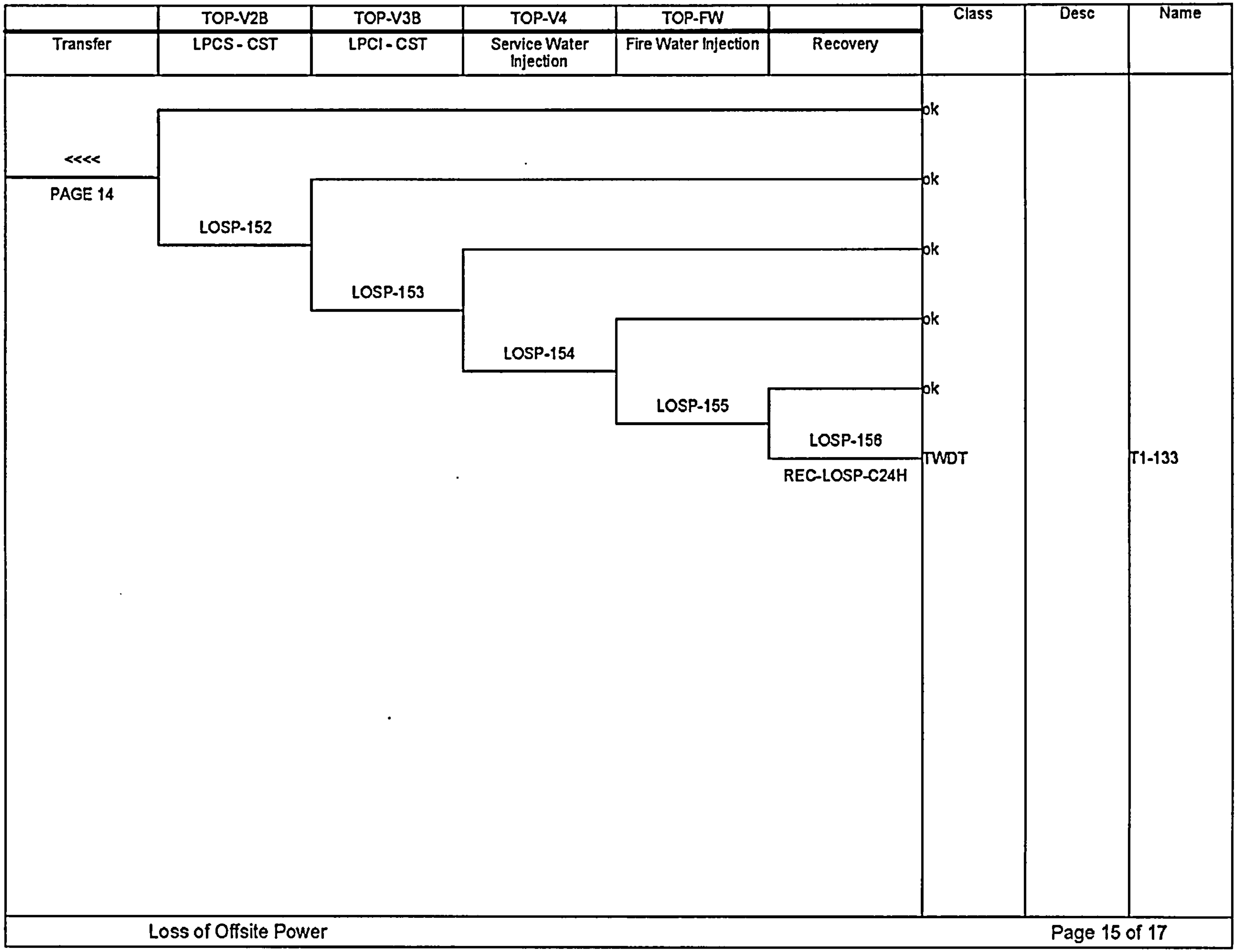
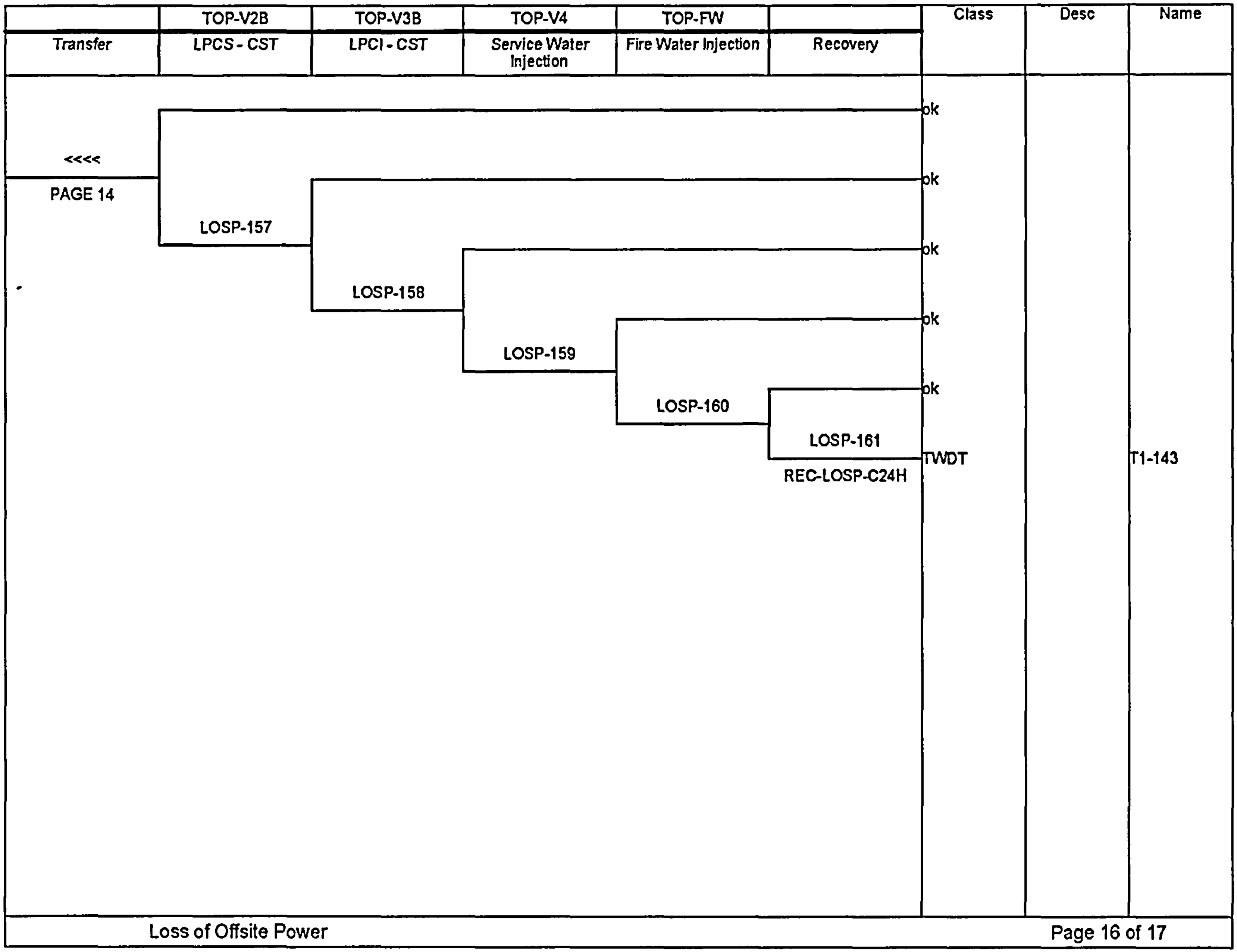
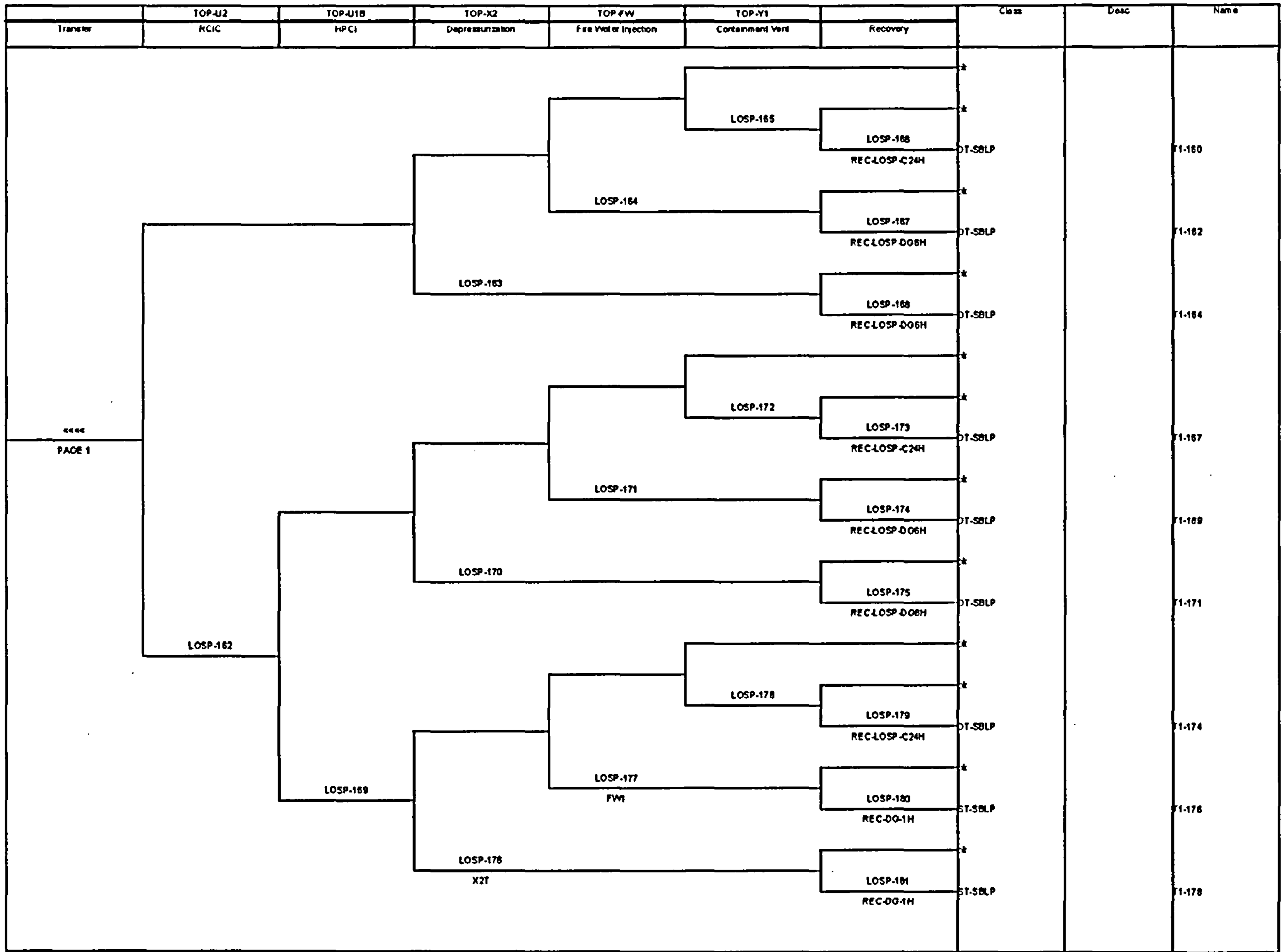

TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
Class
Desc
Name
Transfer
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Recovery
<<<<
PAGE 14
LOSP-152
LOSP-153
LOSP-154
LOSP-155
LOSP-156
REC-LOSP-C24H
ok
ok
ok
ok
ok
TWDT
T1-133
Loss of Offsite Power
Page 15 of 17


TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
Class
Desc
Name
Transfer
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Recovery
<<<<
PAGE 14
LOSP-157
LOSP-158
LOSP-159
LOSP-160
LOSP-161
REC-LOSP-C24H
ok
ok
ok
ok
ok
TWDT
T1-143
Loss of Offsite Power
Page 16 of 17



Loss of Offsite Power

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Figure C.3-3 Loss of Service Water Event Tree

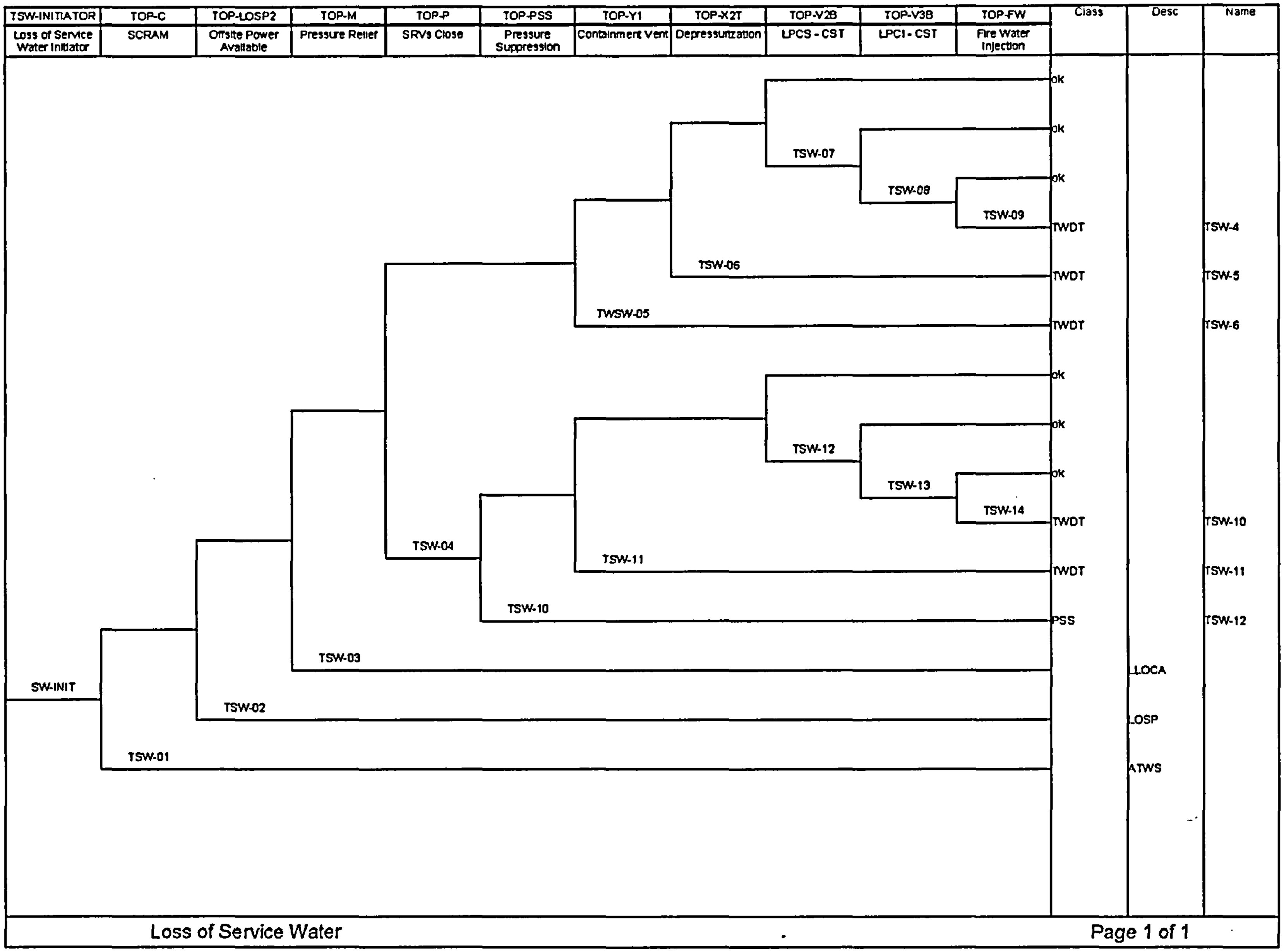
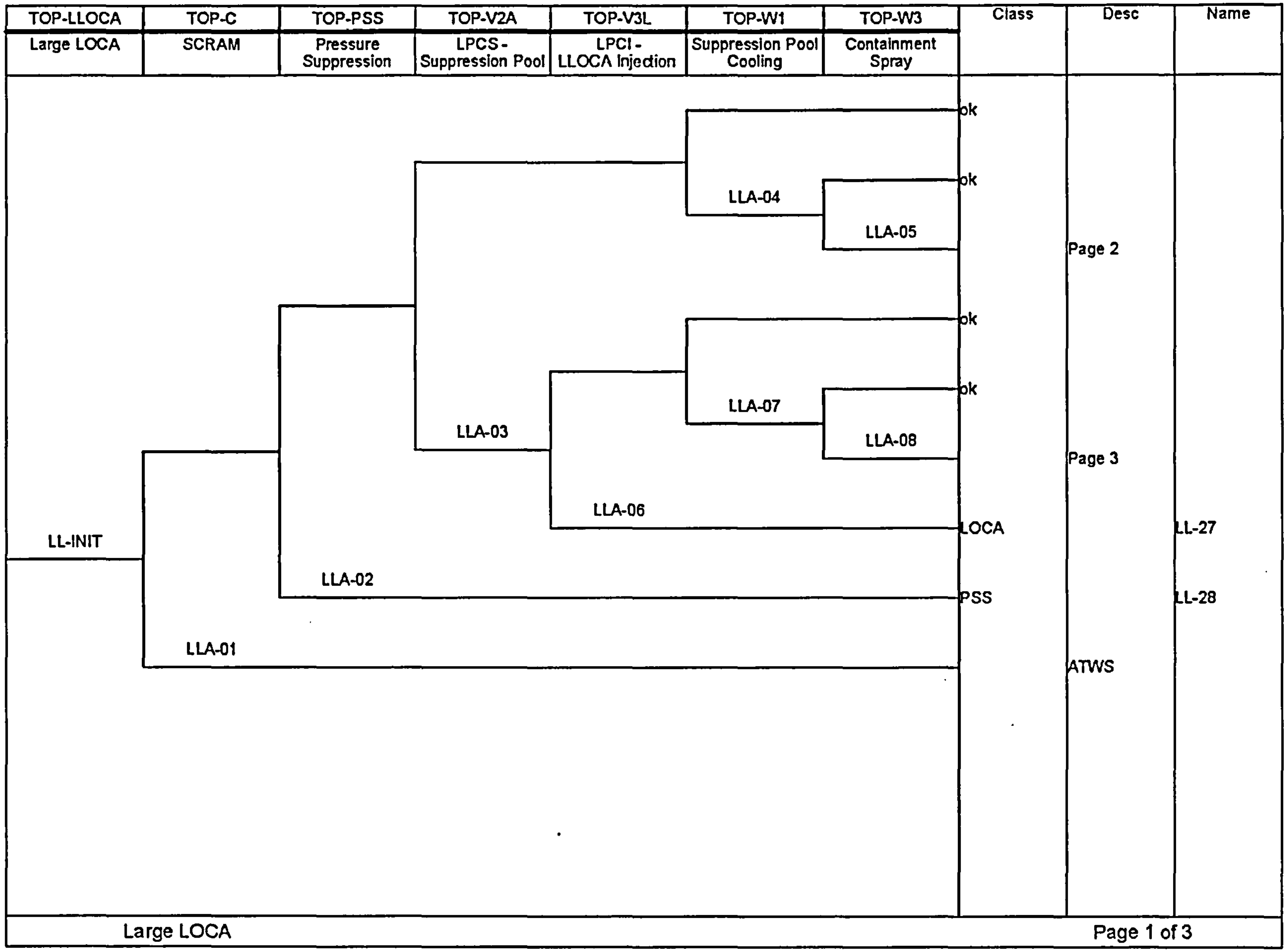
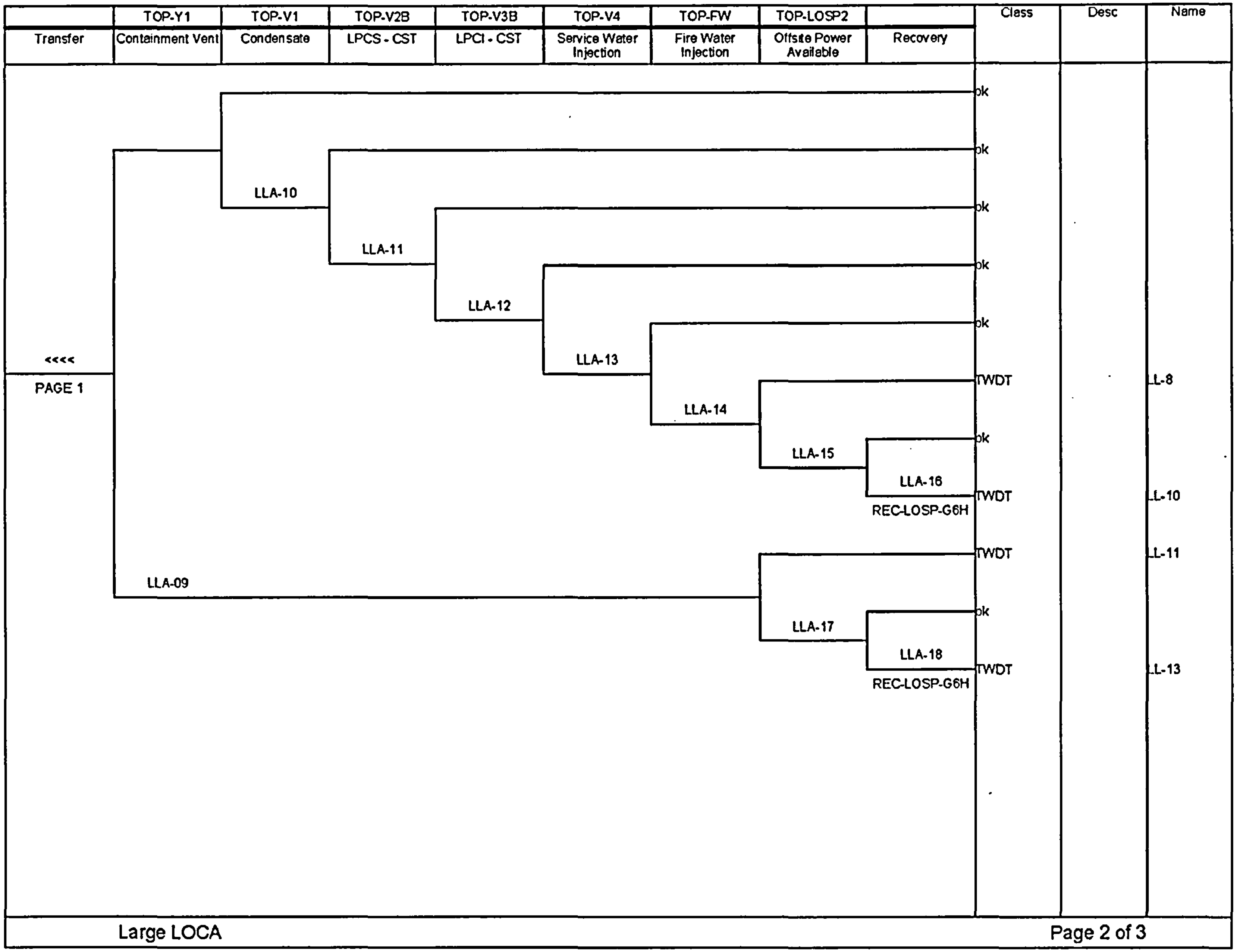


Figure C.3-4 Large LOCA Event Tree




TOP-Y1
TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-LOSP2
Class
Desc
Name
Transfer
Containment Vent
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Offsite Power Available
Recovery
<<<<
PAGE 1
LLA-09
LLA-10
LLA-11
LLA-12
LLA-13
LLA-14
LLA-15
LLA-16
REC-LOSP-G6H
LLA-17
LLA-18
REC-LOSP-G6H
ok
ok
ok
ok
ok
TWDT
LL-8
ok
TWDT
LL-10
TWDT
LL-11
ok
TWDT
LL-13
Large LOCA
Page 2 of 3

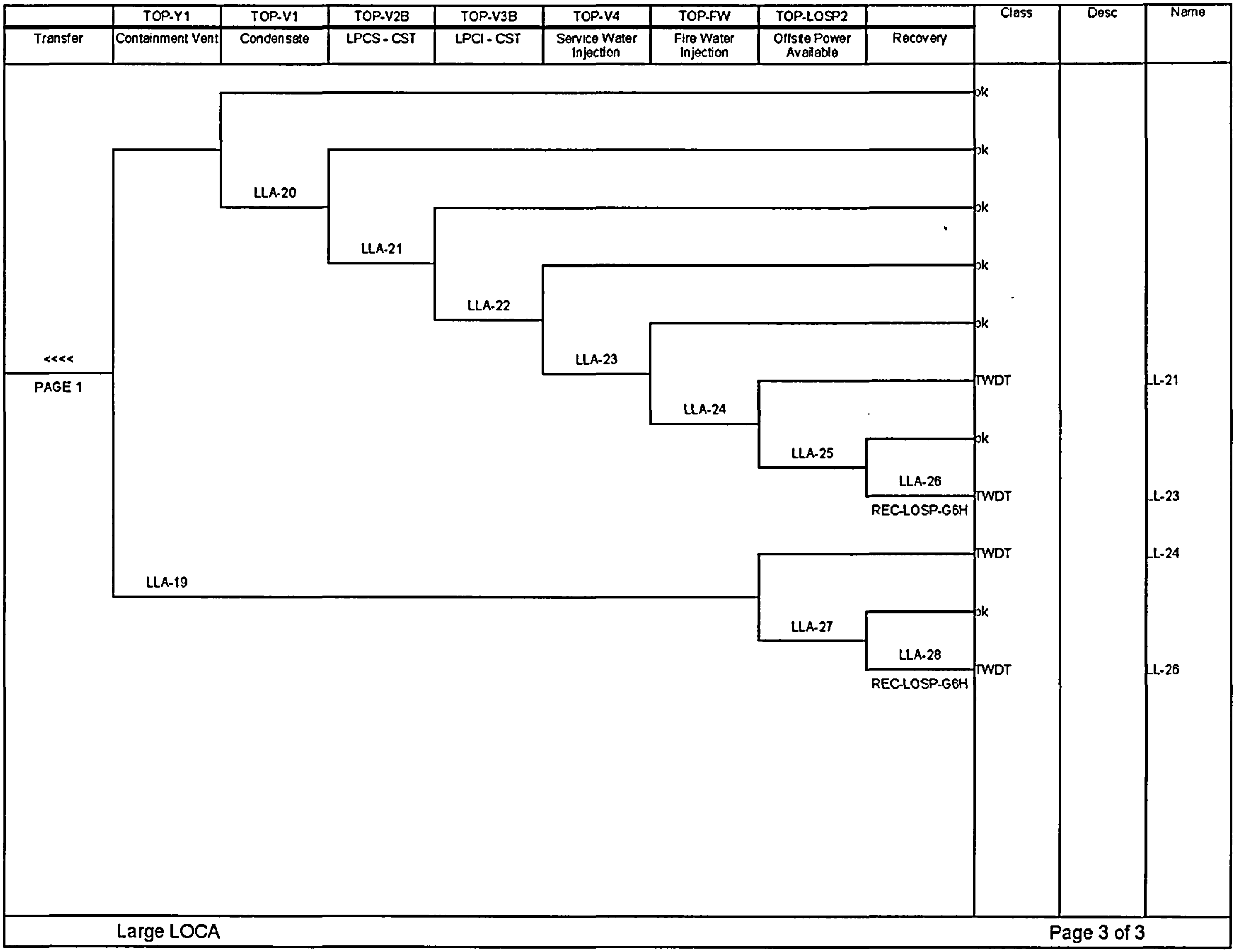
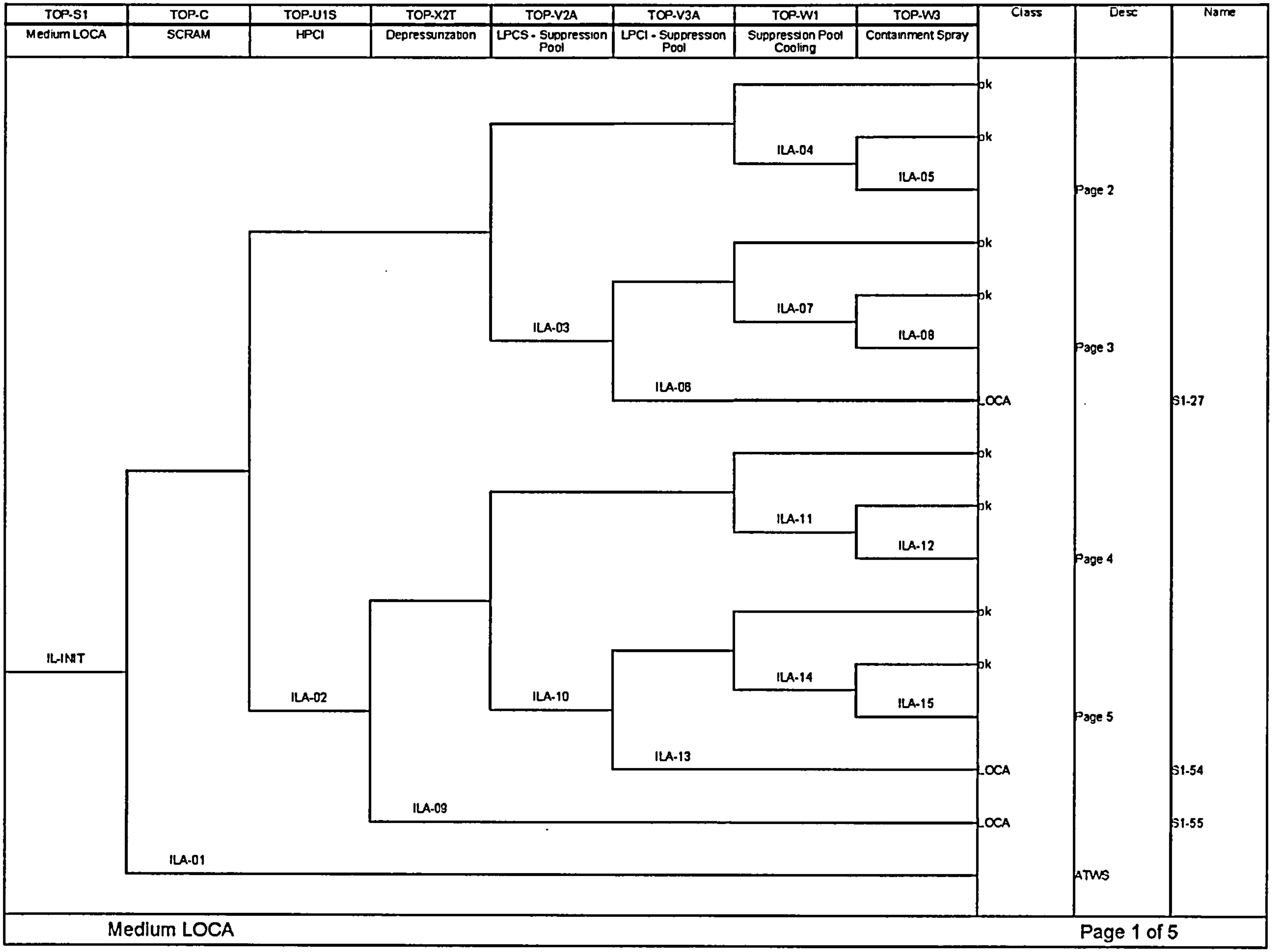
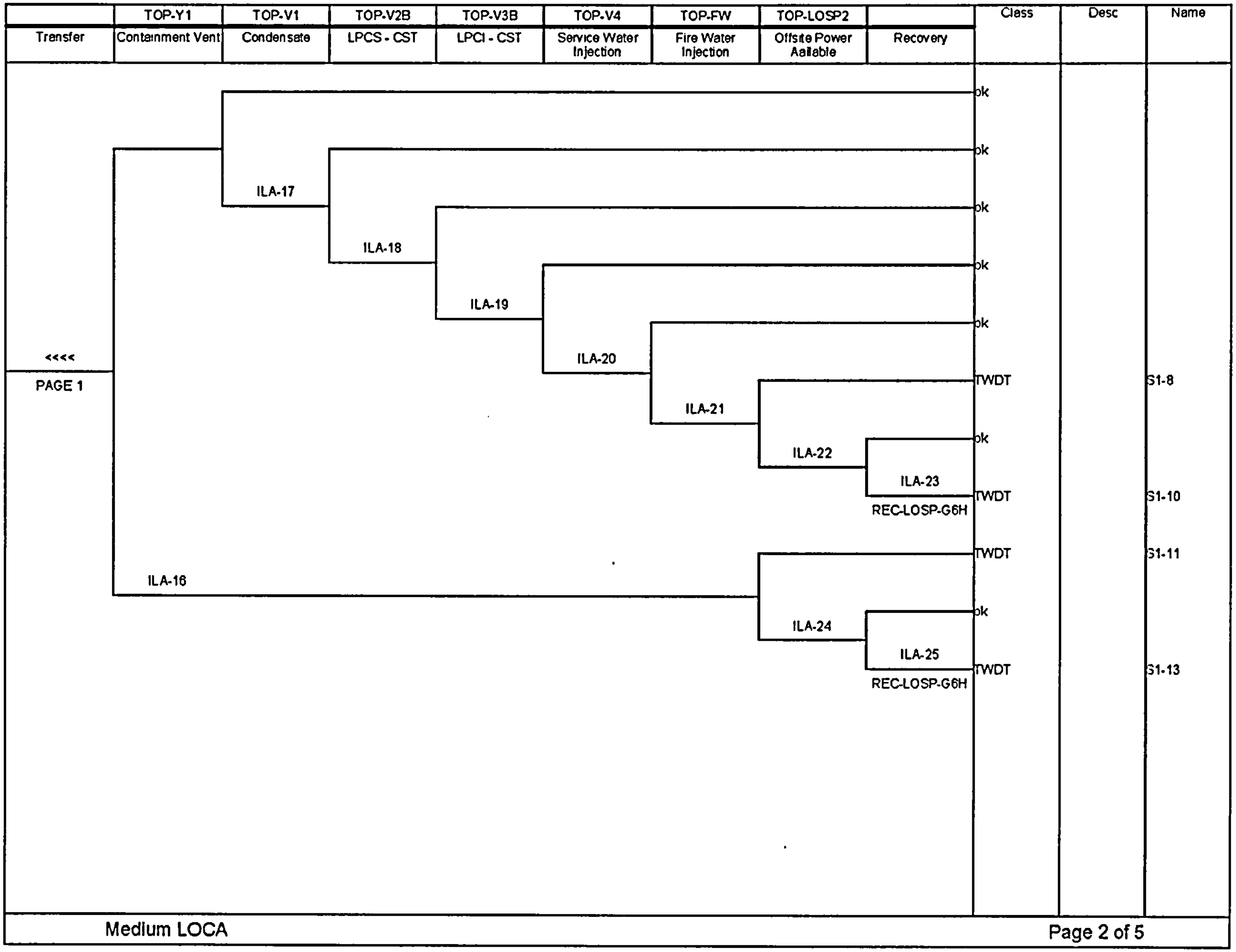
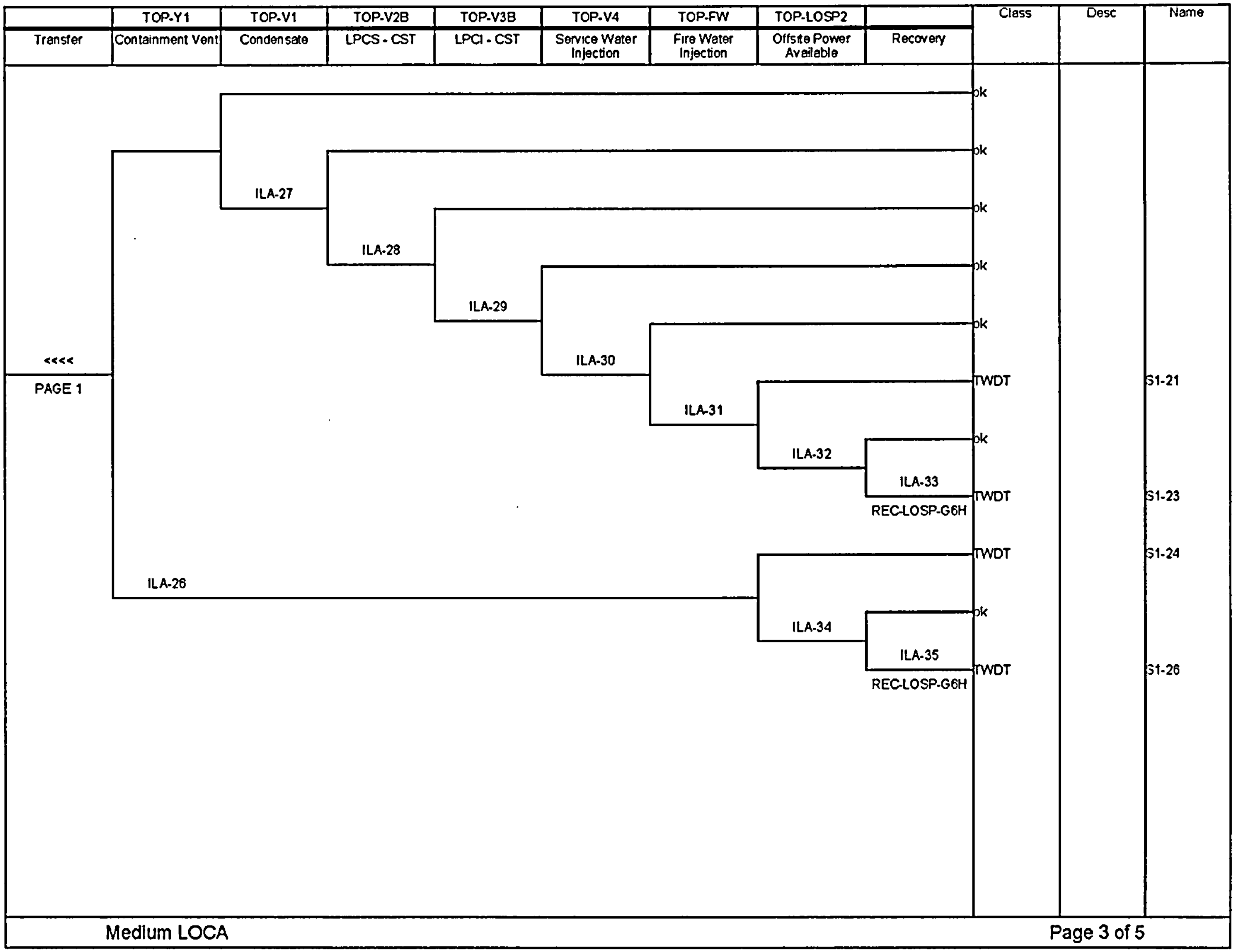
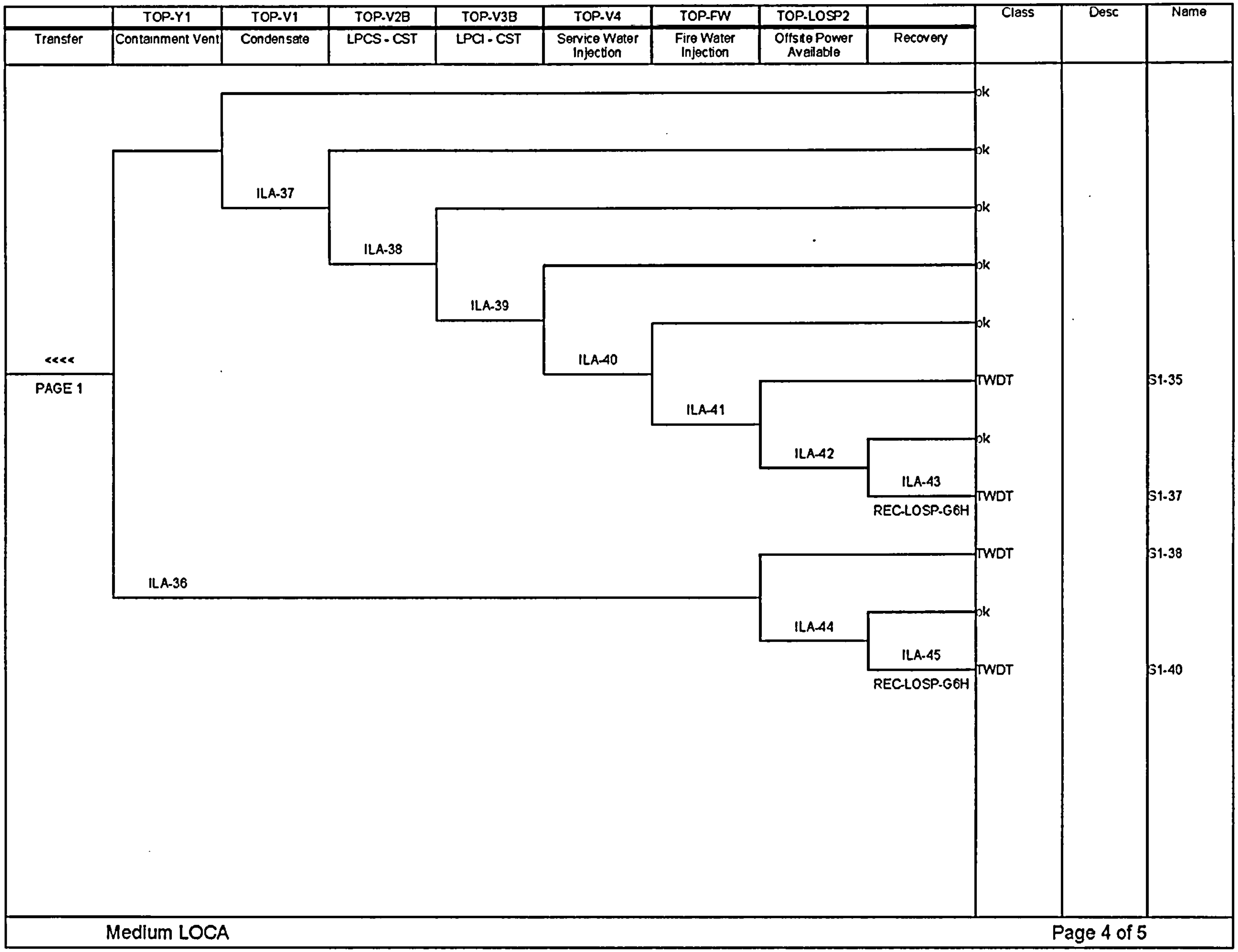

TOP-Y1
TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-LOSP2
Class
Desc
Name
Transfer
Containment Vent
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Offsite Power Available
Recovery
<<<<
PAGE 1
LLA-19
LLA-20
LLA-21
LLA-22
LLA-23
LLA-24
LLA-25
LLA-26
LLA-27
LLA-28
REC-LOSP-G6H
REC-LOSP-G6H
ok
ok
ok
ok
ok
TWDT
ok
TWDT
TWDT
ok
TWDT
LL-21
LL-23
LL-24
LL-26
Large LOCA
Page 3 of 3

Figure C.3-5 Medium LOCA Event Tree




TOP-Y1
TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-LOSP2
Class
Desc
Name
Transfer
Containment Vent
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Offsite Power Aailable
Recovery
<<<<
PAGE 1
ILA-16
ILA-17
ILA-18
ILA-19
ILA-20
ILA-21
ILA-22
ILA-23
REC-LOSP-G6H
ILA-24
ILA-25
REC-LOSP-G6H
ok
ok
ok
ok
ok
TWDT
ok
TWDT
TWDT
ok
TWDT
S1-8
S1-10
S1-11
S1-13
Medium LOCA
Page 2 of 5


TOP-Y1
TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-LOSP2
Class
Desc
Name
Transfer
Containment Vent
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Offsite Power Available
Recovery
ok
ok
ok
ok
ok
<<<<
PAGE 1
ILA-27
ILA-28
ILA-29
ILA-30
ILA-31
TWDT
S1-21
ILA-32
ok
ILA-33
TWDT
REC-LOSP-G6H
S1-23
TWDT
S1-24
ILA-26
ok
ILA-34
ILA-35
TWDT
REC-LOSP-G6H
S1-26
Medium LOCA
Page 3 of 5


TOP-Y1
TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-LOSP2
Class
Desc
Name
Transfer
Containment Vent
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Offsite Power Available
Recovery
ok
ok
ILA-37
ok
ILA-38
ok
ILA-39
ok
<<<<
PAGE 1
ILA-40
TWDT
S1-35
ILA-41
ok
ILA-42
ILA-43
TWDT
S1-37
REC-LOSP-G6H
TWDT
S1-38
ILA-36
ok
ILA-44
ILA-45
TWDT
S1-40
REC-LOSP-G6H
Medium LOCA
Page 4 of 5

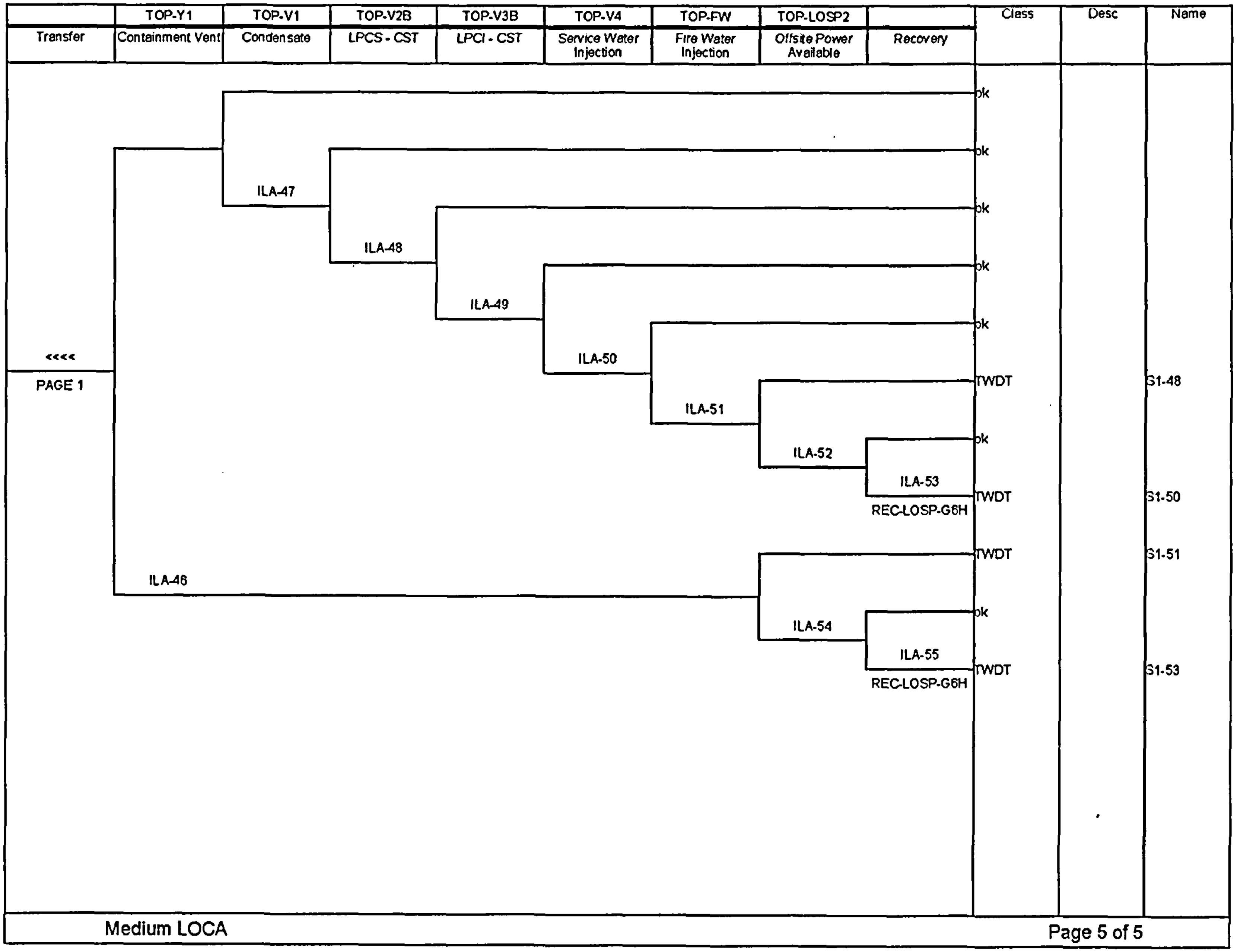
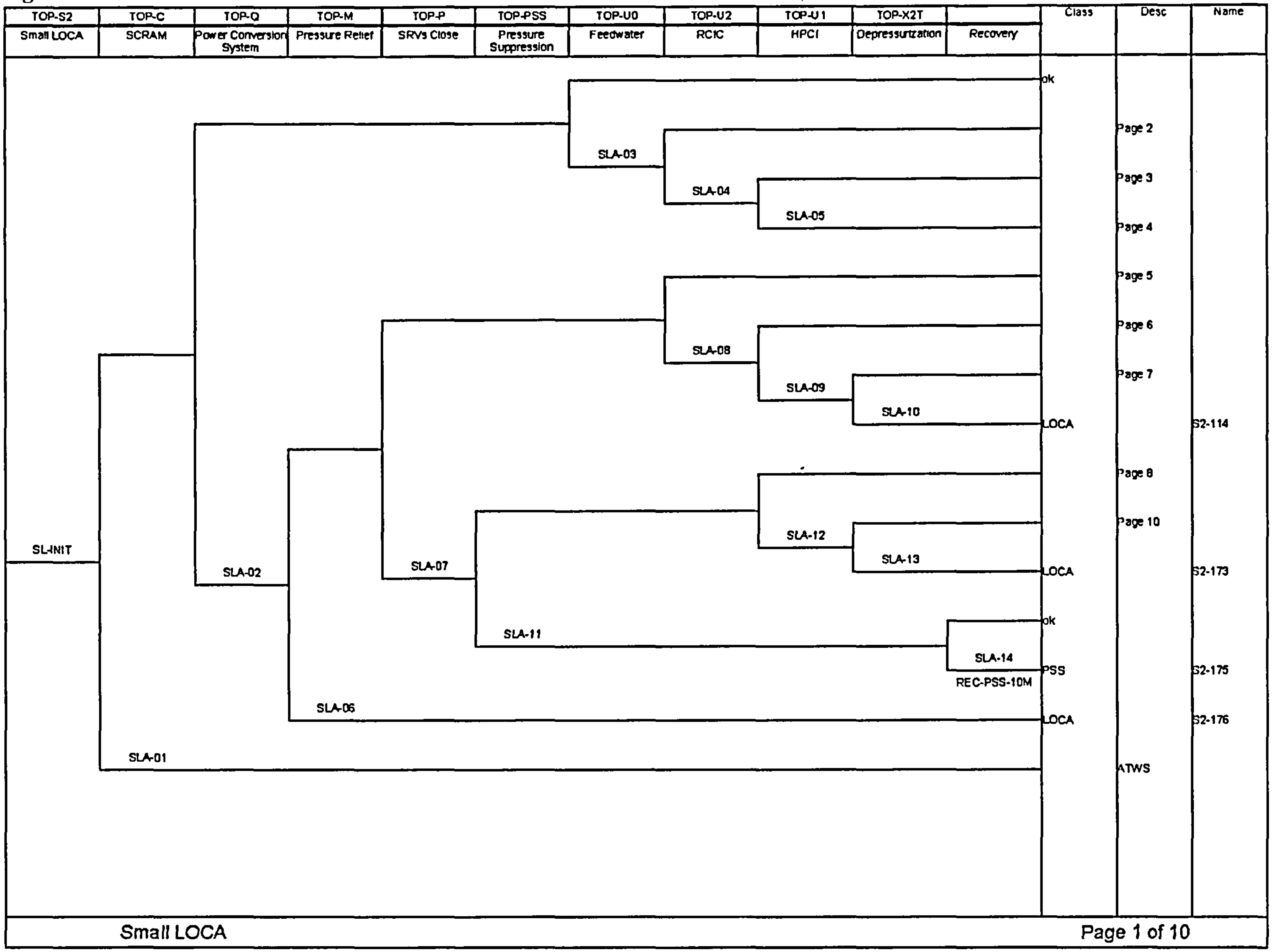
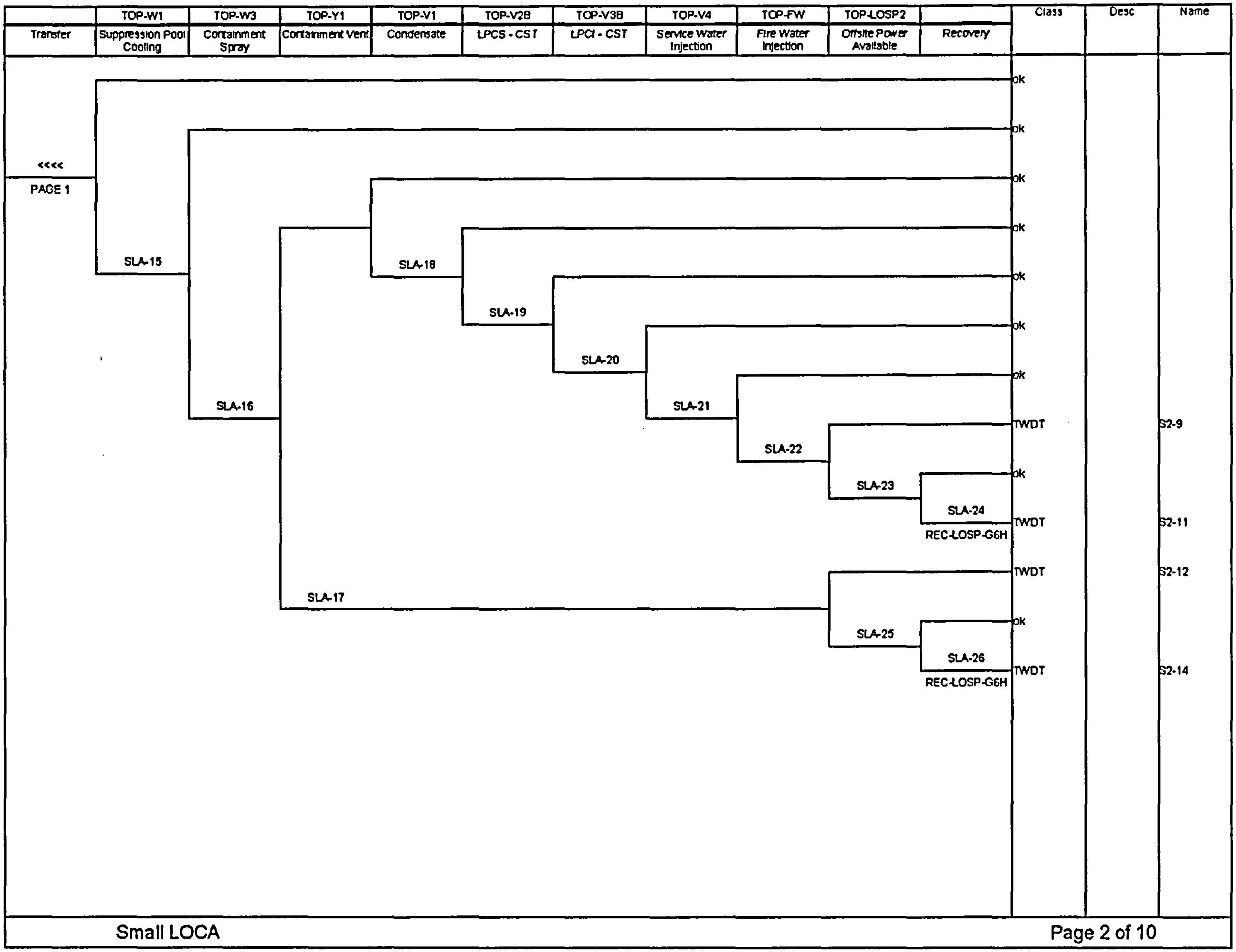
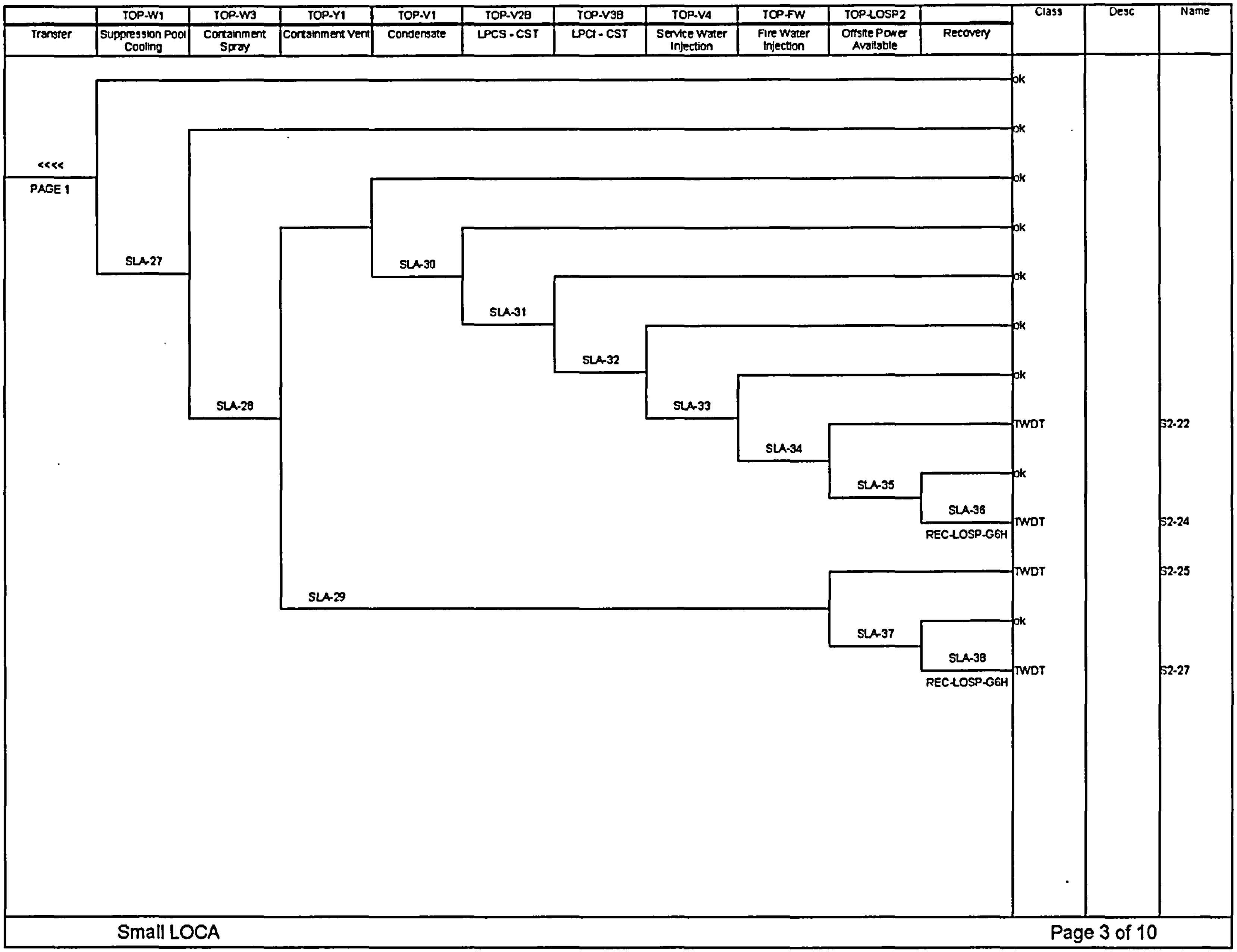

TOP-Y1
TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-LOSP2
Class
Desc
Name
Transfer
Containment Vent
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Offsite Power Available
Recovery
<<<<
PAGE 1
ILA-47
ILA-48
ILA-49
ILA-50
ILA-51
ILA-52
ILA-53
REC-LOSP-G6H
ILA-46
ILA-54
ILA-55
REC-LOSP-G6H
ok
ok
ok
ok
ok
TWDT
ok
TWDT
TWDT
ok
TWDT
S1-48
S1-50
S1-51
S1-53
Medium LOCA
Page 5 of 5

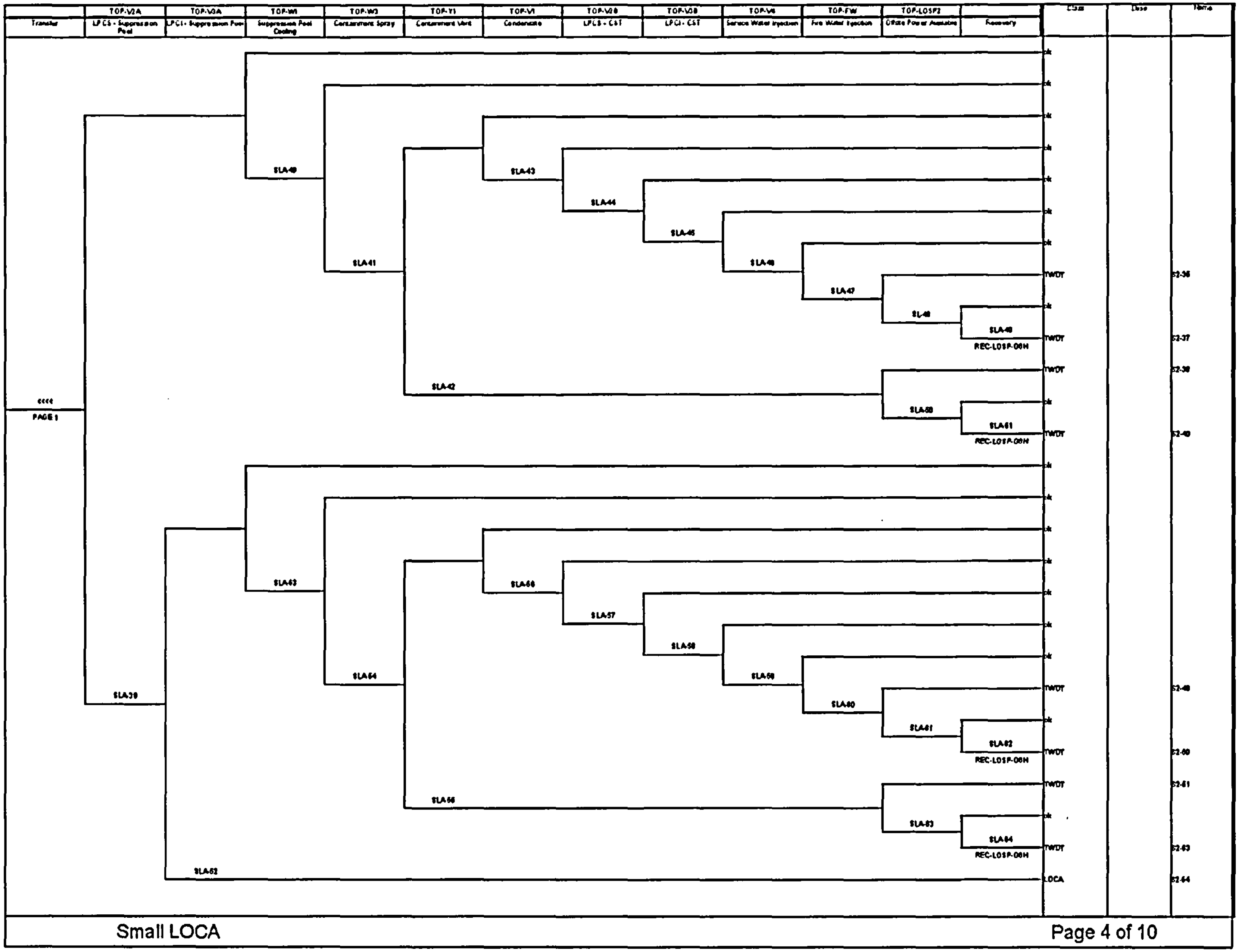
Figure C.3-6 Small LOCA



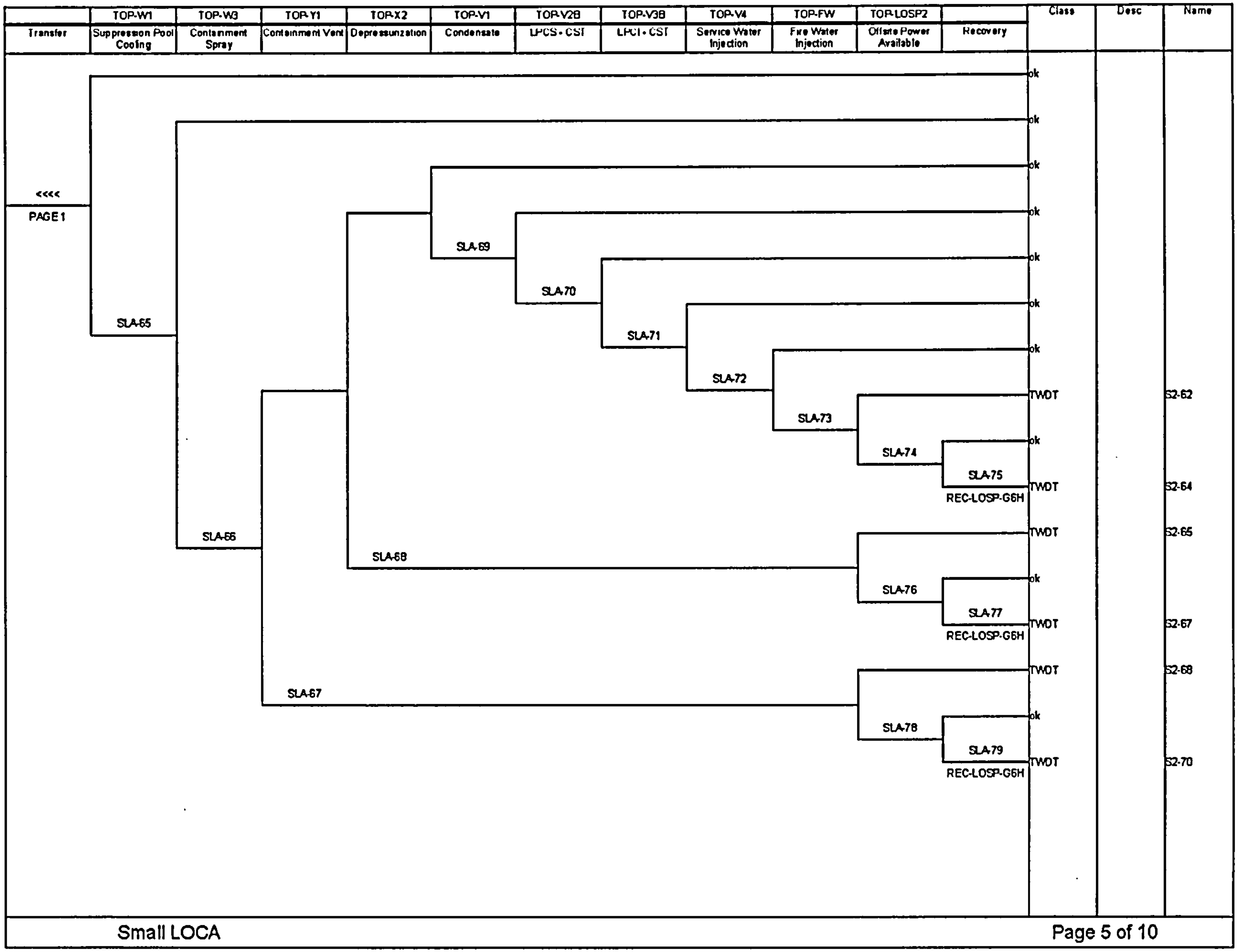


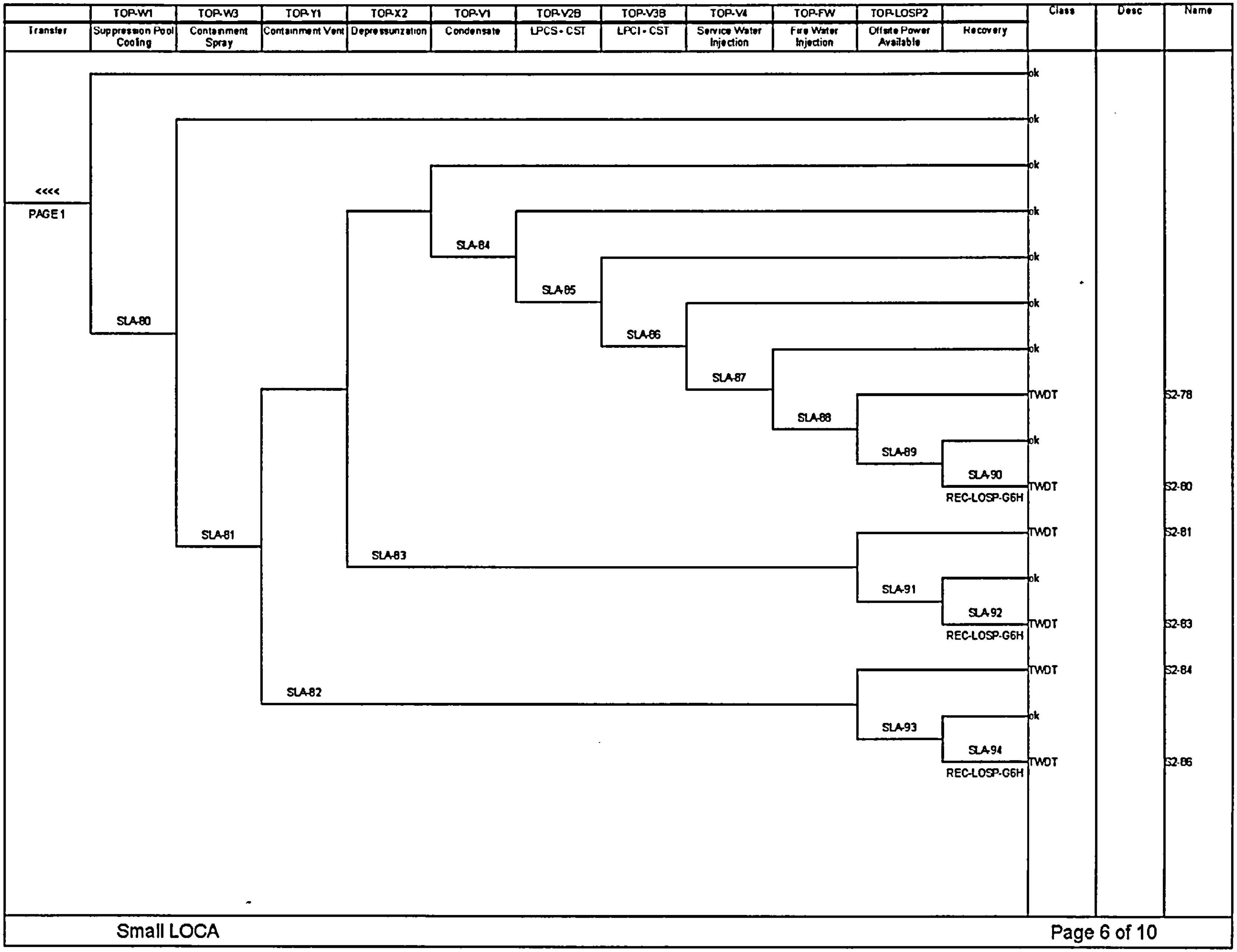
TOP-W1
TOP-W3
TOP-Y1
TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-LOSP2
Class
Desc
Name
Transfer
Suppression Pool Cooling
Containment Spray
Containment Vent
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Offsite Power Available
Recovery
<<<<
PAGE 1
SLA-15
SLA-16
SLA-17
SLA-18
SLA-19
SLA-20
SLA-21
SLA-22
SLA-23
SLA-24
REC-LOSP-G6H
SLA-25
SLA-26
REC-LOSP-G6H
ok
ok
ok
ok
ok
ok
ok
TWDT S2-9
ok
TWDT S2-11
TWDT S2-12
ok
TWDT S2-14
Small LOCA
Page 2 of 10

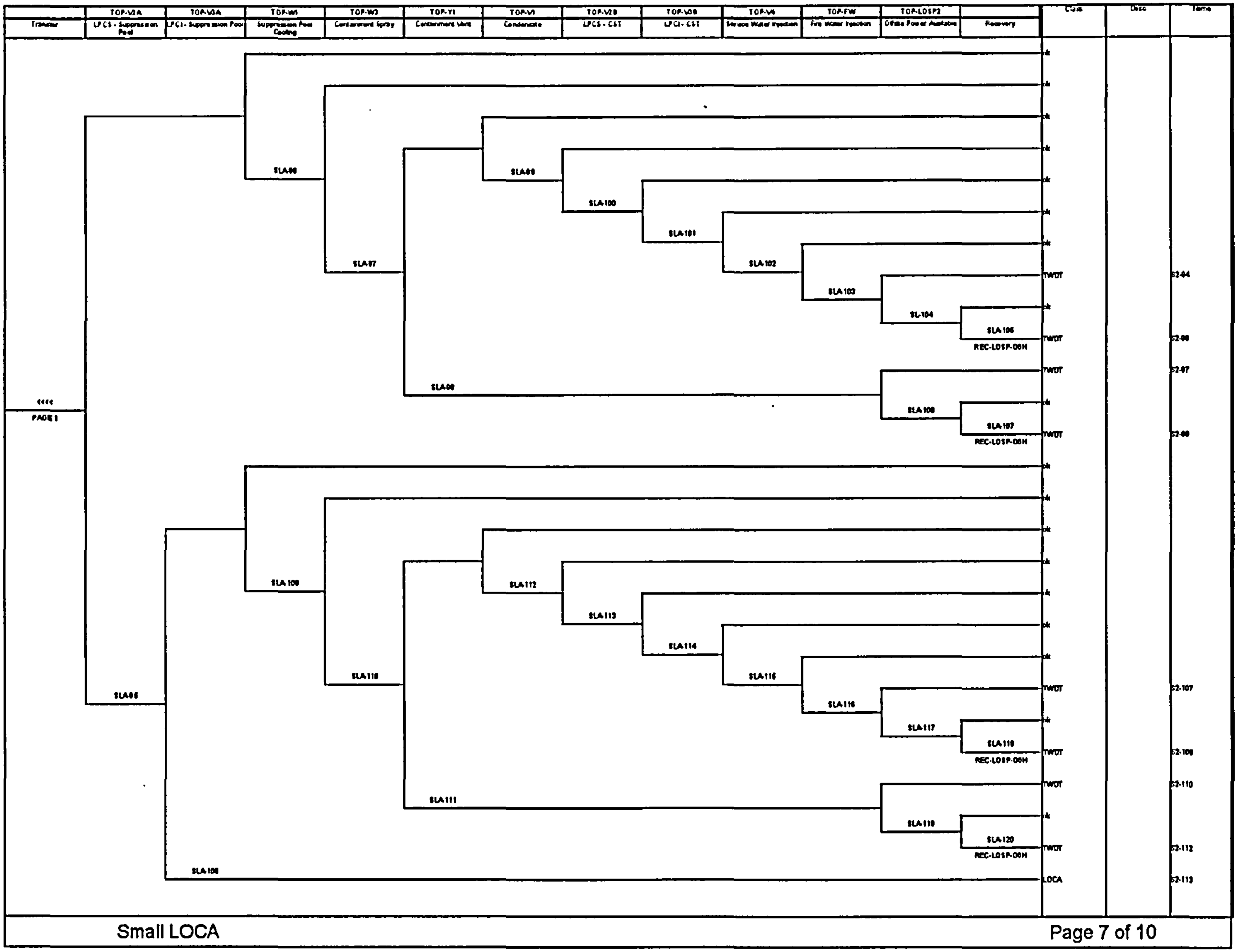

TOP-W1
TOP-W3
TOP-Y1
TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-LOSP2
Class
Desc
Name
Transfer
Suppression Pool Cooling
Containment Spray
Containment Vent
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Offsite Power Available
Recovery
<<<<
PAGE 1
SLA-27
SLA-28
SLA-29
SLA-30
SLA-31
SLA-32
SLA-33
SLA-34
SLA-35
SLA-36
REC-LOSP-G6H
SLA-37
SLA-38
REC-LOSP-G6H
ok
ok
ok
ok
ok
ok
ok
TWDT S2-22
ok
TWDT S2-24
TWDT S2-25
ok
TWDT S2-27
Small LOCA
Page 3 of 10

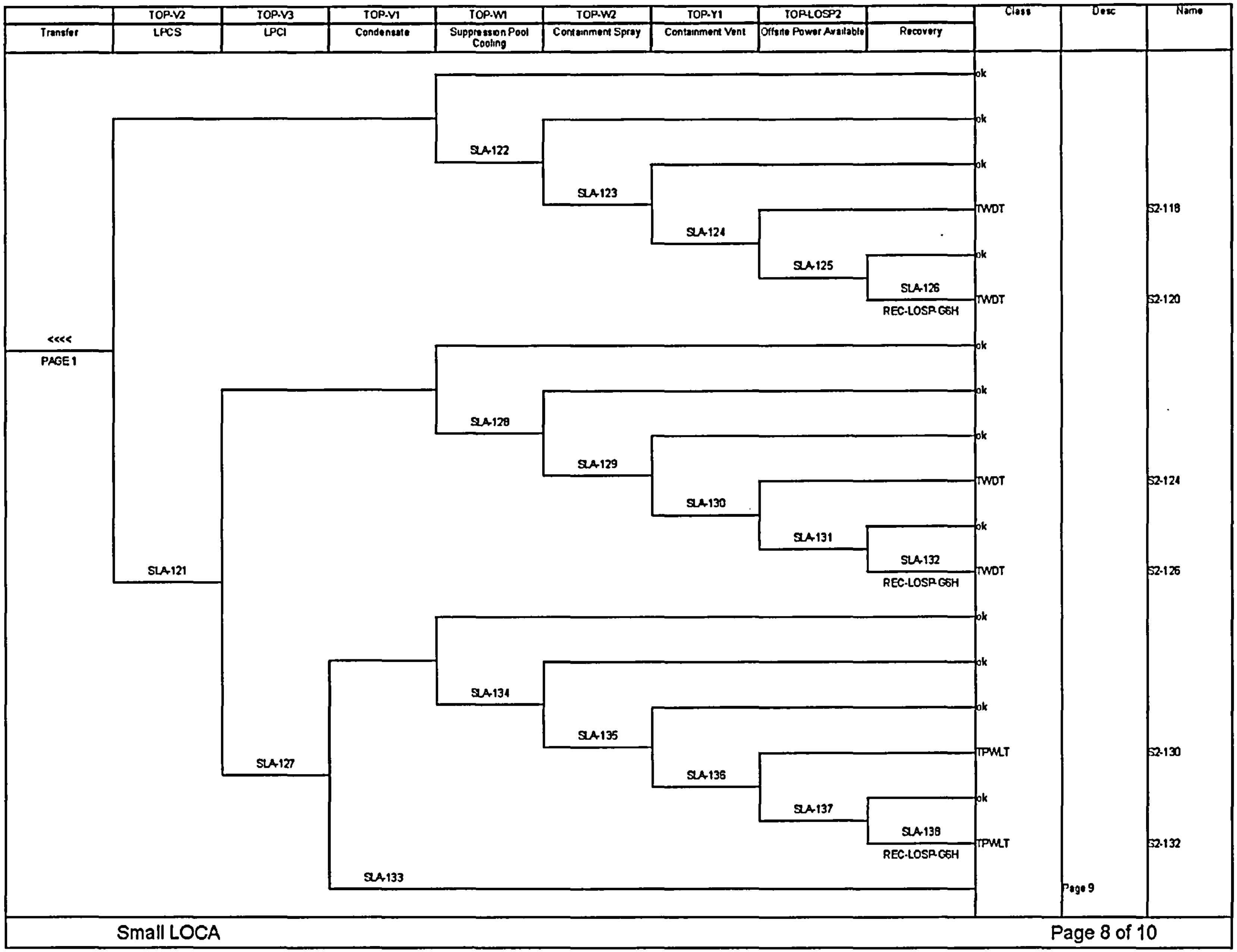


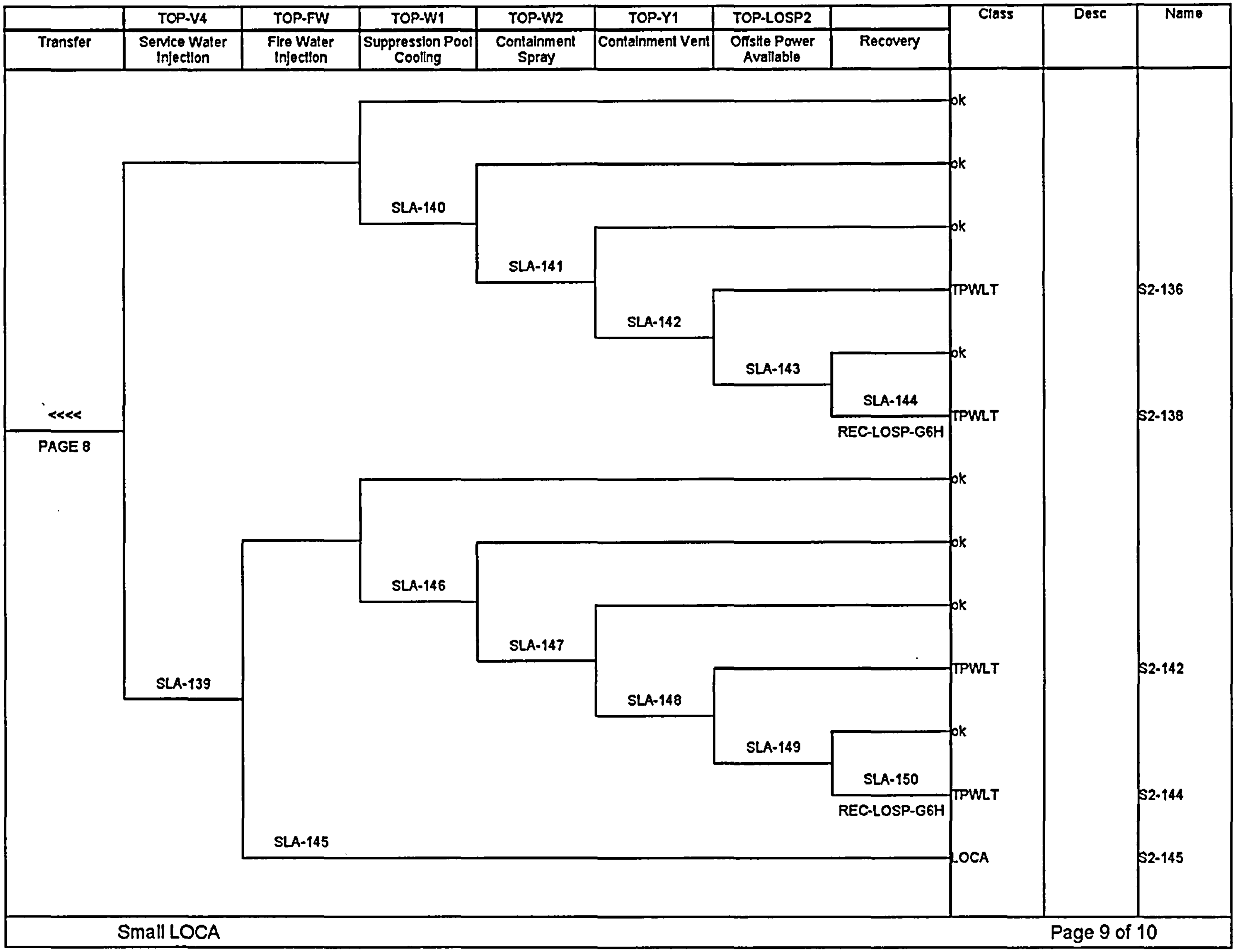
Transfer
LPCS - Suppression Pool
Suppression Pool Cooling
Containment Spray
Containment Vent
Condensate
Service Water Injection
Fire Water Injection
Recovery
TWDT
LOCA
Small LOCA
Page 4 of 10


Transfer
TOP-W1 Suppression Pool Cooling
TOP-W3 Containment Spray
TOP-Y1 Containment Vent
TOP-X2 Depressurization
TOP-V1 Condensate
TOP-V2B LPCS - CST
TOP-V3B LPCI - CST
TOP-V4 Service Water Injection
TOP-FW Fire Water Injection
TOP-LOSP2 Offsite Power Available
Recovery
Class
Desc
Name
<<<<
PAGE 1
SLA-65
SLA-66
SLA-67
SLA-68
SLA-69
SLA-70
SLA-71
SLA-72
SLA-73
SLA-74
SLA-75
REC-LOSP-G6H
SLA-76
SLA-77
REC-LOSP-G6H
SLA-78
SLA-79
REC-LOSP-G6H
ok
ok
ok
ok
ok
ok
ok
TWDT S2-62
ok
TWDT S2-64
TWDT S2-65
ok
TWDT S2-67
TWDT S2-68
ok
TWDT S2-70
Small LOCA
Page 5 of 10


TOP-W1
TOP-W3
TOP-Y1
TOP-X2
TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-LOSP2
Class
Desc
Name
Transfer
Suppression Pool Cooling
Containment Spray
Containment Vent
Depressurization
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Offsite Power Available
Recovery
<<<<
PAGE 1
SLA-80
SLA-81
SLA-82
SLA-83
SLA-84
SLA-85
SLA-86
SLA-87
SLA-88
SLA-89
SLA-90
REC-LOSP-G6H
SLA-91
SLA-92
REC-LOSP-G6H
SLA-93
SLA-94
REC-LOSP-G6H
ok
ok
ok
ok
ok
ok
ok
TWDT S2-78
ok
TWDT S2-80
TWDT S2-81
ok
TWDT S2-83
TWDT S2-84
ok
TWDT S2-86
Small LOCA
Page 6 of 10


Small LOCA
Page 7 of 10


TOP-V2
TOP-V3
TOP-V1
TOP-W1
TOP-W2
TOP-Y1
TOP-LOSP2
Class
Desc
Name
Transfer
LPCS
LPCI
Condensate
Suppression Pool Cooling
Containment Spray
Containment Vent
Offsite Power Available
Recovery
<<<<
PAGE 1
SLA-121
SLA-122
SLA-123
SLA-124
SLA-125
SLA-126
REC-LOSP-G6H
SLA-127
SLA-128
SLA-129
SLA-130
SLA-131
SLA-132
REC-LOSP-G6H
SLA-133
SLA-134
SLA-135
SLA-136
SLA-137
SLA-138
REC-LOSP-G6H
ok
ok
ok
TWDT
S2-118
ok
TWDT
S2-120
ok
ok
ok
TWDT
S2-124
ok
TWDT
S2-126
ok
ok
ok
TPWLT
S2-130
ok
TPWLT
S2-132
Page 9
Small LOCA
Page 8 of 10


TOP-V4
TOP-FW
TOP-W1
TOP-W2
TOP-Y1
TOP-LOSP2
Class
Desc
Name
Transfer
Service Water Injection
Fire Water Injection
Suppression Pool Cooling
Containment Spray
Containment Vent
Offsite Power Available
Recovery
<<<<
PAGE 8
SLA-139
SLA-140
SLA-141
SLA-142
SLA-143
SLA-144
REC-LOSP-G6H
SLA-145
SLA-146
SLA-147
SLA-148
SLA-149
SLA-150
REC-LOSP-G6H
ok
ok
ok
TPWLT S2-136
ok
TPWLT S2-138
ok
ok
ok
TPWLT S2-142
ok
TPWLT S2-144
LOCA S2-145
Small LOCA
Page 9 of 10

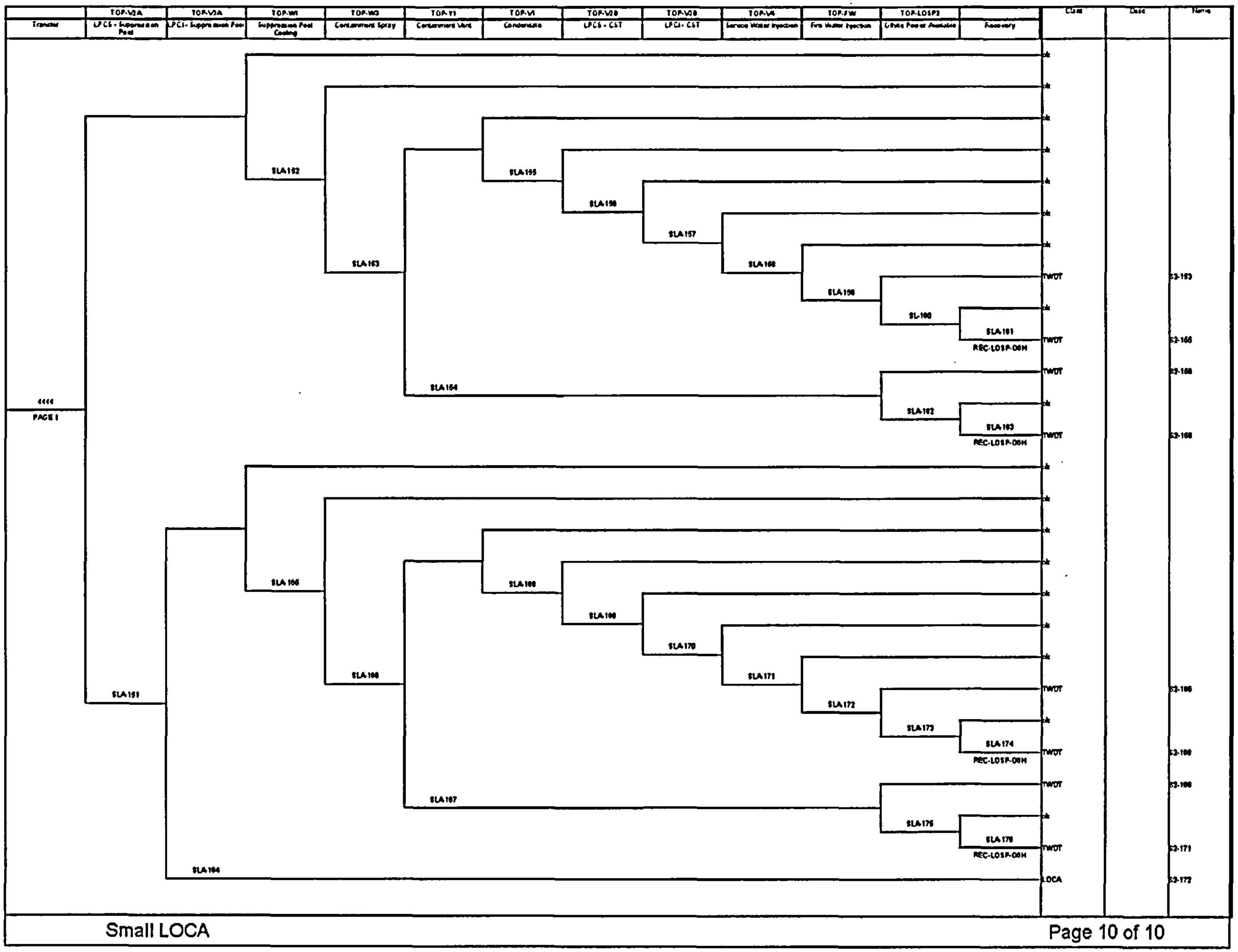
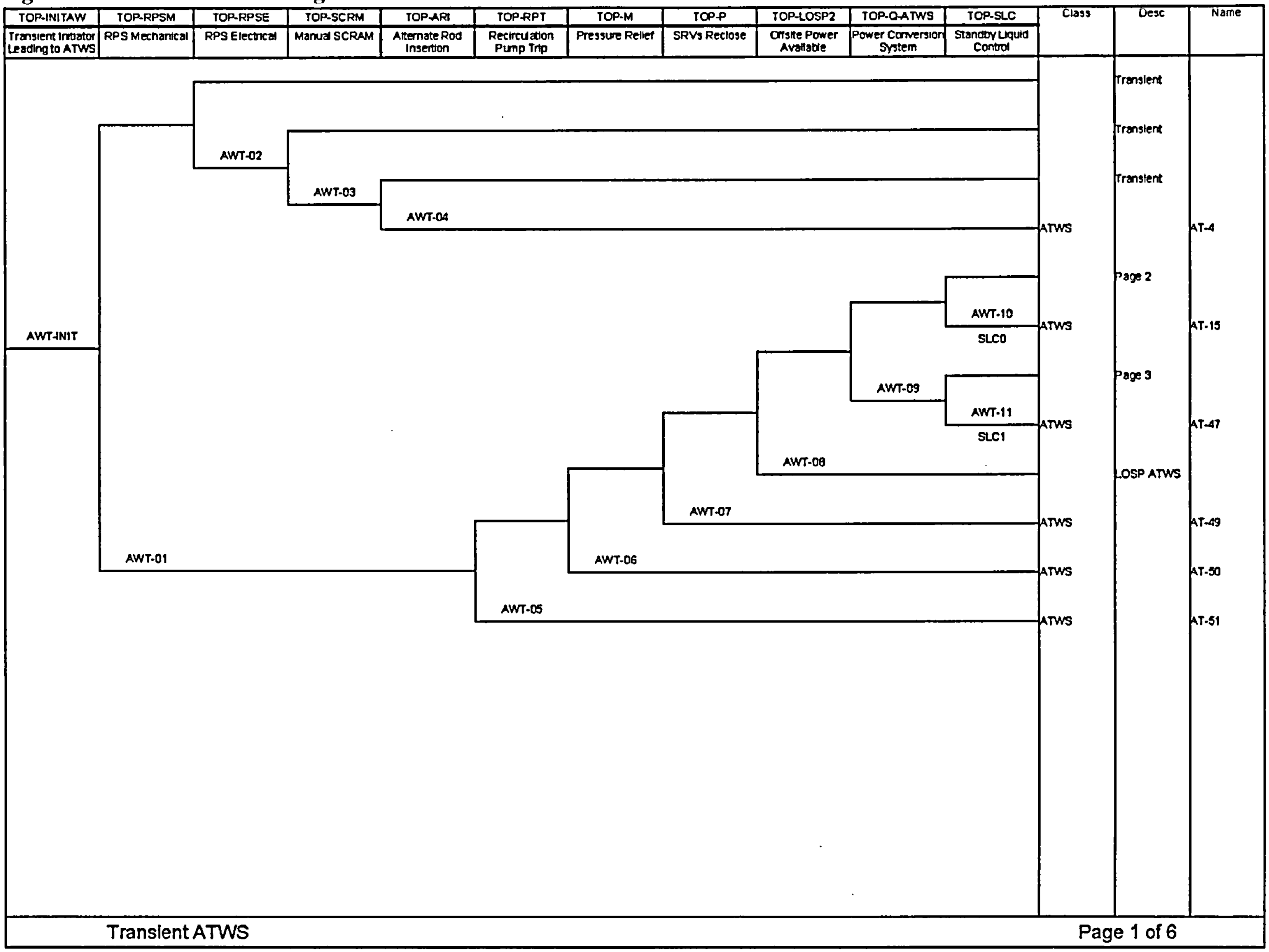
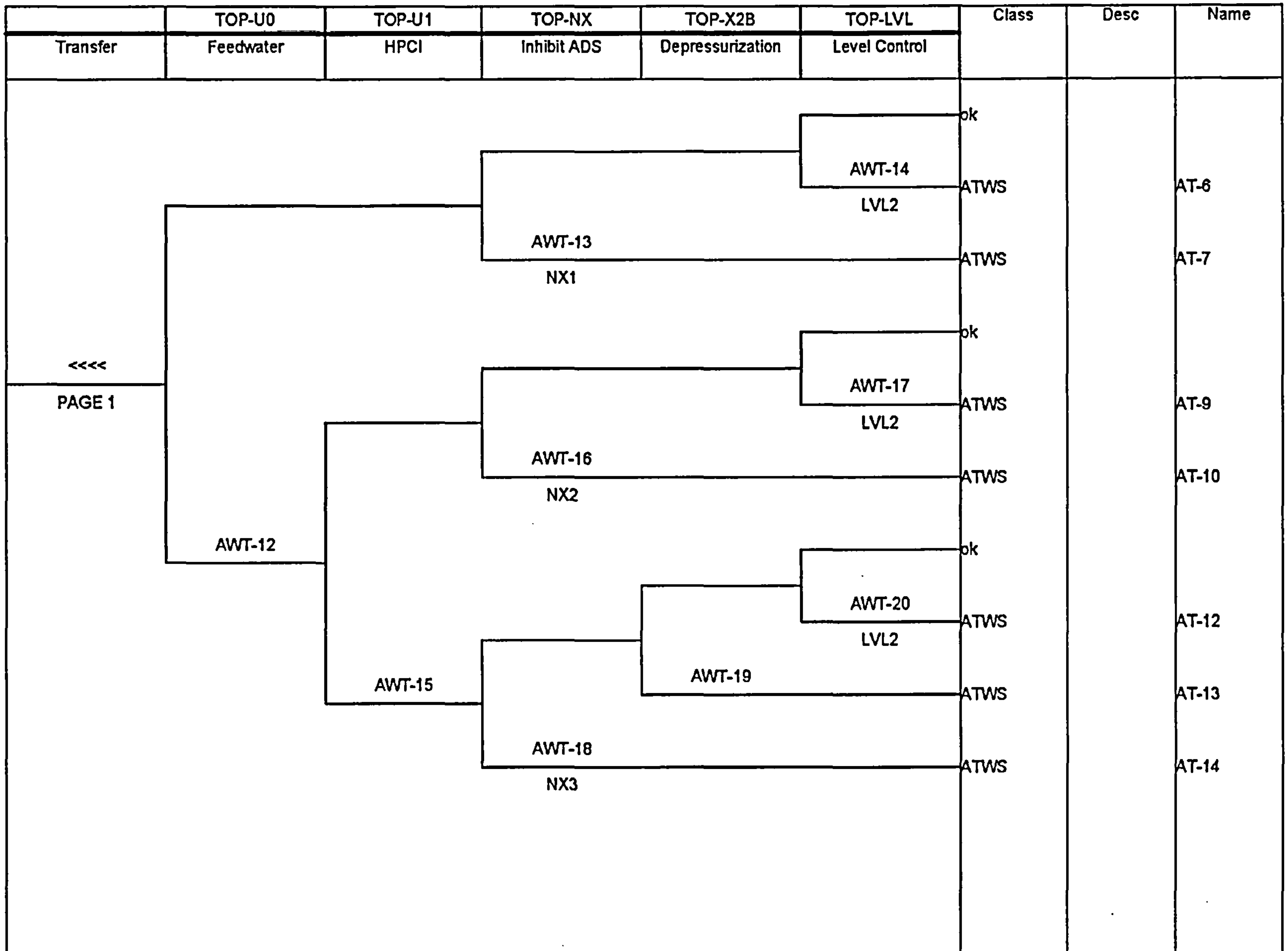

TOP-V2A
TOP-V3A
TOP-W1
TOP-W3
TOP-Y1
TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
TOP-LOSP2
Transfer
LPCS - Suppression Pool
LPCI - Suppression Pool
Suppression Pool Cooling
Containment Spray
Containment Vent
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
Offsite Power Available
Recovery
Class
Desc
Name
PAGE 1
REC-LOSP-06H
TWDT
LOCA
Small LOCA
Page 10 of 10

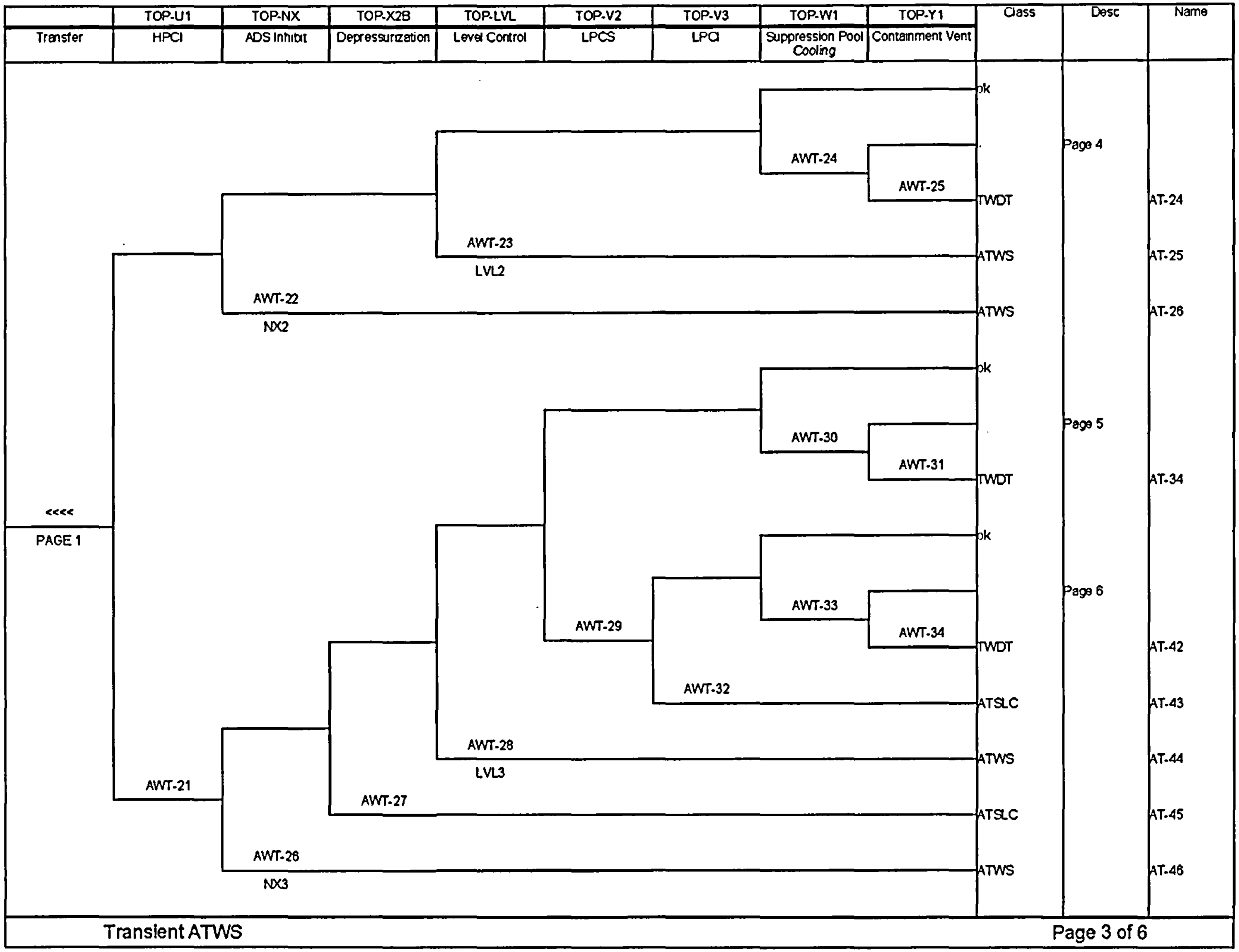
Figure C.3-7 ATWS Following a Transient Event Tree

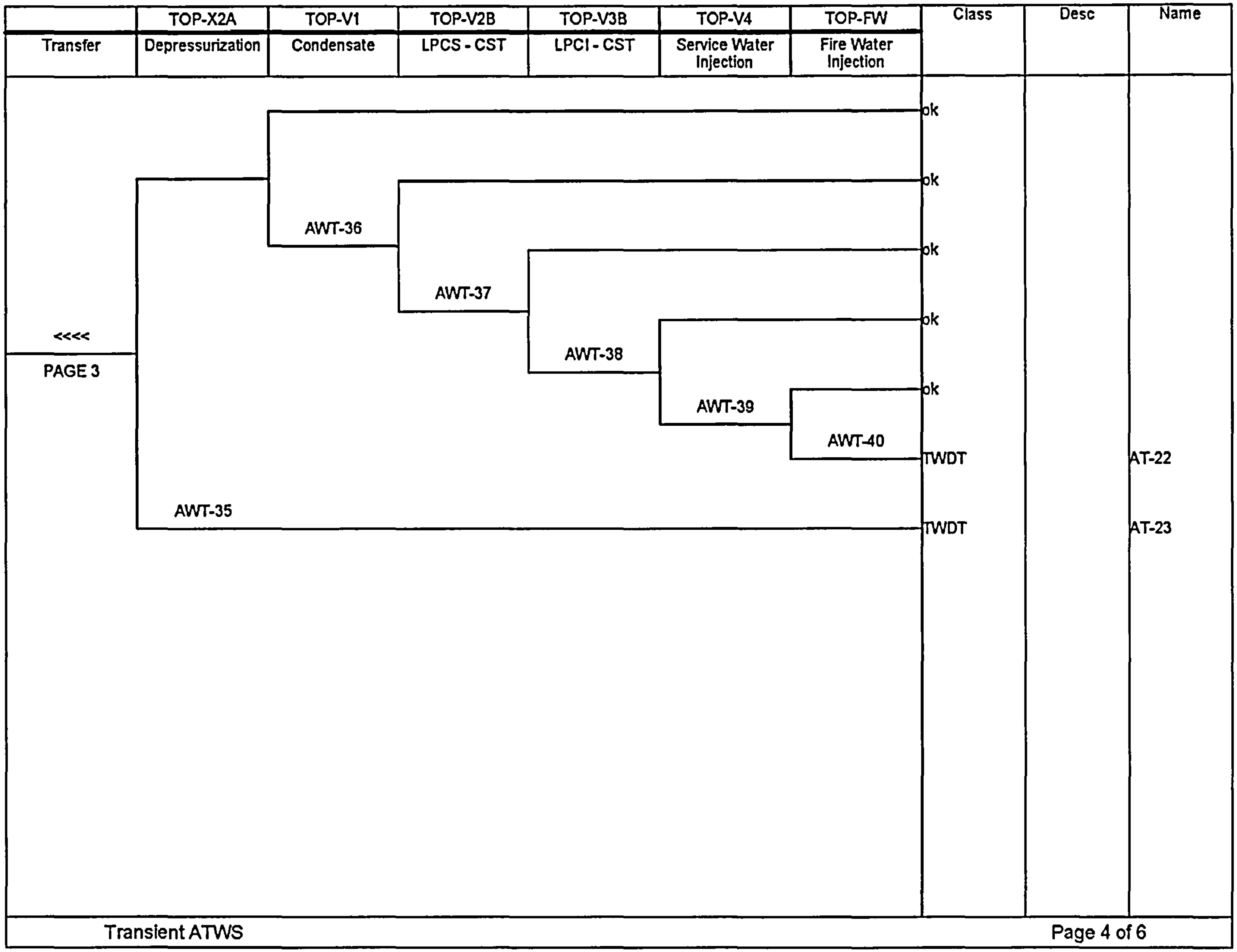


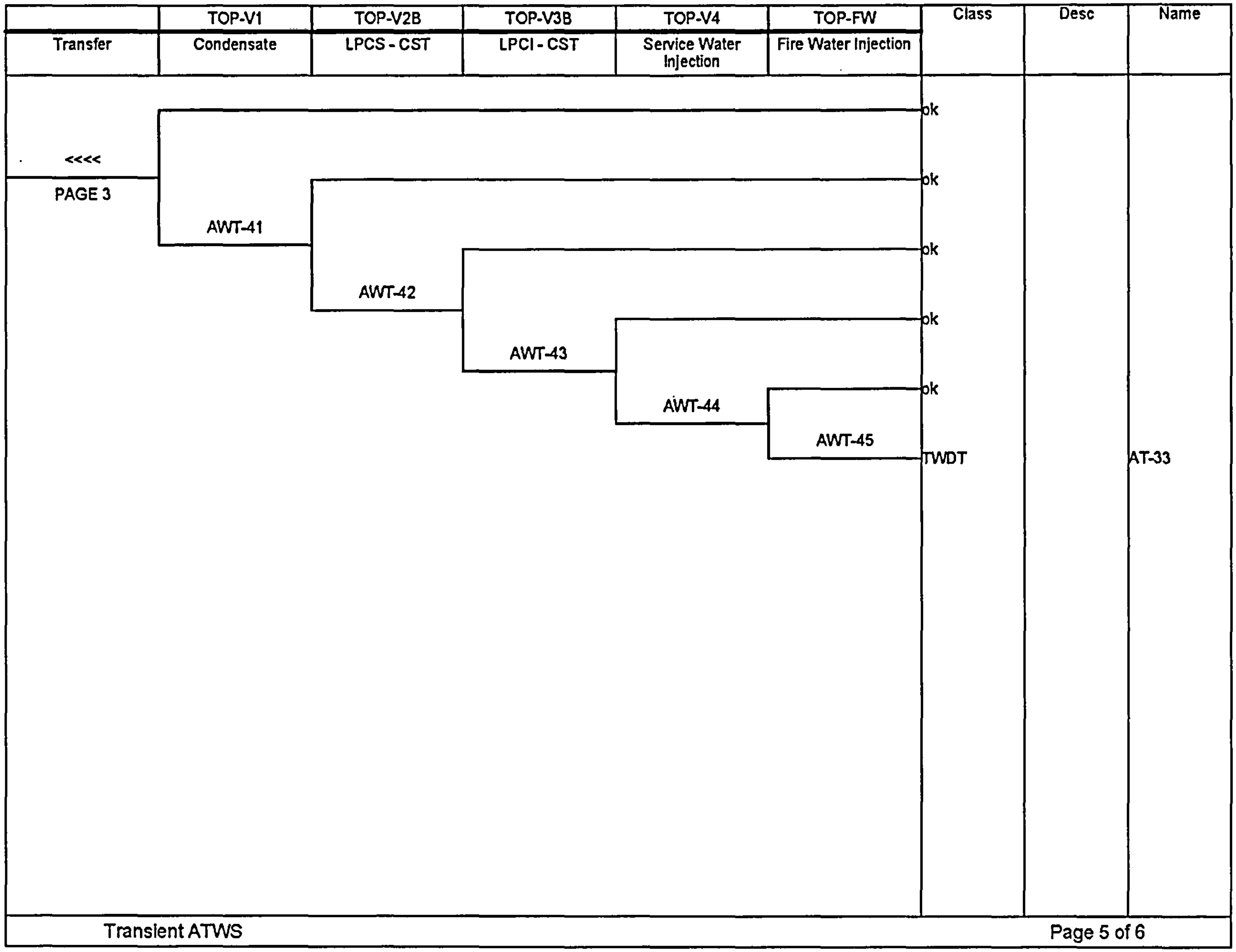


Transient ATWS

Page 2 of 6


Transfer
TOP-U1
HPCI
TOP-NX
ADS Inhibit
TOP-X2B
Depressurization
TOP-LVL
Level Control
TOP-V2
LPCS
TOP-V3
LPCI
TOP-W1
Suppression Pool Cooling
TOP-Y1
Containment Vent
Class
Desc
Name
ok
Page 4
AWT-24
AWT-25
TWDT
AT-24
AWT-23
LVL2
ATWS
AT-25
AWT-22
NX2
ATWS
AT-26
ok
Page 5
AWT-30
AWT-31
TWDT
AT-34
<<<<
PAGE 1
ok
Page 6
AWT-33
AWT-29
AWT-34
TWDT
AT-42
AWT-32
ATSLC
AT-43
AWT-28
LVL3
ATWS
AT-44
AWT-21
AWT-27
ATSLC
AT-45
AWT-26
NX3
ATWS
AT-46
Transient ATWS
Page 3 of 6


TOP-X2A
TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
Class
Desc
Name
Transfer
Depressurization
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
<<<<
PAGE 3
AWT-35
AWT-36
AWT-37
AWT-38
AWT-39
AWT-40
ok
ok
ok
ok
ok
TWDT
AT-22
TWDT
AT-23
Transient ATWS
Page 4 of 6


TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
Class
Desc
Name
Transfer
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
<<<<
PAGE 3
AWT-41
AWT-42
AWT-43
AWT-44
AWT-45
ok
ok
ok
ok
ok
TWDT
AT-33
Transient ATWS
Page 5 of 6

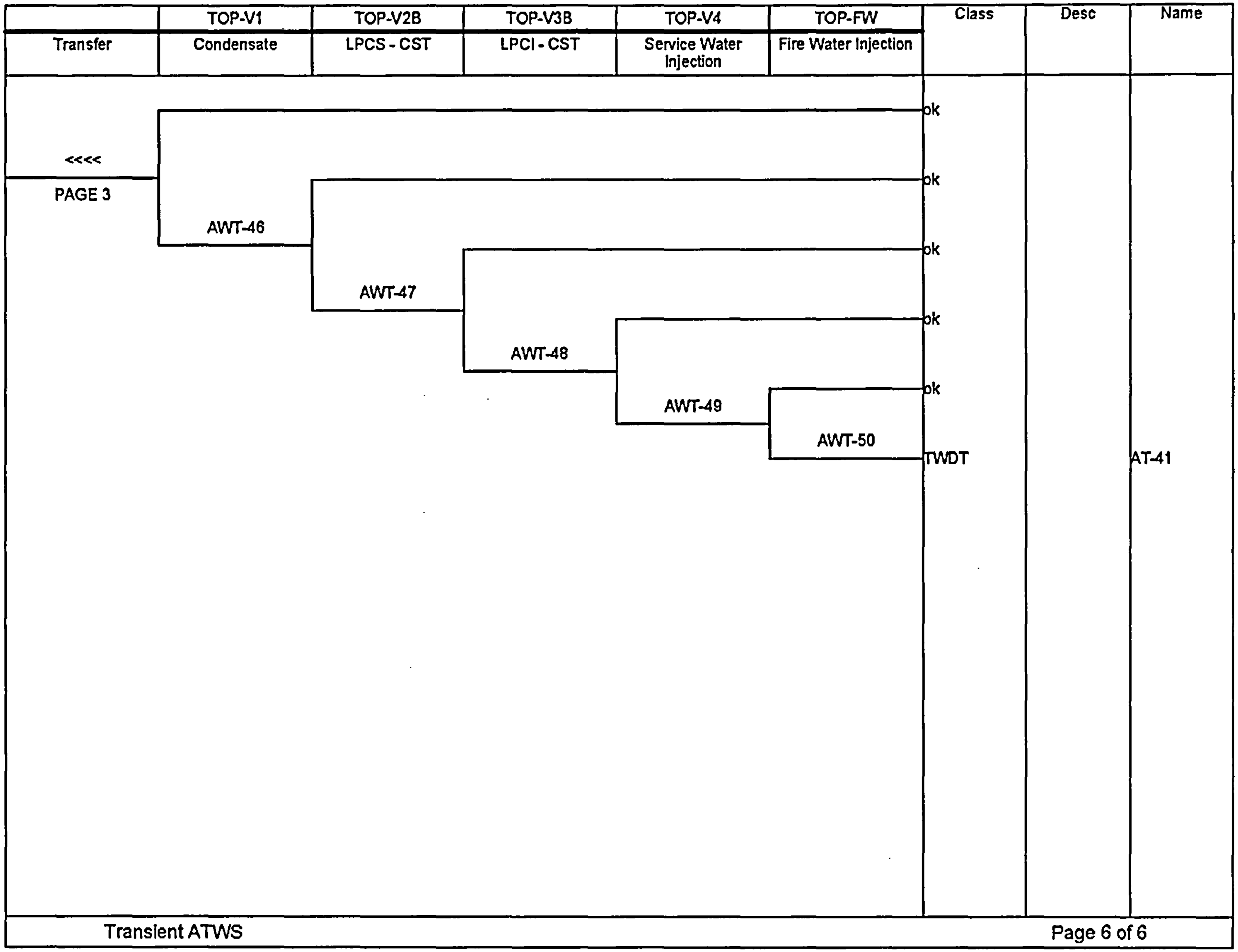
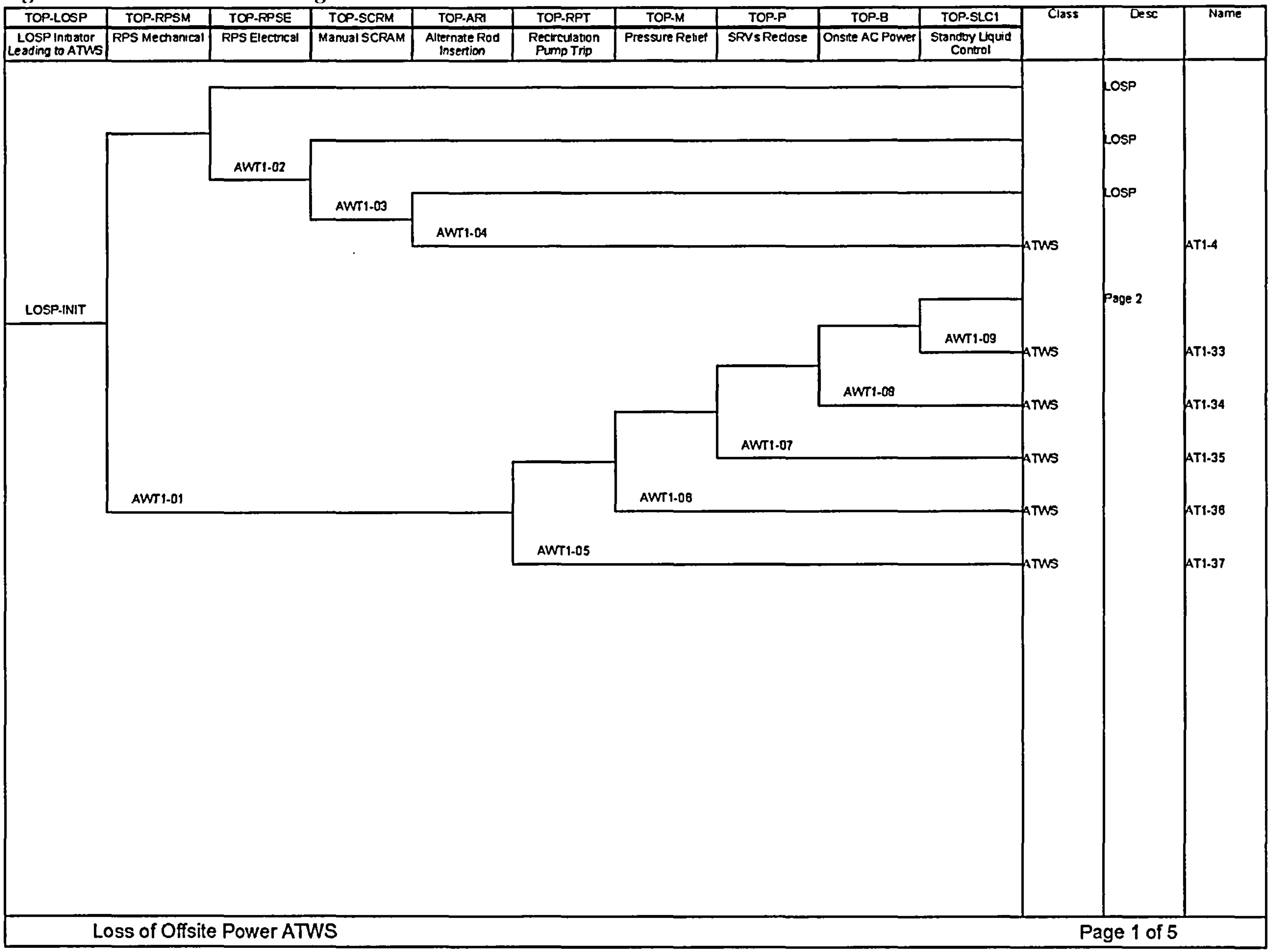
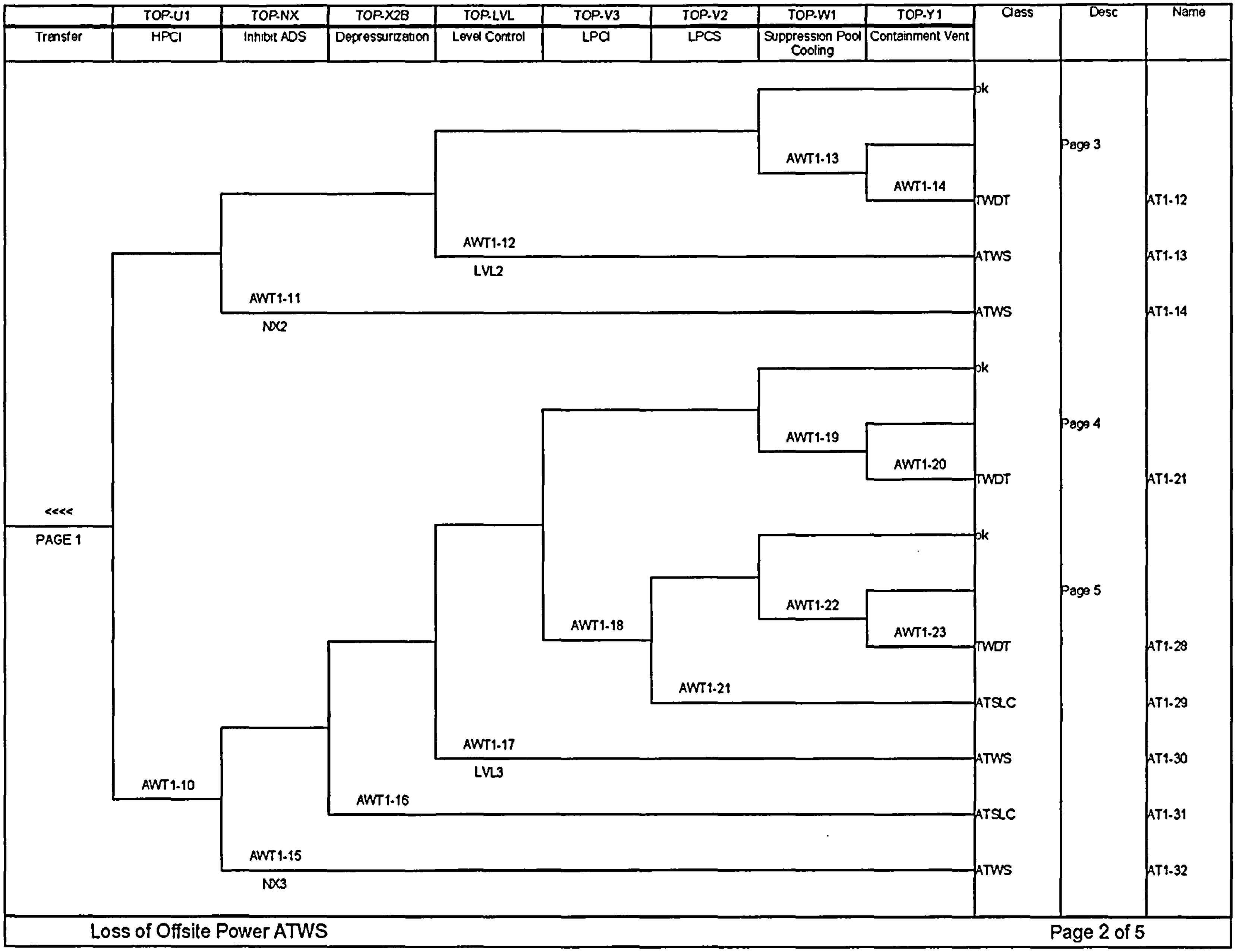
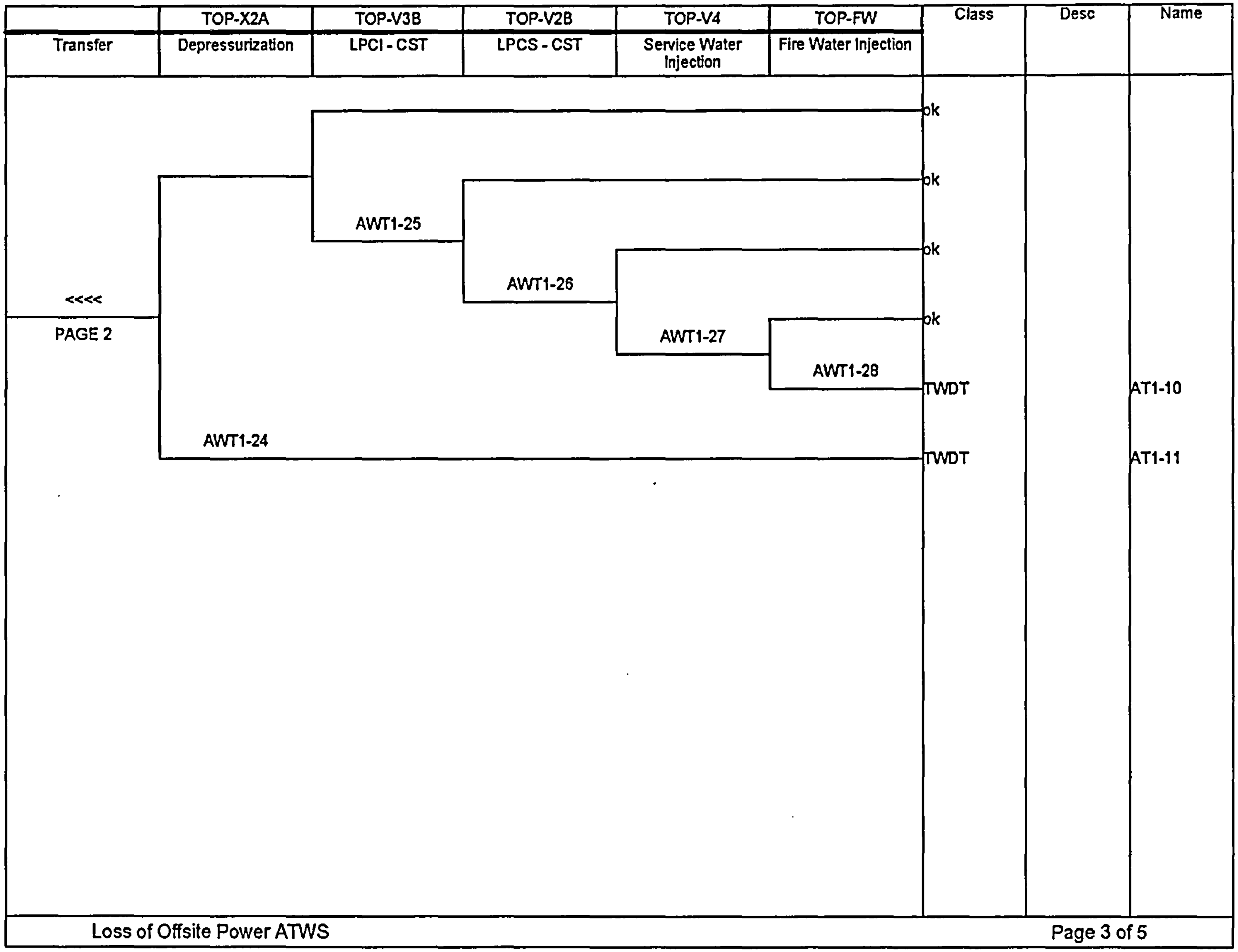
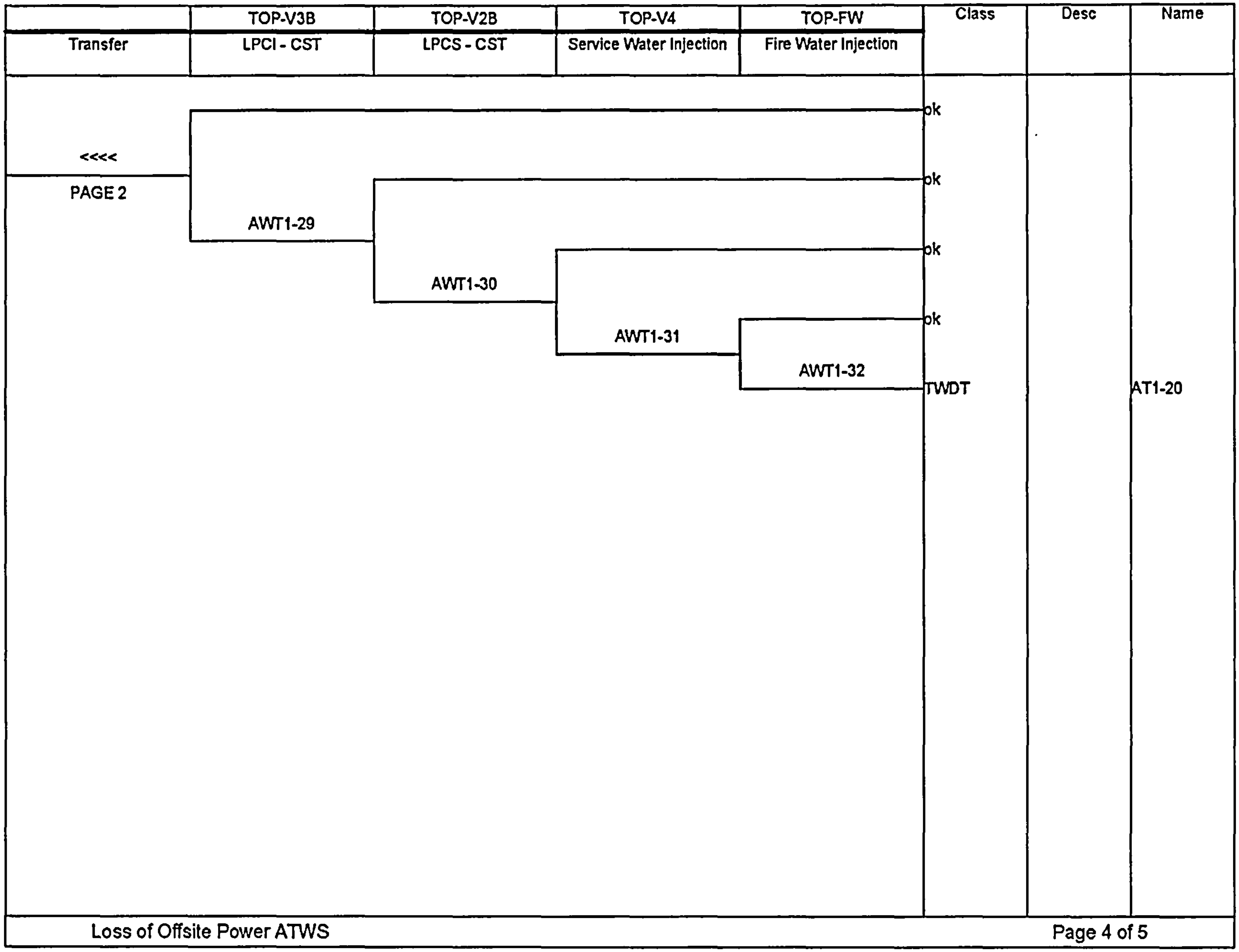

TOP-V1
TOP-V2B
TOP-V3B
TOP-V4
TOP-FW
Class
Desc
Name
Transfer
Condensate
LPCS - CST
LPCI - CST
Service Water Injection
Fire Water Injection
<<<<
PAGE 3
AWT-46
AWT-47
AWT-48
AWT-49
AWT-50
ok
ok
ok
ok
ok
TWDT
AT-41
Transient ATWS
Page 6 of 6

Figure C.3-8 ATWS Following a Loss of Offsite Power Event Tree




TOP-U1
TOP-NX
TOP-X2B
TOP-LVL
TOP-V3
TOP-V2
TOP-W1
TOP-Y1
Class
Desc
Name
Transfer
HPCI
Inhibit ADS
Depressurization
Level Control
LPCI
LPCS
Suppression Pool Cooling
Containment Vent
ok
Page 3
AWT1-13
AWT1-14
TWDT
AT1-12
AWT1-12
ATWS
AT1-13
LVL2
AWT1-11
ATWS
AT1-14
NX2
ok
Page 4
AWT1-19
AWT1-20
TWDT
AT1-21
<<<<
PAGE 1
ok
Page 5
AWT1-22
AWT1-18
AWT1-23
TWDT
AT1-28
AWT1-21
ATSLC
AT1-29
AWT1-17
ATWS
AT1-30
LVL3
AWT1-10
AWT1-16
ATSLC
AT1-31
AWT1-15
ATWS
AT1-32
NX3
Loss of Offsite Power ATWS
Page 2 of 5


TOP-X2A
TOP-V3B
TOP-V2B
TOP-V4
TOP-FW
Class
Desc
Name
Transfer
Depressurization
LPCI - CST
LPCS - CST
Service Water Injection
Fire Water Injection
ok
ok
ok
ok
AWT1-25
AWT1-26
<<<<
PAGE 2
AWT1-27
AWT1-28
TWDT
AT1-10
AWT1-24
TWDT
AT1-11
Loss of Offsite Power ATWS
Page 3 of 5


TOP-V3B
TOP-V2B
TOP-V4
TOP-FW
Class
Desc
Name
Transfer
LPCI - CST
LPCS - CST
Service Water Injection
Fire Water Injection
<<<<
PAGE 2
AWT1-29
AWT1-30
AWT1-31
AWT1-32
ok
ok
ok
ok
TWDT
AT1-20
Loss of Offsite Power ATWS
Page 4 of 5

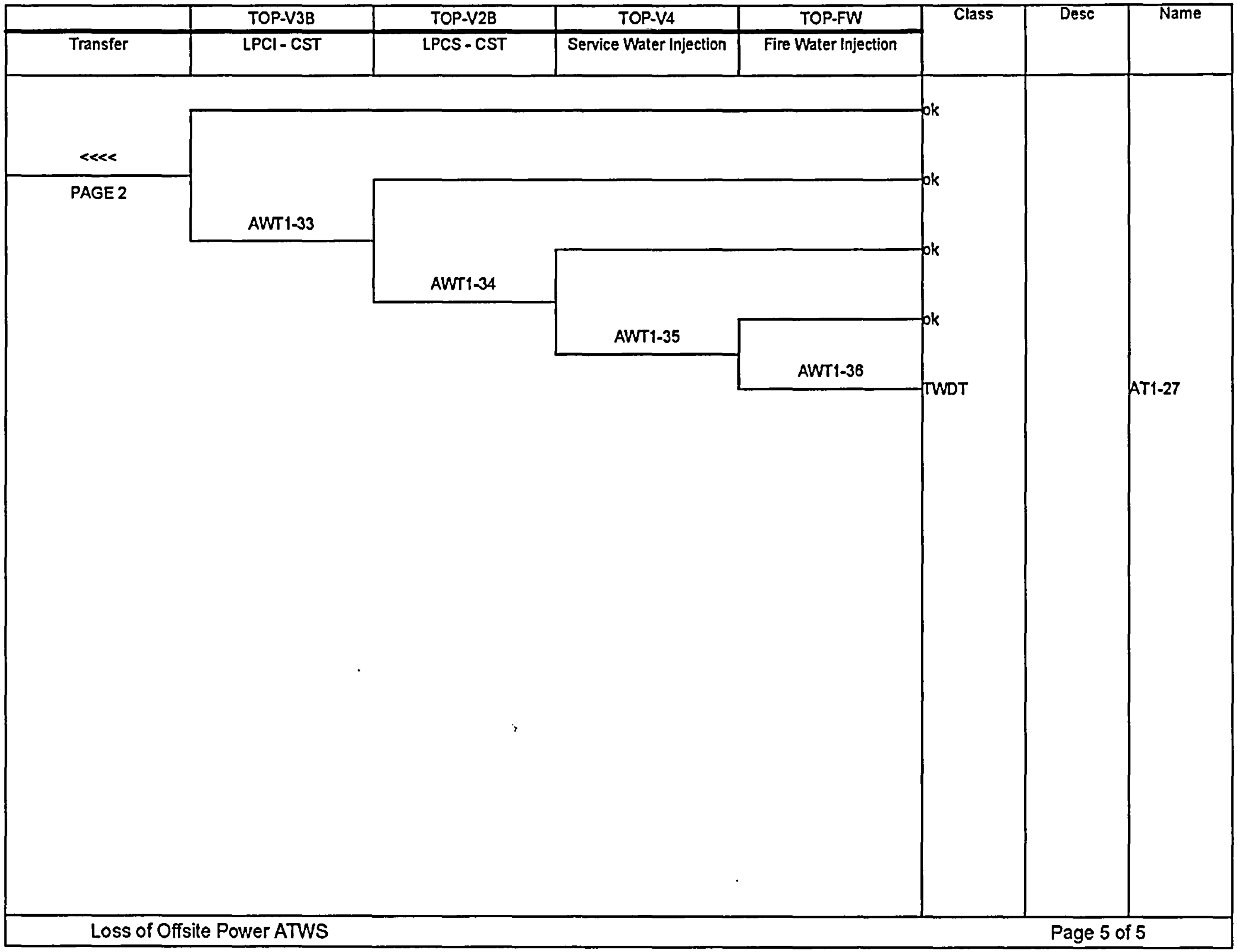

TOP-V3B
TOP-V2B
TOP-V4
TOP-FW
Class
Desc
Name
Transfer
LPCI - CST
LPCS - CST
Service Water Injection
Fire Water Injection
<<<<
PAGE 2
AWT1-33
AWT1-34
AWT1-35
AWT1-36
ok
ok
ok
ok
TWDT
AT1-27
Loss of Offsite Power ATWS
Page 5 of 5

Figure C.3-9 ATWS Following a Large or Medium LOCA Event Tree

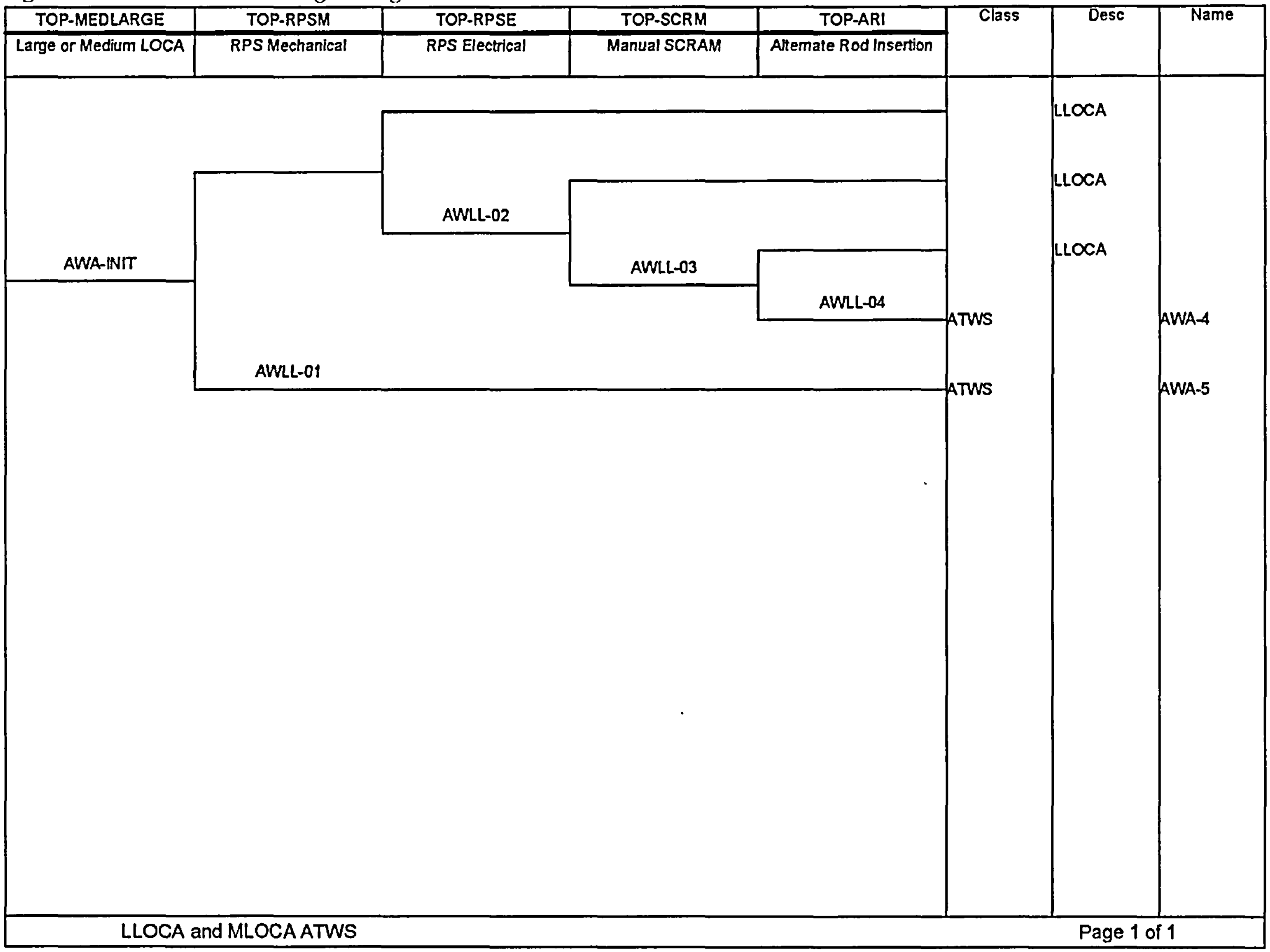


Figure C.3-10 Range of Plant Specific LBLOCA Frequencies

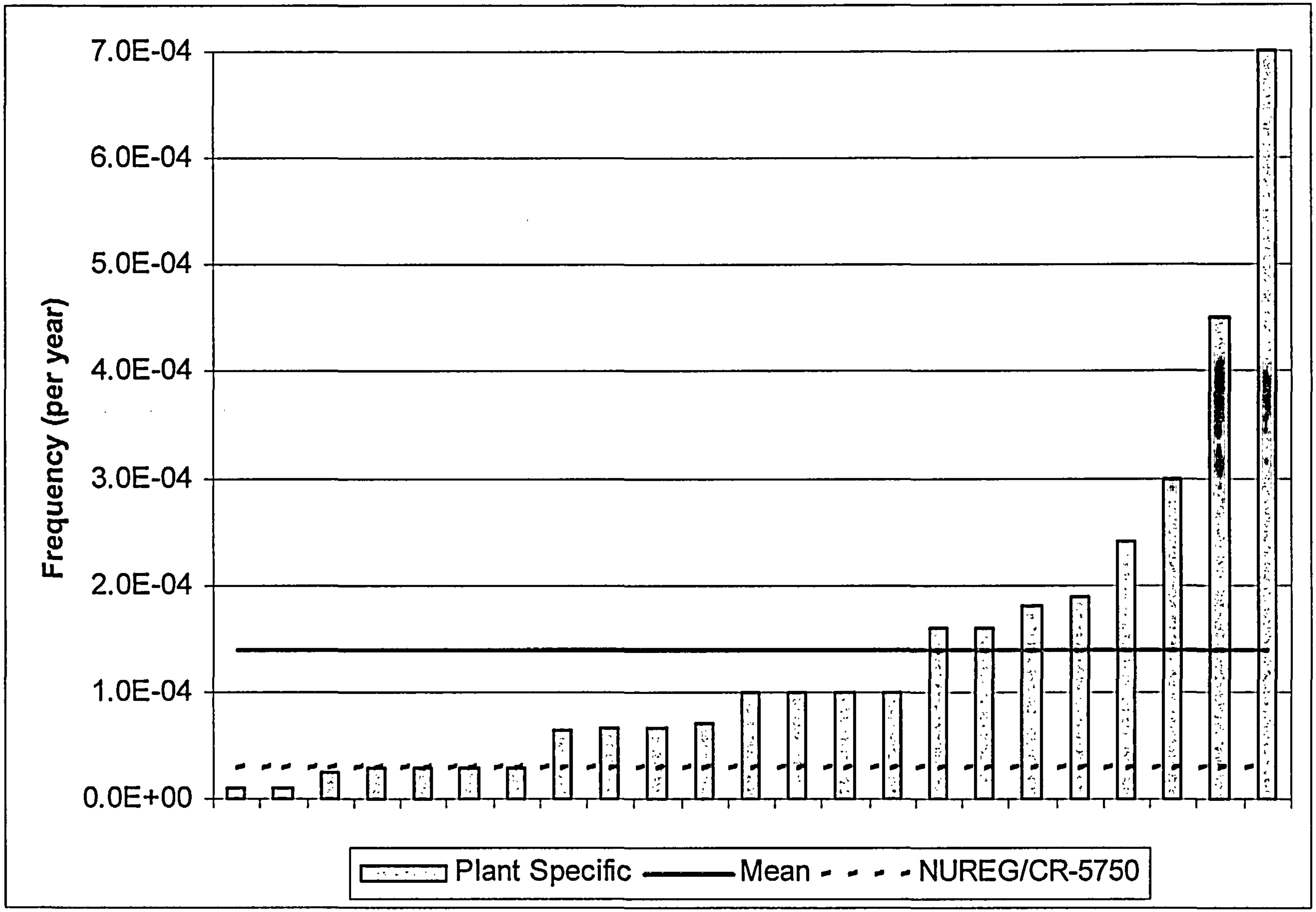


Figure C.3-11 Range of Plant Specific MBLOCA Frequencies

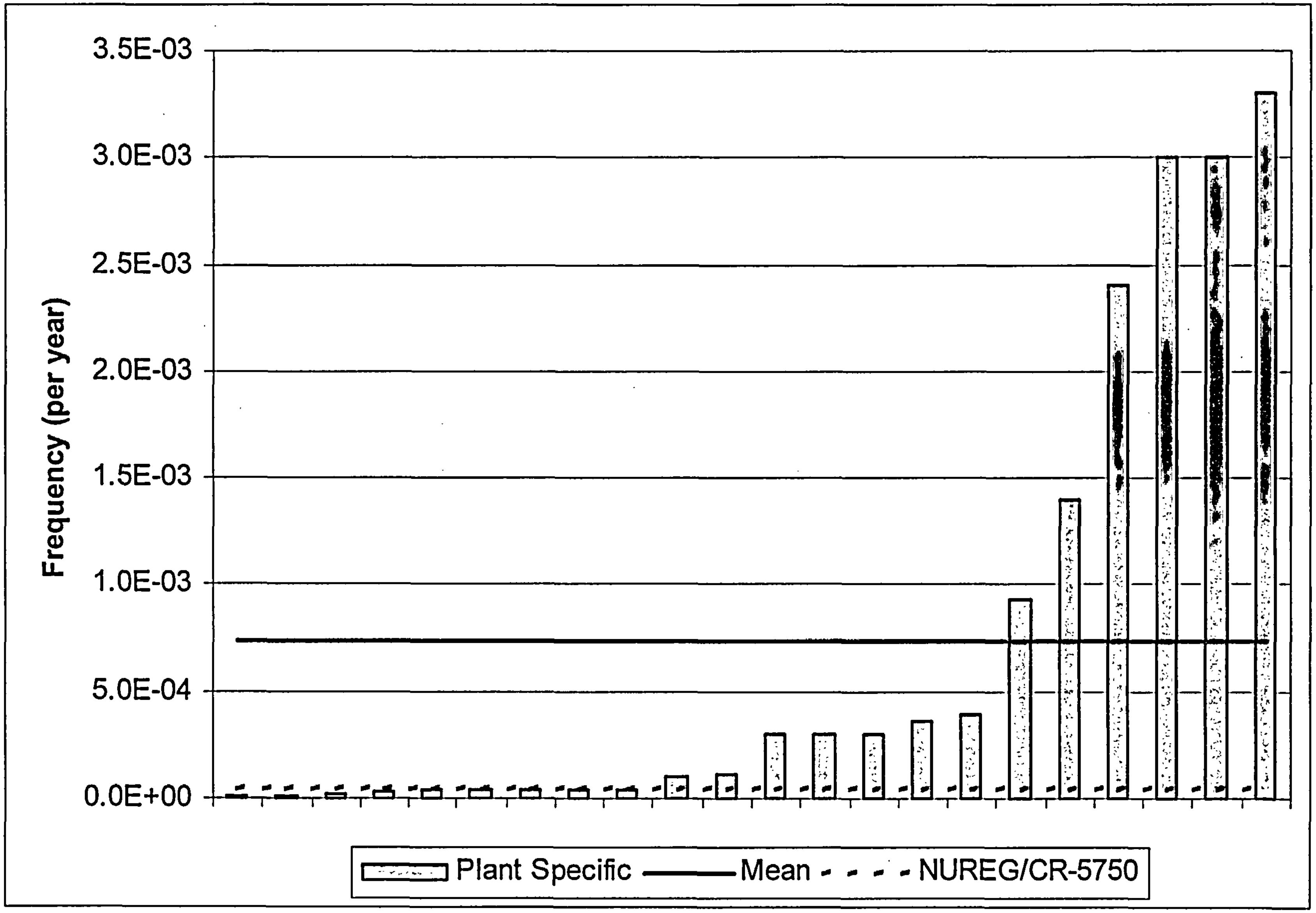


Figure C.3-12 Range of Plant Specific SBLOCA Frequencies

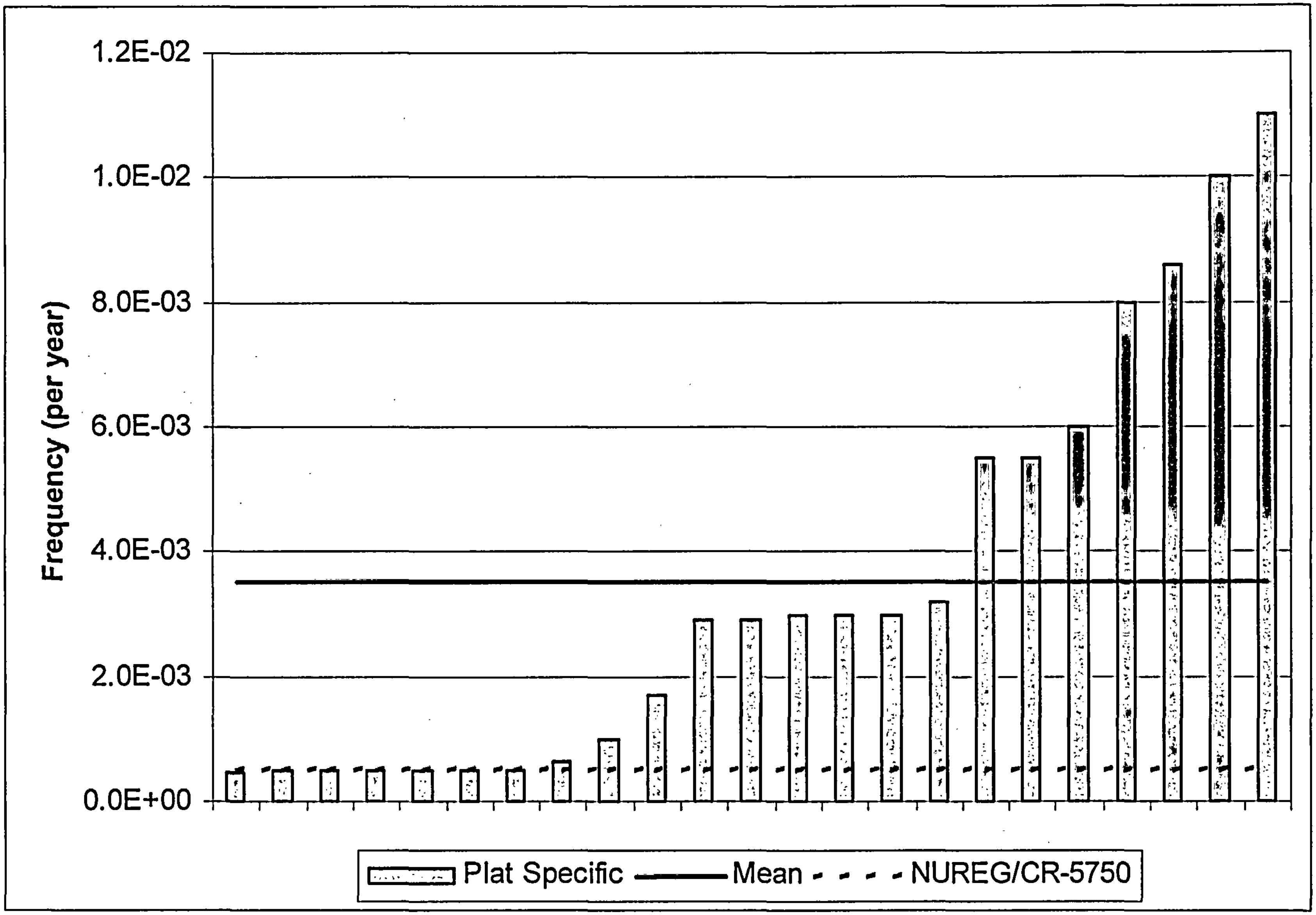


Figure C.3-13 Range of Plant Specific LOOP Frequencies

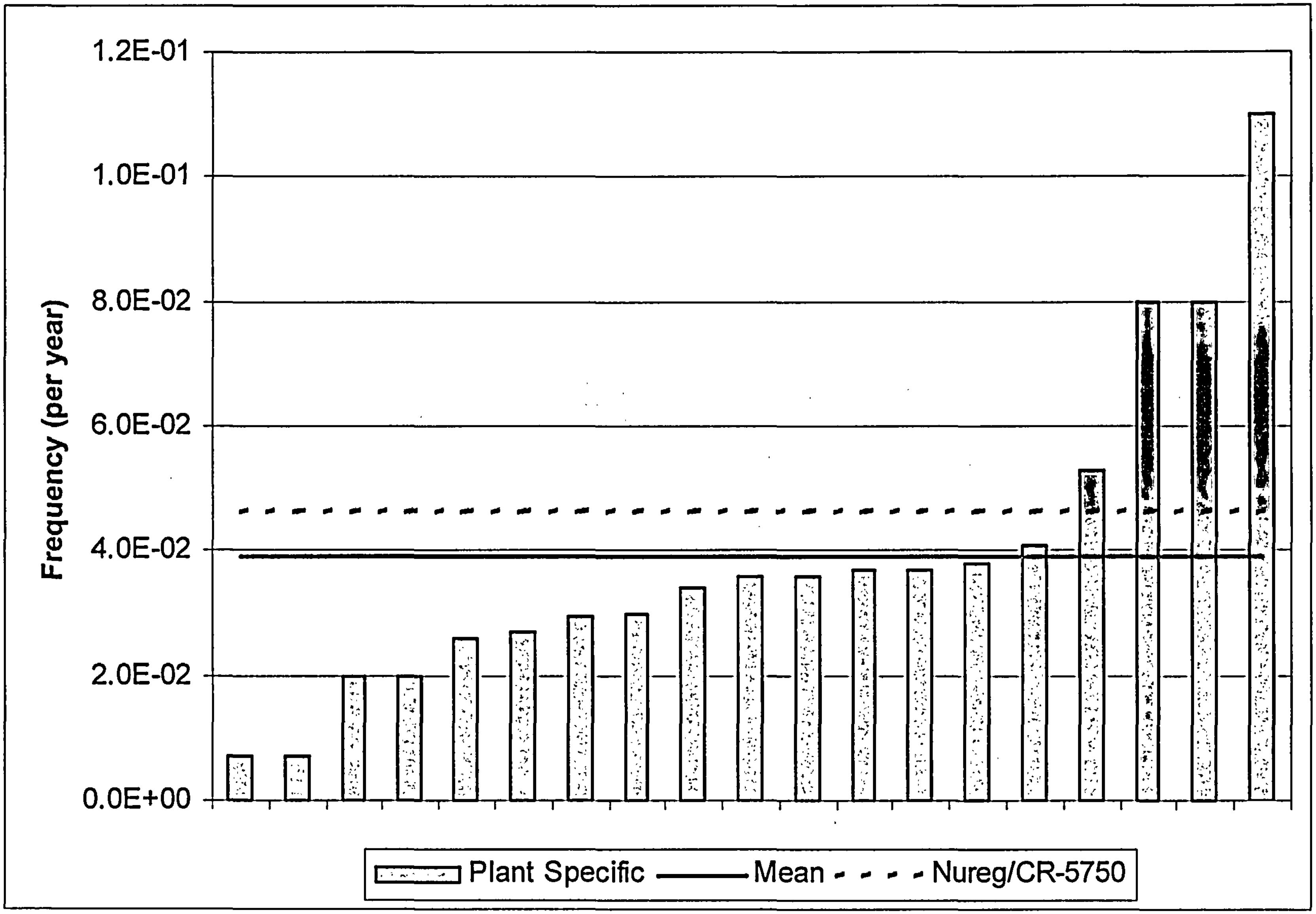


Figure C.3-14 Distribution for BWR 4, No AC Independent Injection, No Independent AC Power

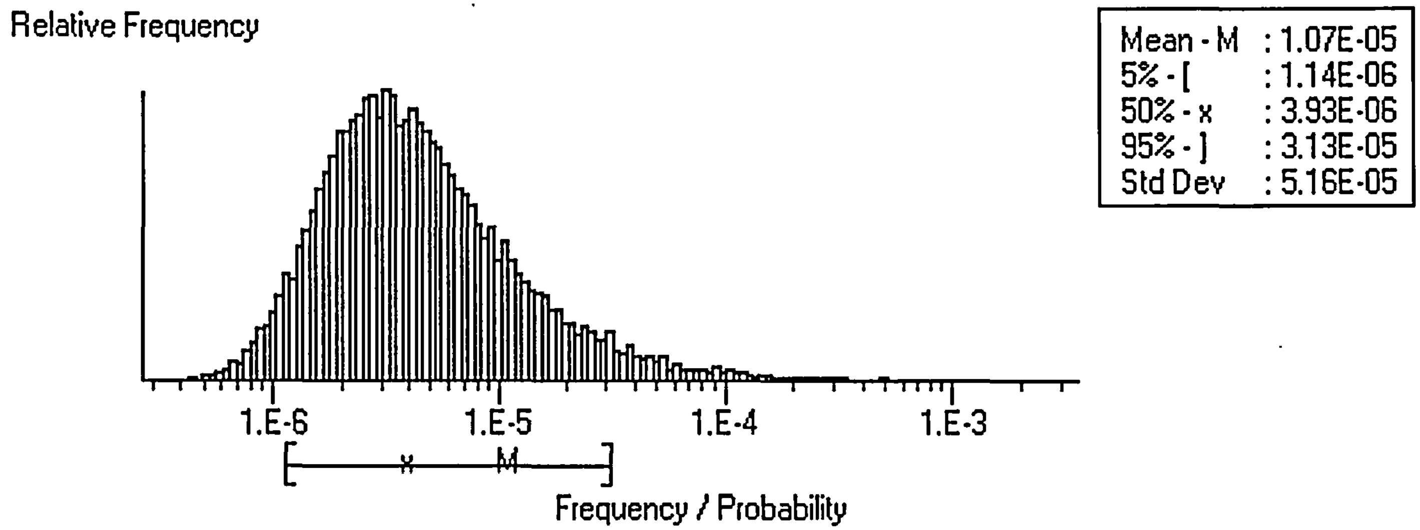


Figure C.3-15 Distribution for BWR 4, With AC Independent Injection, No Independent AC Power

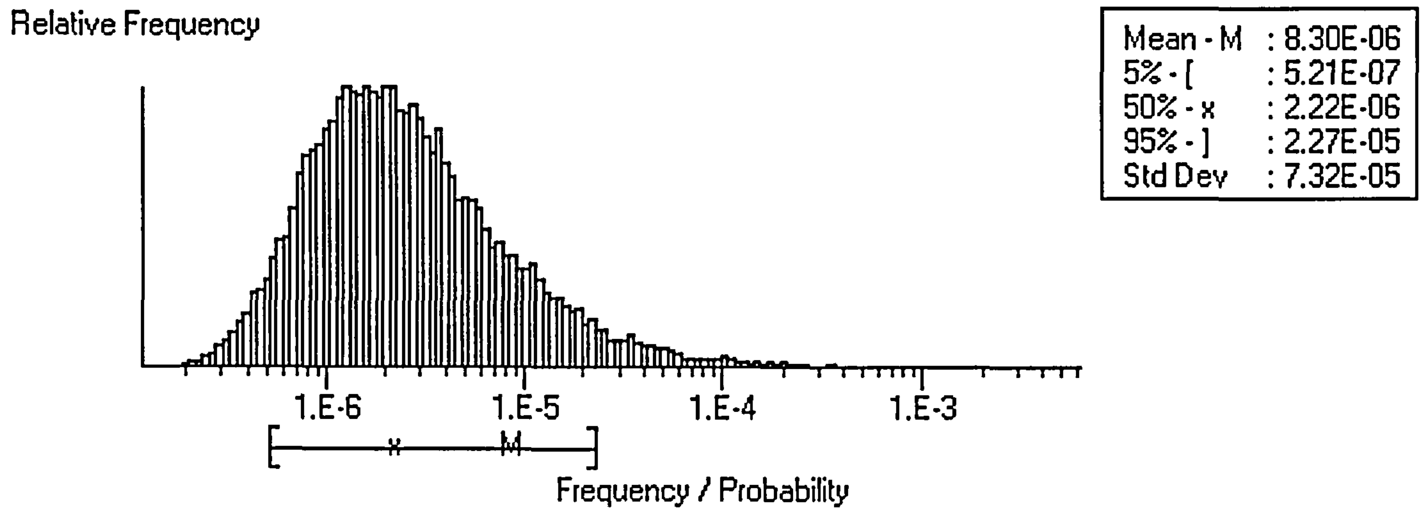


Figure C.3-16 Distribution for BWR 4, With Independent AC Power

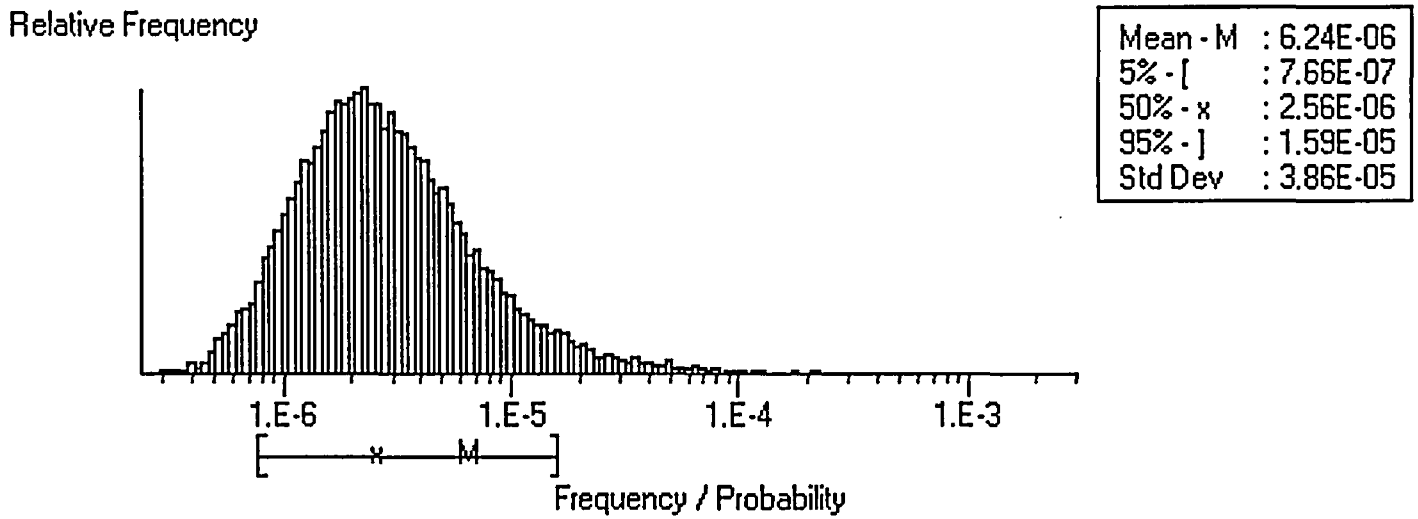


Figure C.3-17 Distribution for BWR 6, No AC Independent Injection, No Independent AC Power

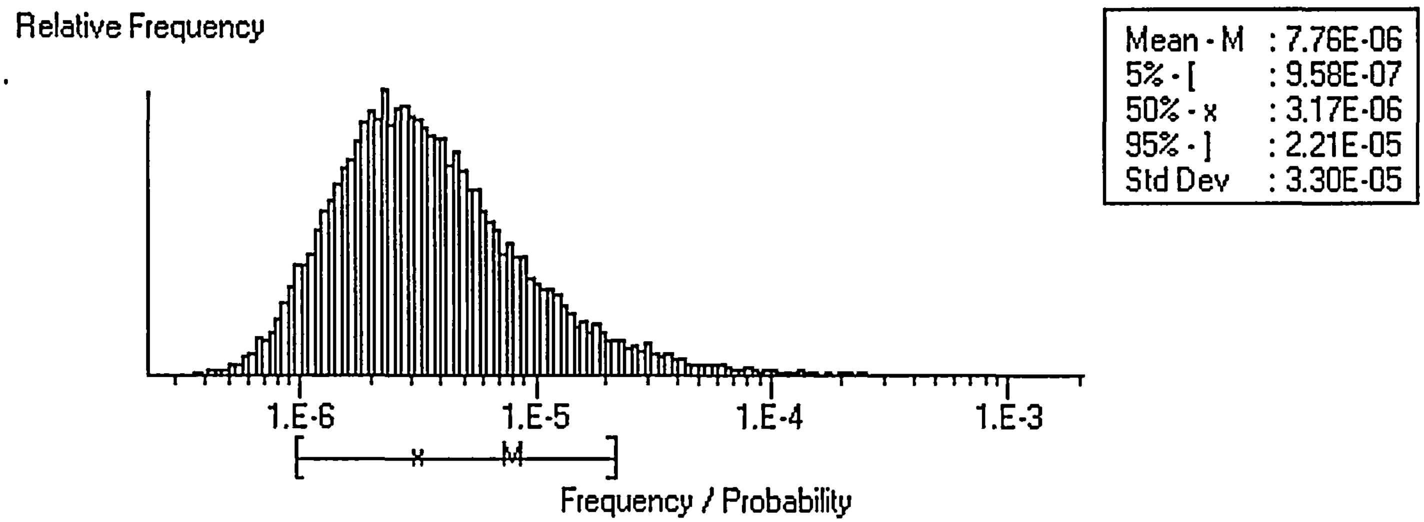


Figure C.3-18 Distribution for BWR 6, With AC Independent Injection, No Independent AC Power

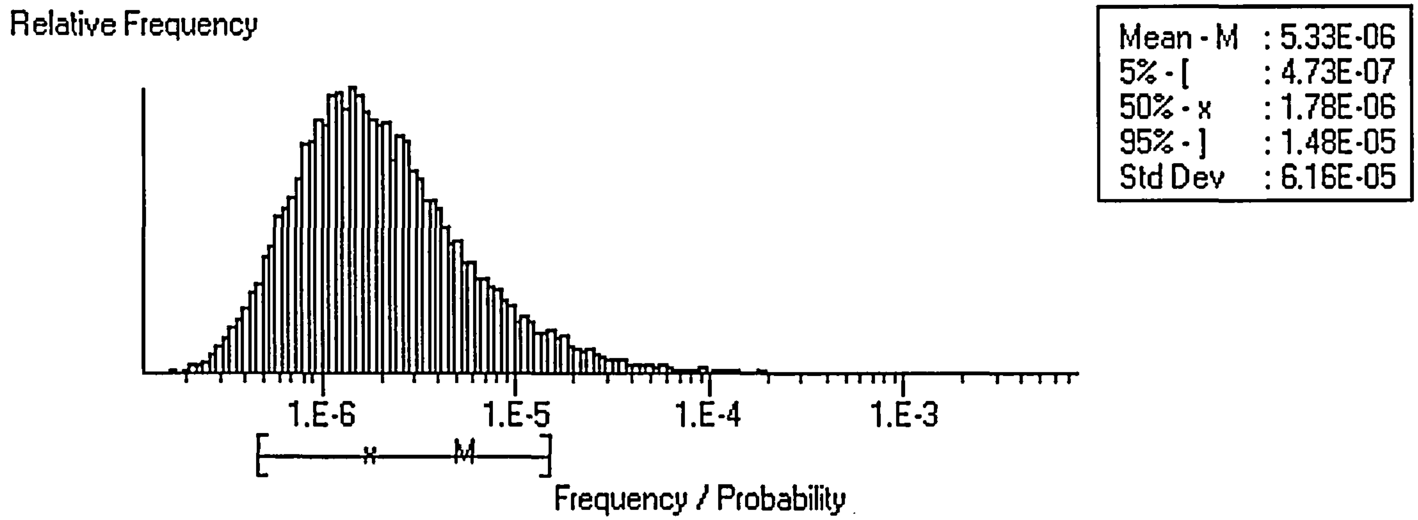


Figure C.3-19 Distribution for BWR 6, With Independent AC Power

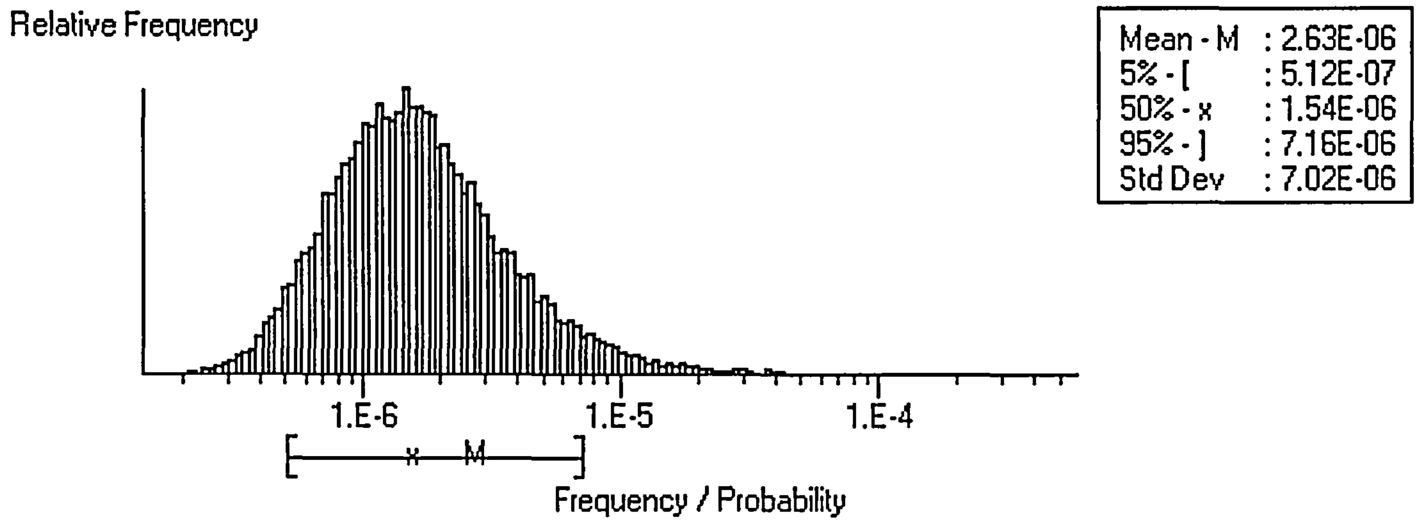


Figure C.4-1 Loss Of Offsite Power Initiating Event Logic

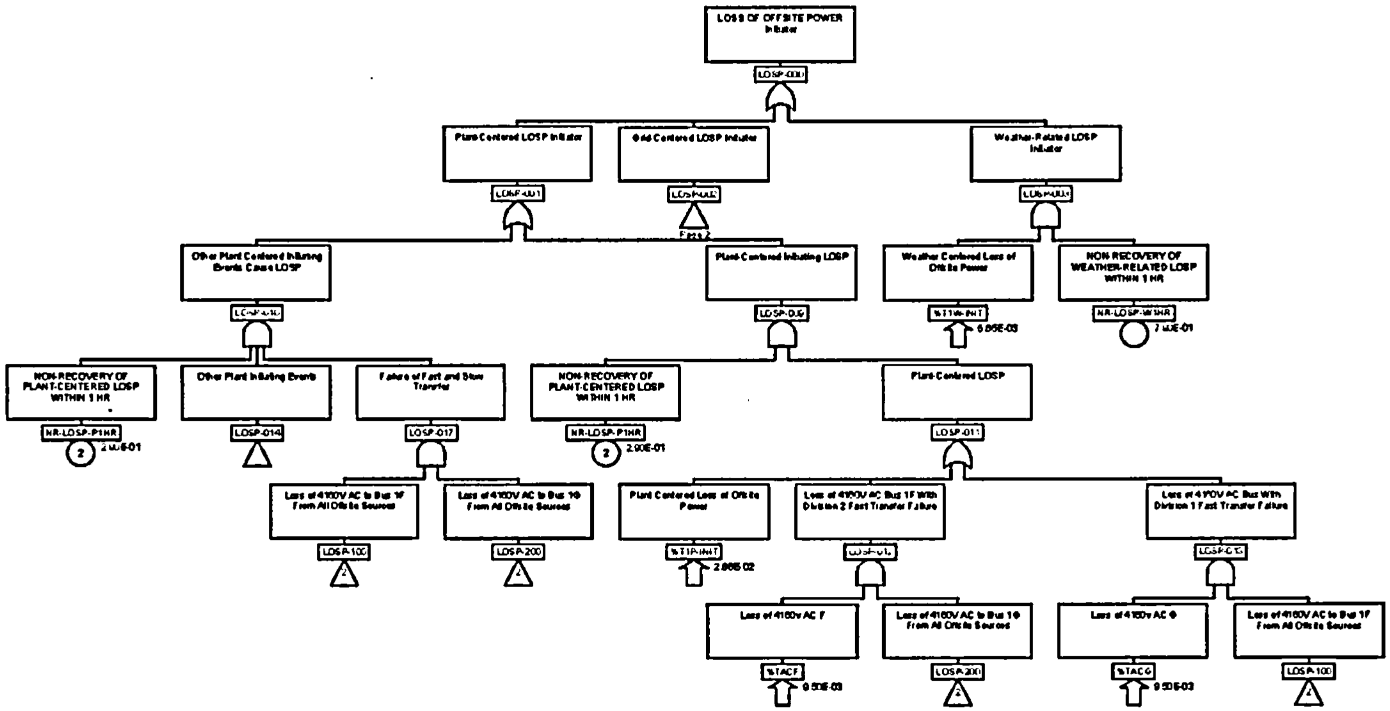


	Figure C.4-1		Page 1 of 2

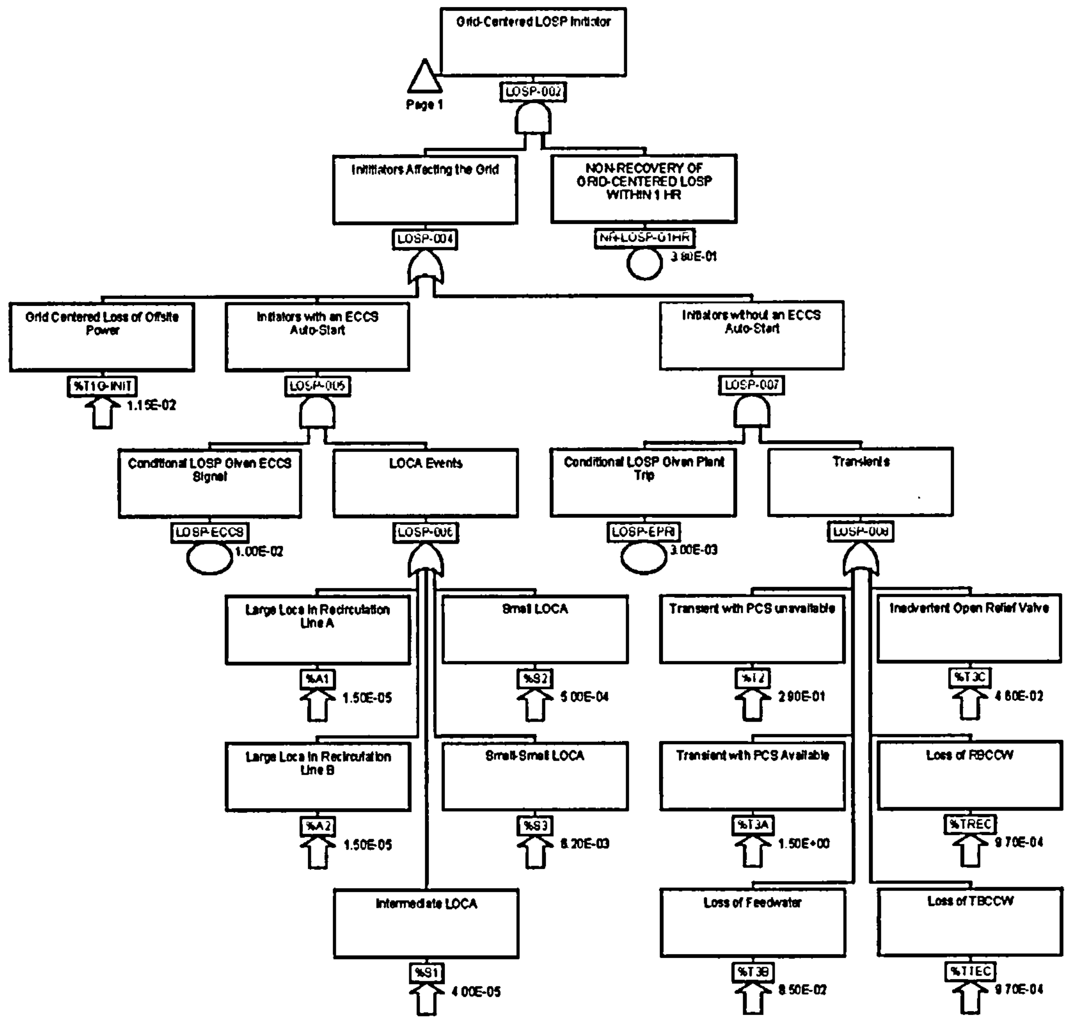
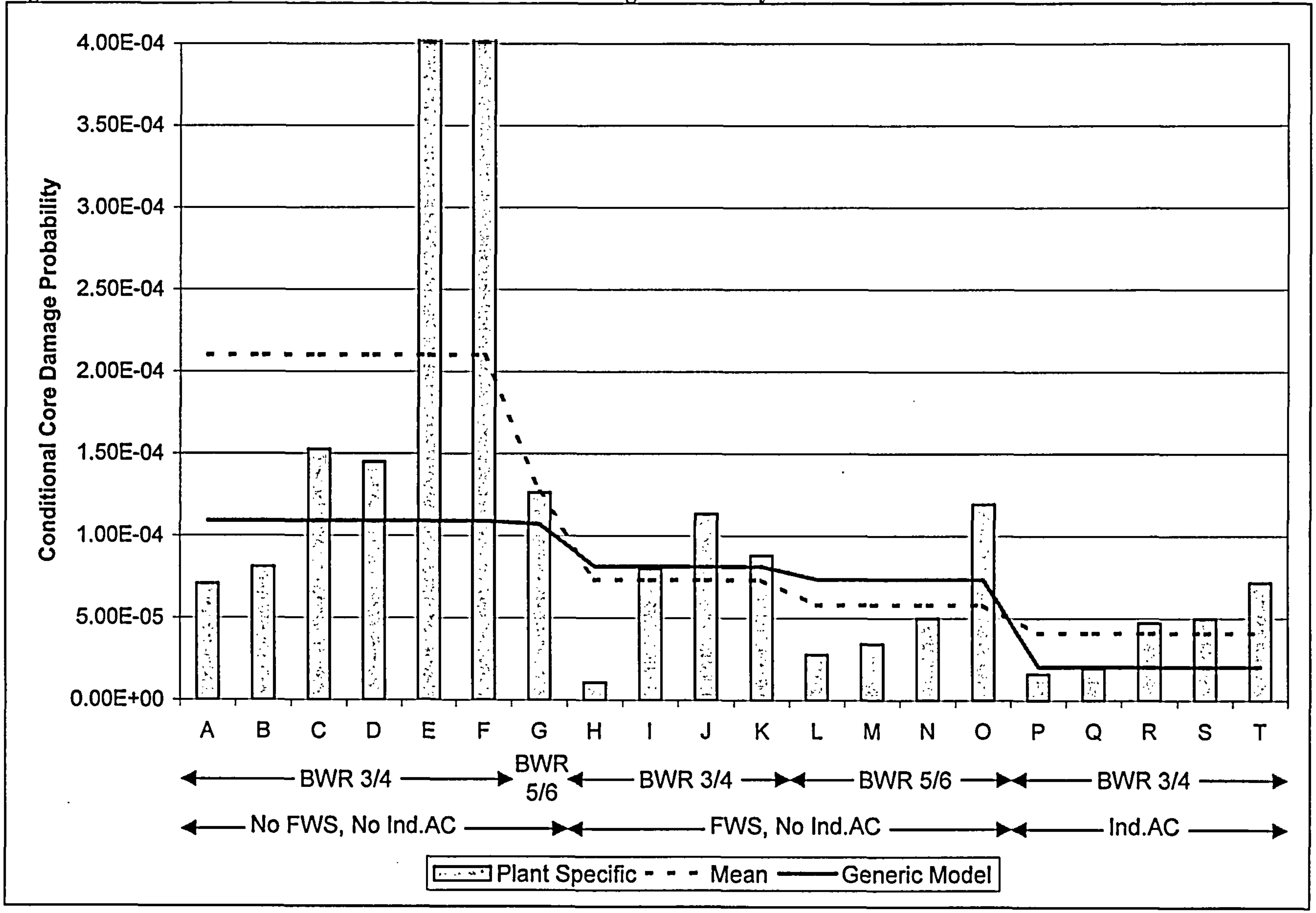


	Figure C.4-1		Page 2 of 2

Figure C.4-2 Loss of Offsite Power Conditional Core Damage Probability



Plant Attributes for Figure C.4-2

Plant	Type	AC Independent Injection	Independent AC Power Source	Battery Life	Number of Diesel Generators per Unit
A	BWR 4	No	No	4 hr	4
B	BWR 4	No	No	4 hr	2
C	BWR 4	No	No	4 hr	4
D	BWR 4	No	No	2.5 hr	2 + swing
E	BWR 4	No	No	2 hr	2
F	BWR 4	No	No	2 hr	2
G	BWR 6	No	No	4 hr	2
H	BWR 4	Yes	No	8 hr	4
I	BWR 4	Yes	No	5 hr	2
J	BWR 4	Yes	No	4 hr	2
K	BWR 4	Yes	No	2 hr	4
L	BWR 5	Yes	No	6 hr	3
M	BWR 5	Yes	No	7 hr	2 + swing for 1 division
N	BWR 6	Yes	No	8 hr	No response
O	BWR 6	Yes	No	4 hr	2
P	BWR 4	Yes	Yes	4 hr	1.5
Q	BWR 3	Yes	Yes	4 hr	1.5
R	BWR 4	No	Yes	4 hr	4
S	BWR 4	No	Yes	4 hr	4
T	BWR 4	Yes	No	2 hr	2

Figure C.4-3 Medium LOCA Conditional Core Damage Frequency

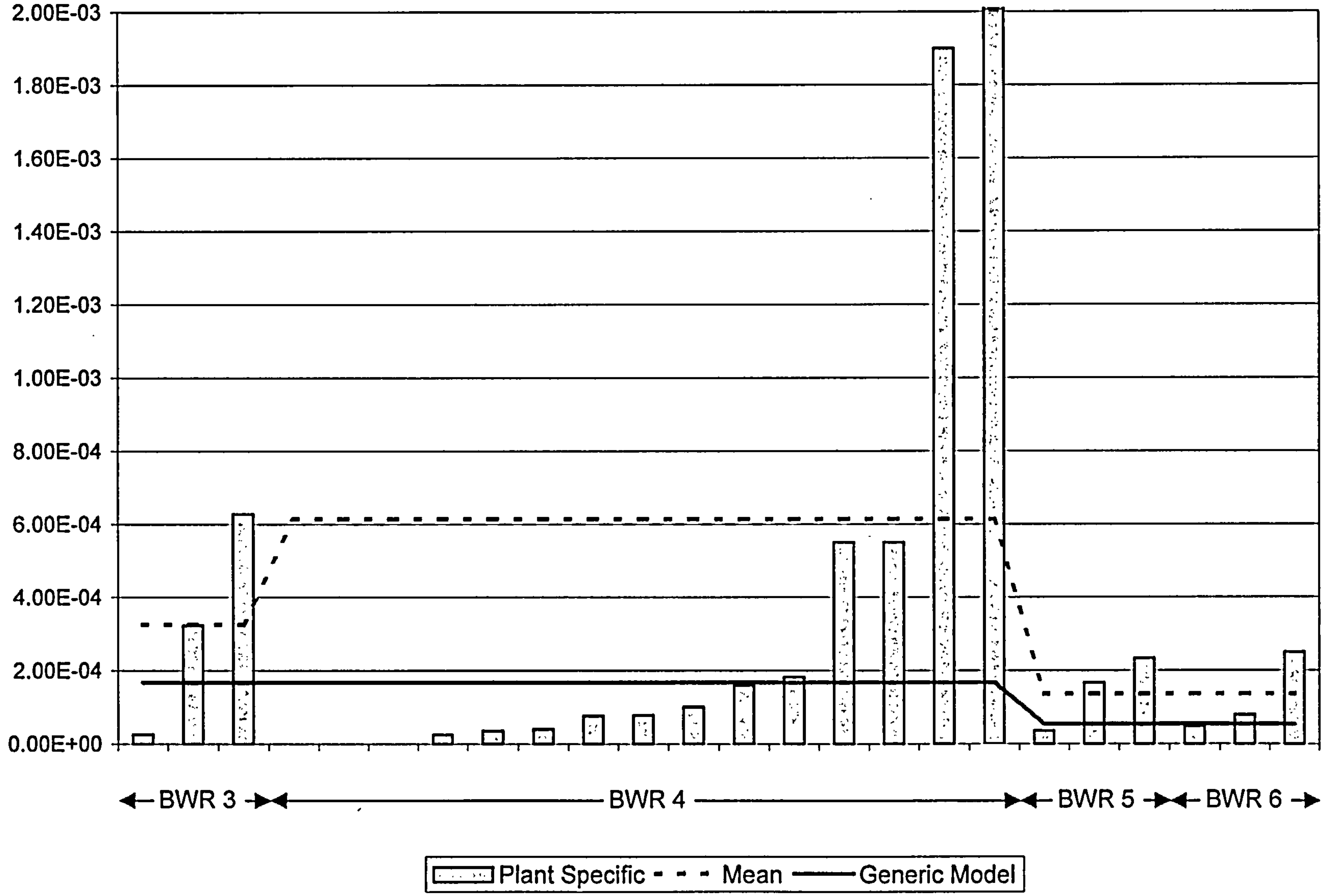


Figure C.5-1 CDF Offset by Varying EDG Parameters