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United States Nuclear Regulatory Commission
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Subject: Individual Plant Examination of External Events for Severe Accident Vulnerabilities for the Davis-Besse Nuclear Power Station, Unit 1 (Response to Generic Letter 88-20, Supplement 4)

Ladies and Gentlemen:

Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-20, Supplement 4, "Individual Plant Examination of External Events for Severe Accident Vulnerabilities," dated June 28, 1991 requested each licensee to undertake an individual plant examination of external events (IPEEE) to identify vulnerabilities, if any, to severe accidents and report the results to the NRC. Toledo Edison has completed the IPEEE for the DBNPS. The purpose of this letter is to transmit the summary report for the DBNPS IPEEE.

By letter dated December 17, 1991 (Serial Number 1997), Toledo Edison committed to performing an IPEEE for the Davis-Besse Nuclear Power Station (DBNPS), including an assessment of the five external events (seismic, internal fire, high winds and tornadoes, external floods, and transportation and nearby facility accidents) using NRC accepted methodology. Toledo Edison also committed to coordinate activities associated with the closure of Unresolved Safety Issue (USI) A-46, "Verification of Seismic Adequacy of Equipment in Operating Plants," (USI a-46) with those required to perform the seismic portion of the IPEEE. Toledo Edison did not identify a specific methodology to be used for evaluation of seismic events pending the issuance of the Seismic Qualification Utility Group (SQUG) Generic Implementation Plan (GIP) for resolution of USI A-46. Toledo Edison committed to complete the IPEEE, with the exception of the seismic evaluation, by September 1995.

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The NRC, by letter dated August 11, 1992 (Log Number 3804), requested that TE identify the specific methodology to be utilized and reconsider the submittal schedule. Toledo Edison responded to this request by letter dated September 18, 1992 (Serial Letter 2089) and committed to performing a Focused-Scope Seismic Margins Assessment (SMA) using the Electric Power Research Institute (EPRI) Seismic Margins Methodology described in EPRI Report NP-6041-SL, Revision 1, " A Methodology for Assessment of Nuclear Power Plant Seismic Margin," (NP-6041-SL), with the enhancements discussed in Section 3.2.4 of NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," (NUREG-1407). Toledo Edison also committed to using the NRC accepted EPRI Fire Induced Vulnerability Evaluation (FIVE) Methodology and evaluation of high winds, external floods, and transportation and nearby facility accidents in accordance with the screening approach shown in Figure 1 of GL 88-20, Supplement 4, and described in Section 5 of NUREG-1407. Further, where the acceptance criteria of NUREG 75/087 (now NUREG-0800), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," dated December 1975, are not satisfied additional evaluations would be performed to the extent necessary to ensure that any vulnerabilities are identified and addressed or shown to be insignificant. In order to coordinate the seismic walkdowns required for completion of the IPEEE with those required for resolution of USI A-46, limit adverse impact of these walkdowns on the upcoming refueling outage critical path schedules and plant availability, and utilize DBNPS engineers, TE could not realistically revise the commitment for completion of the IPEEE before September 1, 1995.

By letter dated August 25, 1994, (Serial Number 2242) Toledo Edison revised its commitments related to the DBNPS IPEEE. Specifically, TE revised its plans to address the seismic portion of the IPEEE by performing a Reduced-Scope SMA rather than a Focused-Scope SMA. Toledo Edison realized that the NRC Staff, as discussed in Information Notice (IN) 94-32, "Revised Seismic Hazards," dated April 24, 1994, was re-evaluating the appropriate SMA scope for all plants and expected to complete their re-evaluation by December, 1994. The NRC Staff was then to notify licensees of their conclusions. The NRC issued IN 94-32 in order to disseminate the information contained in NUREG-1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," performed by Lawerence Livermore National Laboratory (LLNL). However, due to the Ninth Refueling Outage commencing on October 1, 1994, Toledo Edison had to decide whether it would remain a focused-scope SMA plant or implement a reduced-scope SMA, in order to determine the appropriate extent of information gathering to be performed during the Ninth Refueling Outage walkdowns.

The change to a reduced-scope SMA was based on the stated schedule constraints, TE's review of the revised LLNL report, IN 94-32, and SECY-94-166, dated June 17, 1994, and an estimated cost savings in excess of $250,000 by performing a reduced-scope SMA rather than a focused-scope SMA. During the Ninth Refueling Outage (October 1, 1994 - November 15, 1994), TE completed the necessary walkdowns and information collection for the reduced-scope SMA.

Toledo Edison also extended its IPEEE submittal date to March 1, 1996, based on a determination that with a change in the scope of the study to a Reduced-Scope, a further reduction in reliance on resources external to TE was possible, therefore creating an even greater emphasis on performing most of the work in-house to further control costs and retain the insight and knowledge gained from such a study.

Generic Letter 88-20, Supplement 5, dated September 8, 1995, provided the results of the NRC Staff's re-evaluation of the SMA scope based on the 1994 revised LLNL report. Generic Letter 88-20, Supplement 5, identified modifications in the NRC's recommended scope of seismic reviews that are performed as part of the IPEEE for focused-scope and full-scope Seismic Margins Assessment (SMA) plants. Generic Letter 88-20, Supplement 5 discussed the results of the Lawerence Livermore National Laboratory (LLNL) report NUREG-1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Plant Sites East of the Rocky Mountains," dated April 4, 1994. The results of the revised LLNL seismic estimates showed the perceived seismic hazards and associated risks are less than previously perceived for plants located in the central and eastern United States. Licensees of focused-scope and full-scope SMA plants who planned to modify their seismic IPEEE to utilize the guidance in GL 88-20, Supplement 5 were requested to inform the NRC of their modification, including any schedule changes.

By letter dated November 6, 1995, (Serial Number 2341), Toledo Edison provided its response to Generic Letter 88-20, Supplement 5. Toledo Edison committed to continue to perform its reduced-scope SMA using the SQUG methodology. For these evaluations the selection of equipment is based on the guidance given in NP-6041-SL. In addition, as requested in GL 88-20, Supplement 5, Toledo Edison committed to perform the following items: identification of "bad actor" relays, identification and evaluation of flat bottom tanks whose failure could significantly affect plant safety, and consideration of "other items" as identified in Attachment 1 to GL 88-20, Supplement 5. Toledo Edison stated that these evaluations would utilize the DBNPS design basis earthquakes.

Generic Letter 88-20, Supplement 5, also requested that masonry and block walls be evaluated. In TE Serial Letter Number 2341 Toledo Edison stated that based on review of the docketed responses for the DBNPS to I.E. Bulletin 80-11 concerning the design margin for masonry and block walls, it was concluded that although these walls meet the licensing and design bases, there is insignificant additional margin beyond the design basis to justify an additional reanalysis effort. As stated in both SECY-95-213, "Proposed NRC Generic Letter 88-20 Supplement 5," dated August 17, 1995, and GL 88-20, Supplement 5, the scope of the seismic IPEEE may be revised by eliminating the need to calculate certain site effects which would not result in cost beneficial improvements. Accordingly, due to expected high modification costs, masonry and block walls are not included within the scope of the seismic IPEEE for the DBNPS.

Since GL 88-20, Supplement 5, identified additional items to be addressed, Toledo Edison rescheduled the IPEEE report submittal date to November 30, 1996. This date was chosen to support the DBNPS Tenth Refueling Outage pre-outage and outage activities and provide sufficient time to address the items requested.

The IPEEE assessments described in the summary report address the external events identified in Supplement 4 of GL 88-20, namely seismic events, internal fires, and other external phenomena such as high winds, extreme rainfall, transportation accidents, and any other credible external events.

Based on the results of the reduced-scope seismic margins analysis, no vulnerabilities to severe accidents were identified that would be attributable to seismic events, and no actions beyond those already identified by the SQUG program were found necessary.

For the internal fire assessment, supplementary probabilistic risk assessment methods were employed in conjunction with the EPRI FIVE Methodology. While four compartments were identified which did not fall below the FIVE screening criteria for compartment-specific core damage frequency, the bounding values calculated were of insufficient magnitude to constitute a vulnerability. Follow-up actions will include initiating a review of fire response procedures associated with plant areas which did not meet screening criteria to ensure that actions taken to satisfy 10 CFR 50 Appendix R compliance (e.g., pre-emptive tripping of plant equipment) are optimized with respect to maintaining the overall plant risk as low as reasonably achievable. In addition follow-up action is required in relocating two gas cylinders which could be a source of seismically induced fires.

For the assessment of other applicable external phenomena, a progressive screening approach was utilized as recommended in Section 5 of NUREG-1407. No events such as high winds, floods, or transportation accidents were found to be above the screening criteria. Therefore, no vulnerabilities were identified. Follow-up actions include Updated Safety Analysis Report and administrative program revisions to address on-site chemical hazards, and resolution of a Potential Condition Adverse to Quality Report initiated to address as-found roof drain conditions.

The IPEEE Generic Letter Supplement listed a variety of Unresolved Safety Issues, Generic Issues and other programs which were to be addressed as part of the external event analysis. Of those applicable to DBNPS, USI A-45, "Shutdown Decay Heat Removal Requirements," and NUREG/CR-5088, "Fire Risk Scoping Study," are considered to be resolved. It should be noted that the issues identified as part of USI A-46 are being addressed as part of the DBNPS SQUG program.

The principle finding of the IPEEE for DBNPS is that no vulnerabilities have been identified when evaluating any of the phenomena and plant areas considered. Follow-up actions are being taken, however, to improve overall plant risk with respect to external events where appropriate and cost effective.

By letter dated December 2, 1996, (Serial Letter 2422) Toledo Edison informed the NRC that the IPEE summary report would be submitted by December 16, 1996.

Should you have any questions or require additional information, please contact Mr. James L. Freels, Manager - Regulatory Affairs, at (419) 321-8644.

Very truly yours,

John K. Wood

DHL/s

Enclosures

cc: A. B. Beach, Regional Administrator, NRC Region III
A. G. Hansen, DB-1 NRC/NRR Project Manager
S. Stasek, DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board

TRANSMITTAL OF SUMMARY REPORT

OF THE

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS

FOR SEVERE ACCIDENT VULNERABILITIES

FOR

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

IN RESPONSE TO GENERIC LETTER NUMBER 88-20

This letter is submitted in conformance with Section 182a of the Atomic Energy Act of 1954 as amended, and 10CFR50.54(f). Enclosed is Toledo Edition's Summary Report of the Individual Plant Examination of External Events for Severe Accident Vulnerabilities in response to Generic Letter 88-20, Individual Plant Examination of External Events for Severe Accident Vulnerabilities, Supplements 4 and 5.

By: John K. Wood
John K. Wood, Vice President - Nuclear

Sworn to and subscribed before me this

Lori J. Strauss
Notary Public, State of Ohio

LORI J. STRAUSS
Notary Public, State of Ohio
My Commission Expires 3/22/98

INDIVIDUAL PLANT EXAMINATION
OF EXTERNAL EVENTS
FOR THE
DAVIS-BESSE NUCLEAR POWER STATION

submitted in response to

U.S. Nuclear Regulatory Commission

Generic Letter 88-20 Supplement 4

by

The Toledo Edison Company

December 1996

Summary Contents

PART 1
EXECUTIVE SUMMARY

EXECUTIVE SUMMARY

This document reports the results of the Individual Plant Examination of External Events (IPEEE) of the Davis-Besse Nuclear Power Station (DBNPS). The examination has been performed by the Toledo Edison Company in response to Generic Letter 88-20, Supplement 4 and the associated submittal guidance of NUREG-1407 (Ref. 1). Information included in Supplement 5 of Generic Letter 88-20 regarding the scope of the seismic review portion of the IPEEE has also been taken into account.

Previously, Generic Letter 88-20 (including Supplements 1- 3) requested the performance of an Individual Plant Examination (IPE) for internal events. This report was completed by Toledo Edison and submitted in February, 1993. The NRC transmitted its Staff Evaluation Report of the IPE on October 2, 1996 (Ref. 2), which concluded that "the DBNPS IPE has met the intent of GL 88-20."

The IPEEE assessments described in this report address the "external" events identified in Supplement 4 of the Generic Letter. This includes seismic events, internal fires and other external phenomena such as high winds, extreme rainfall, transportation accidents and any other credible external events.

The objectives for the IPEEE, as stated by the NRC, are for licensees:

1. to develop an appreciation of severe accident behavior,
2. to understand the most likely severe accident sequences that could occur at its plant under full power operating conditions,
3. to gain a qualitative understanding of the overall likelihood of core damage and radioactive material release, and
4. if necessary, to reduce the overall likelihood of core damage and radioactive material releases by modifying hardware and procedures that would help prevent or mitigate severe accidents.

For seismic events, a reduced-scope seismic margins analysis was performed in close cooperation with site implementation of the Seismic Qualification Utility Group (SQUG) methodology. In addition to the reduced-scope analysis, additional items (e.g., identification of "bad actor" relays) were evaluated as requested in Supplement 5 of Generic Letter 88-20. Based on these analyses, no vulnerabilities to severe accidents were identified that would be attributable to seismic events, and no actions beyond those already identified by the SQUG program were found to be necessary.

For the internal fire assessment, the EPRI Fire-Induced Vulnerabilities Evaluation (FIVE) methodology was utilized. Supplementary probabilistic risk assessment (PRA) analyses were also employed. While four compartments were identified which did not fall below the FIVE screening criteria for compartment-specific core damage frequency, the bounding values calculated were of insufficient magnitude to constitute a vulnerability. Follow-up actions will include initiating a review of fire response procedures associated with plant areas which did not meet the screening criteria. This is to ensure that -tions taken to satisfy 10 CFR 50 Appendix R compliance (e.g., preemptive tripping of plant equipment) are optimized with respect to maintaining the overall plant risk as low as reasonably achievable. In dition, follow-up action will be taken to address seismic-fire interaction issues associated with two small gas cylinders located in chemistry facilities within the Auxiliary Building.

For the assessment of other applicable external phenomena, a progressive screening approach was utilized as recommended in Section 5 of NUREG-1407. No events such as high winds, floods, or transportation accidents were found to be above the screening criteria utilized. Therefore, no vulnerabilities were identified. Follow-up actions include USAR and administrative program revisions to address on-site chemical hazards, and initiation of a Potential Condition Adverse to Quality Report (PCAQR) to address as-found roof drain conditions.

The IPEEE Generic Letter Supplement listed a variety of Unresolved Safety Issues, Generic Issues and other programs which were to be addressed as part of the external events analysis. Of those applicable to Davis-Besse, USI A-45, Shutdown Decay Heat Removal Requirements, and NUREG/CR-5088, Fire Risk Scoping Study, are considered to be resolved. It should be noted that issues identified as part of USI A-46 are being addressed separately as part of the site SQUG program.

The principal finding of this evaluation is that no vulnerabilities have been identified when evaluating any of the phenomena and plant areas considered. Follow-up actions are being taken, however, to improve overall plant risk with respect to external events where appropriate and cost effective.

REFERENCES FOR PART 1

1. NUREG-1407, Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, June 1991.
2. Toledo Edison Log 4925, Staff Evaluation Report for Generic Letter 88-20, "Individual Plant Examination -- Davis-Besse Nuclear Power Station, Unit No. 1" (TAC No. M74402), October 2, 1996.

PART 2
EXAMINATION DESCRIPTION

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EXAMINATION DESCRIPTION

2.1 Conformance with Generic Letter and Supporting Material

This document reports the results of the Individual Plant Examination of External Events (IPEEE) of the Davis-Besse Nuclear Power Station (DBNPS). The examination has been performed by the Toledo Edison Company in response to Generic Letter 88-20, Supplement 4 and the associated submittal guidance of NUREG-1407 (Ref. 1). Information included in Supplement 5 of Generic Letter 88-20, regarding the scope of the seismic review portion of the IPEEE, has also been taken into account.

Previously, Generic Letter 88-20 (including Supplements 1 - 3) requested the performance of an Individual Plant Examination (IPE) for internal events. This report was completed by Toledo Edison and submitted in February, 1993. The NRC transmitted its Staff Evaluation Report of the IPE on October 2, 1996 (Ref. 2), which concluded that "(1) the IPE is complete with regard to the information requested by GL 88-20 (and associated guidance in NUREG-1335) and (2) the IPE results are reasonable given the DBNPS design, operation, and history. As a result, the staff concludes that DBNPS's IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities. Therefore, that the DBNPS IPE has met the intent of GL 88-20."

The IPEEE assessments described in this report address the "external" events identified in the Generic Letter, namely seismic events, internal fires and other external phenomena such as high winds, floods, extreme rainfall, and any other credible external events.

The objectives for the IPEEE, as stated by the NRC, are for licensees:

1. to develop an appreciation of severe accident behavior,
2. to understand the most likely severe accident sequences that could occur at its plant under full power operating conditions,
3. to gain a qualitative understanding of the overall likelihood of core damage and radioactive material release, and
4. if necessary, to reduce the overall likelihood of core damage and radioactive material releases by modifying hardware and procedures that would help prevent or mitigate severe accidents.

In conjunction with the IPE process, Toledo Edison believes that the above objectives have been satisfied at Davis-Besse.

The IPEEE work was performed in three parts, with each general technical area being performed by the appropriate personnel. The seismic portion was performed primarily by the civil engineering personnel responsible for implementation of the site SQUG program. The internal fire and the "other phenomena" portions were performed primarily by the nuclear engineering personnel, who have the responsibility for maintenance and use of the Davis-Besse PRA. Several of these individuals were involved previously with the performance of the IPE analysis.

The respective independent reviews of these efforts are described separately in each portion of this report. In general, however, it can be stated that an independent peer review was conducted for each portion of the IPEEE to provide further assurance of the technical accuracy of the described analyses and to ensure the validity of the overall conclusions.

2.2 General Methodology

Davis-Besse is a plant which has utilized the SQUG methodology for resolution of Unresolved Safety Issue A-46. As such, as summarized in the Toledo Edison submittal dated November 6, 1995 (Ref. 3), a reduced-scope seismic margins analysis (SMA) was performed for the seismic portion of the IPEEE. As delineated in Ref. 3, this SMA was performed utilizing the SQUG methodology, with the selection of evaluated equipment based on the guidance given in EPRI NP-6041 Rev. 1 (Ref. 4). In addition, the presence of any "bad actor" relays was identified, any flat bottom tanks whose failure could significantly affect plant safety were identified and evaluated, and the "other items" from Generic Letter 88-20 Supplement 5 Attachment 1 were also considered. These evaluations utilized the DBNPS design basis earthquakes.

For the internal fire assessment, the EPRI Fire-Induced Vulnerabilities Evaluation (FIVE) methodology (Ref. 5) was utilized, with supplementary probabilistic risk assessment (PRA) analyses also employed. The FIVE methodology is a screening technique based on conservative assumptions using plant specific data bases for evaluating fire event sequences. The overall objective is to determine the availability of plant systems, components, and cabling to achieve and maintain safe and stable shutdown of the reactor and thereby prevent core damage. The supplementary PRA analyses were largely drawn from the EPRI sponsored Fire PRA Implementation Guide (Ref. 6), which includes insights from SANDIA and EPRI fire research programs. This allowed removal of some of the conservatism included in the original FIVE methodology but preserved the capability to assess the plant safe shutdown ability.

For the assessment of other applicable external phenomena, a progressive screening approach was utilized as recommended in Section 5 of NUREG-1407. As the initial screening determined that no plant-unique events with potential severe accidents vulnerability exist for Davis-Besse, the detailed examination included high winds, floods, and transportation and nearby facility accidents.

2.3 Information Assembly

2.3.1 Plant Familiarization

Davis-Besse is located on the southwestern shoreline of Lake Erie in Ottawa County, Ohio, approximately six miles northeast of the town of Oak Harbor. The site consists of 954 acres, of which approximately 733 acres is marshland leased to the U.S. Government as a national wildlife reserve. The topography is flat with marsh areas bordering the lake and the upland area rising to only 10 to 15 feet above the lake low water datum level. Areas surrounding the station structures have been built up 6 to 14 ·t to an elevation of 584 feet above sea level to provide for flood protection. The station structures are located approximately in the center of the site and are built on a bedrock foundation.

Davis-Besse is a 906 MWe pressurized water reactor (PWR). The nuclear steam supply system was furnished by The Babcock & Wilcox Company. The Bechtel Corporation and its affiliate, The Bechtel Company, provided the architect-engineering services for the station design and construction management services for the construction. The construction permit was granted in March 1971, and the operating license was issued by the NRC in April 1977. Following initial fuel loading and testing, commercial operation began in July 1978.

The reactor containment consists of two structures: a steel containment vessel and a reinforced concrete shield building. The containment vessel is a large, dry, free-standing cylindrical steel pressure vessel with a hemispherical dome and ellipsoidal bottom. It is completely enclosed by the concrete shield building, and there is an annular space between the two. Except for the concrete under the containment vessel, there are no structural ties between the containment vessel and the shield building above the foundation, thus allowing virtually unlimited freedom for differential movement between the two. An emergency ventilation system maintains a negative pressure on this annulus during accident conditions and exhausts through a high efficiency filter network to prevent unfiltered leakage of contaminated air to the environment. The design maximum internal pressure for the steel vessel is 40 psig at a coincident temperature of 264 F.

The containment houses the reactor coolant system, which consists of the reactor vessel, two vertical once-through steam generators (OTSGs), four shaft-sealed reactor coolant circulating pumps, an electrically heated pressurizer, and interconnecting piping. The system is arranged into two transport loops, each containing two circulating pumps and one steam generator. The vertical OTSGs are raised above the core vessel nozzles to promote natural circulation and to provide an inventory of water to help cover the fuel during a loss-of-coolant accident (LOCA). The reactor coolant system is designed to contain and circulate reactor coolant at pressures and flows necessary to transfer the heat generated in the core to the secondary fluid in the steam generators. In addition to serving as a heat transport medium, the coolant also serves as a neutron moderator and reflector and as a solvent for the soluble boron used for chemical shim reactivity control. The secondary fluid is completely separate from the reactor coolant and is used to transfer energy from the steam generators to the main turbine generator and auxiliary loads.

Heat transfer from the primary coolant, via the steam generators, to the feedwater systems is the preferred means of decay heat removal following a reactor trip. This can be accomplished by either the main feedwater system or the auxiliary feedwater system. In the event the main feedwater system is unavailable, the turbine-driven auxiliary feedwater pumps are automatically initiated by the steam and feedwater rupture control system. As an additional backup, Control Room operators have the capability to start a motor-driven feed pump to supply feedwater in the event both the main and turbine-driven auxiliary feedwater pumps are not available. In addition, makeup/high pressure injection (HPI) cooling is available as yet another means of cooling the core if the steam generators are unavailable. The decay heat removal pumps provide normal cooldown of the primary at lower pressures and temperatures by transferring heat from the primary to the component cooling water system. The decay heat removal system can also provide low pressure injection from the borated water storage tank during a large LOCA. Heat transferred to the component cooling water system is then transferred to the service water system, and then to the ultimate heat sink (Lake Erie). Two-train independence for each of these systems prevents a single failure from disrupting functional operation.

Engineered safety features are provided to protect the fuel cladding, ensure containment vessel integrity, and reduce the driving force for containment leakage. Emergency injection of coolant to the reactor coolant system satisfies the first function, and cooling of the containment vessel atmosphere satisfies the latter two functions. The emergency core cooling system includes the core flood tanks, high pressure coolant injection and low pressure coolant injection. Containment spray and containment air coolers are responsible for removing heat and reducing pressure within containment during an accident. Each of these systems consists of two independent trains that are controlled automatically by the safety features actuation system (SFAS). SFAS continuously evaluates key parameters, and would sequentially initiate and coordinate appropriate equipment if a LOCA were to occur.

Plant equipment is normally supplied with ac power from an auxiliary transformer connected to the plant's main generator. Two start-up transformers, supplied from different 345kV switchyard sections, serve as the reserve power source for the station auxiliaries in the event power from the main generator is not available. If the normal and reserve power supplies were both unavailable, two redundant emergency diesel generators are provided as on-site standby power sources. Each emergency diesel generator is connected to an essential 4kV bus and is capable of supplying all essential loads for one train. A third standby source, the station blackout diesel generator, would be available to supply power in the event that the normal, reserve, and emergency power supplies failed. The station blackout generator can be manually started from the Control Room or at a local control station, and it has its own auxiliaries to provide independence from the normal plant systems.

2.3.2 IPEEE-Specific Plant Information

Specific plant information as well as general references utilized in the Davis-Besse IPEEE are summarized in each respective Section. Where needed, simple figures denoting plant details are included. The models utilized in support of the probabilistic quantifications are the same as those used in the IPE study, except as noted to more accurately model fire effects. As the Staff Evaluation Report for the IPE was not received until just prior to completion of this study, major revisions to PRA models and assumptions were not performed. For the work supporting each area of analysis, plant walk downs were conducted to ensure that plant details were accurately assessed.

Where coordination was needed between different portions of the IPEEE effort, such as the assessment of seismically induced fires, the respective analysts were assembled to plan the best approach and avoid duplication of effort. At Davis-Besse, the entire nuclear staff is located on-site, which greatly facilitates this process. This fact also facilitates the ability to perform thorough plant walk downs.

2.4 Submittal of Specific Safety Features and Potential Plant Improvements

As noted previously and discussed in detail in subsequent sections of this report, no vulnerabilities have been identified with respect to potential severe accidents.

For the seismic event analysis, no actions beyond those previously identified for the SQUG program were identified. This is considered to be a safety feature of the plant which demonstrates the overall seismic adequacy of the Davis-Besse design.

With respect to the internal fire analysis, four plant compartments were identified which had calculated bounding core damage frequency (CDF) values above the screening criteria of 1E-6/yr. For each case the bounding CDF was found to lie between 1E-6/yr and 1E-5/yr. Given these results, the Severe Accident Issue Closure Guidelines (Ref. 9) were reviewed to ascertain the relative importance of these estimations. Section 4.4 and associated Table 1 of the Closure Guidelines indicate that for fire compartments that fall in this CDF range, the licensee should ensure that severe accident management guidelines will be in place with the emphasis on prevention/mitigation of core damage or vessel failure, and containment failure. In addition to the in-place emergency operating procedures which center on prevention of core damage, Davis-Besse has committed to having Severe Accident Management Guidelines in place by December 31, 1997 (Ref. 10). Therefore, as delineated in the Reference 9 closure guidelines, no further actions are deemed necessary. However, to further reduce the plant risk to postulated internal fire events, Davis-Besse will review fire response procedures associated with plant areas which did not screen to ensure that specified actions are optimized with respect to maintaining the overall plant risk as low as sonably achievable. In addition, as a result of the seismic-fire interaction analysis, it was determined that additional action is required to address concerns regarding two small compressed flammable gas cylinders located in chemistry facilities in the Auxiliary Building.

For the analysis of high winds, floods and other external events, all phenomena were found to screen below the applicable screening criteria. Again, as with the seismic portion of the analysis, this demonstrates the overall adequacy of the Davis-Besse siting and design. Several actions were taken, however, to further reduce the plant risk to postulated significant external events:

1. Potential Condition Adverse to Quality Report (PCAQR) 96-0186 was initiated to address the issue of on-site hazards from hazardous material.
2. USAR Change Notice 96-58 was initiated to revise the description of the hazards from chemicals stored or transported on-site.
3. The controlled materials program was revised so that new materials approved for use on-site will be evaluated for Control Room habitability.
4. PCAQR 96-0956 was initiated to document plugged roof drains and standing water on the 643 foot elevation of the Auxiliary Building roof.
5. PCAQR 96-1841 was initiated to address seismic-fire interaction issues associated with the location and mounting of two compressed flammable gas cylinders.

5 IPEEE Team and Peer Review

The IPEEE team was a multi-disciplinary team drawn mostly from the site engineering staff. For the non-seismic portions of the analysis, the lead organization was the Nuclear Engineering Section of the Design Basis Engineering Department. Personnel from this Section involved with the IPEEE had over 60 years of cumulative nuclear power plant experience in thermal-hydraulic analyses, PRA applications, and Operations. One member of the Nuclear Engineering team was previously SRO qualified at Davis-Besse. For the seismic portion of the analysis, the Design Engineering - Mechanical/Structural Section was the lead organization. Overall, site organizations with major responsibilities included the following:

Mechanical/Structural Design Engineering	Seismic IPEEE, Fire/Seismic Interactions
Nuclear Engineering	Fire IPEEE, "Other" IPEEE, Seismic IPEEE
Plant Engineering	Fire IPEEE, Fire/Seismic Interactions

Where needed, additional information was gathered from the applicable site organization, including Design Engineering - Electrical/I&C (cable identification and circuit functions, etc.), Operations (procedural implementation, operator actions and timing, etc.), and Radiation Protection (transient combustible characterizations, etc.). As such, it can be readily seen that Toledo Edison personnel were involved in all aspects of the IPEEE effort.

In addition to the above described plant IPEEE team, SAROS, Inc. and NUS (now SCIENTECH) provided consulting services for the non-seismic portions of this analysis. SAROS was closely involved in development of the Davis-Besse IPE analyses, and provided PRA quantification support. The principal reviewer from SAROS had over 15 years of probabilistic risk experience, including previous IPEEE work. NUS performed an independent peer review of the non-seismic portions of the IPEEE (Ref. 7, 8). The principal reviewer from NUS had over 15 years of probabilistic risk experience, including work on previous IPE and IPEEE submittals.

The staff performing the seismic portion of the analysis was augmented by EQE International. Personnel from EQE aided in the seismic walk downs done during the refueling outage. EQE International is recognized as a leader in seismic evaluations in the nuclear industry. All of the EQE engineers utilized were SQUG and IPEEE trained and qualified.

Results of the peer review for the internal fire portion of the analysis indicated that "The analysis appears to be generally sound, well documented and correctly applies the FIVE methodology. Where the analysis deviates from strict adherence to FIVE ... the rationale appears to be reasonable and consistent with other studies." Specific peer review recommendations were made in the areas of fire induced LOCAs, combustible concentrations at compartment interfaces and the treatment of transient fire frequencies. In addition, minor comments were noted with respect to the estimation of fixed source fire frequencies. All comments were evaluated and satisfactorily dispositioned.

Results of the peer review for the high winds, floods and other external phenomena portion licated that the analysis "Generally looks to be complete .... well organized and satisfies the requirements of the IPEEE." Specific comments were made with respect to clarification of any plant-unique external events, and roof equipment and loadings. All comments were evaluated and satisfactorily dispositioned.

Additional detail regarding the qualifications of personnel involved performing the seismic margins evaluations and consultants utilized for the seismic portion of the IPEEE are contained in Section 3. Peer review efforts for the seismic portion are also delineated in Section 3. It should be noted that given the close relation between site SQUG effort and the IPEEE-seismic effort, details of personnel and the peer review were prepared in a format similar to that provided for the Davis-Besse SQUG submittal.

2.6 NRC Issue Closure

The IPEEE Generic Letter Supplement listed a variety of Unresolved Safety Issues, Generic Issues and other programs which were to be addressed as part of the external events analysis. Of those applicable to Davis-Besse the following are considered to be resolved:

USI A-45, Shutdown Decay Heat Removal Requirements - As noted previously, no specific vulnerabilities have been identified as part of this report, which included evaluation of the ability to remove reactor decay heat. Previously identified issues associated with the USI A-46 program are being addressed separately as part of the site SQUG program. As such, consistent with conclusions of the IPE study for internal events, USI A-45 is also considered to be resolved for external events.

NUREG/CR-5088, Fire Risk Scoping Study - This issue was specifically addressed as part of the internal fire portion of this report. As noted, the identified issues have been adequately addressed at Davis-Besse, and therefore, this issue is considered resolved.

REFERENCES FOR PART 2

1. NUREG-1407, Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, June 1991.
2. Toledo Edison Log 4925, Staff Evaluation Report for Generic Letter 88-20, "Individual Plant Examination -- Davis-Besse Nuclear Power Station, Unit No. 1" (TAC No. M74402), October 2, 1996.
3. Toledo Edison Serial Number 2341, Response to Generic Letter 88-20, Supplement 5, Individual Plant Examination of External Events for Severe Accident Vulnerabilities, November 6, 1995.
4. EPRI NP-6041-SL Rev. 1, A Methodology for Assessment of Nuclear Power Plant Seismic Margins, August 1991.
5. Fire-Induced Vulnerability Evaluation, EPRI, EPRI TR-100370, April 1992.
6. EPRI TR-105928, Fire PRA Implementation Guide December, 1995.
7. NUS (Scientech) Peer Review, Internal Fire Analysis, R&R-PG-96-510, August 1996.
8. NUS (Scientech) Peer Review, High Winds, Floods and Other External Events, R&R-PG-96-705, December 1996.
9. NEI 91-04 Rev. 1, Severe Accident Issue Closure Guidelines, December 1994.
10. Toledo Edison Serial Number 2275, Implementation of Industry Policy on Severe Accident Management, February 22, 1995.

PART 3
SEISMIC ANALYSIS

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SEISMIC ANALYSIS

Methodology Section

This section summarizes Davis-Besse Nuclear Power Station's (DBNPS) results of the resolution to the seismic portion of the Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-20 entitled "Individual Plant Examination for External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR50.54 (f)".

Davis-Besse was originally categorized as a "focused-scope" site by the NRC in NUREG-1407. However, in a letter to the NRC on August 25, 1994 (Serial Number 2242), Toledo Edison informed the NRC Staff of its plan and justification for revising the DBNPS commitment for the seismic IPEEE program from a "focused-scope" Seismic Margin Assessment (SMA) program to a "reduced-scope" SMA program, using the Seismic Qualification Utility Group (SQUG) methodology, (Ref. 1). Supplement 5 to GL 88-20 provided some relaxation to the original scope of the IPEEE seismic program. On November 6, 1995, Toledo Edison reiterated its position to the NRC Staff on conducting a "reduced-scope" SMA program (Serial Number 2341, Ref. 2). In addition, as requested in GL 88-20, Supp. 5, Toledo Edison committed to identify "bad actor" relays, identify and evaluate flat bottom tanks whose failure could significantly affect plant safety, and consider "other items" as identified in Attachment 1 to GL 88-20, Supp. 5.

Based on the above, DBNPS performed a reduced scope evaluation using the SQUG methodology seismic adequacy determination of identified equipment for the resolution of IPEEE seismic.

3.1 Seismic Margin Method

3.1.1 Review of Plant Information, Screening, and Walkdown

3.1.1.1 General Plant Description

The station site is located on the southwestern shore of Lake Erie in Ottawa County, Ohio and consists of 954 acres of which approximately 733 acres is marshland which is leased to the U.S. Government as a wildlife refuge. The topography of the site and vicinity is flat with marsh areas bordering the lake and the upland areas rising to only 10 to 15 feet above the lake low water datum level in the general surrounding area. The site itself varies in elevation from marsh bottom, below lake level, to approximately six feet above lake level.

The site areas surrounding the station structures have been built up from 6 to 14 feet above the existing grade elevation to an elevation of 584 feet above sea level, International Great Lakes Datum (IGLD), or 15.4 feet above Lake Erie Low Water Datum of 568.6 feet IGLD. This provides flood protection from the maximum credible water level conditions of Lake Erie.

1.1.2 Site Geology

The site region is located in the Lake Plains subprovince of the Central Lowland physiographic province. The Lake Plains subprovince is nearly flat and has poor surface drainage characteristics. The surficial soils are glacial deposits. Local sedimentary bedrock exposures have a very slight dip.

The bedrock formation is the Tymochtee formation which consists of argillaceous dolomite with interbedded gypsum, anhydrite, and shale strata. The argillaceous dolomite can be divided into two major units: a massive dolomite and a bedded dolomite. A description of each dolomitic rock unit and representative static and dynamic properties follows.

The massive dolomite occurs in a 8-ft to 10 ft. thick stratum, the top of which is located approximately 10 ft below the bedrock surface. The massive dolomite is medium hard to hard, buff to gray, and argillaceous.

The bedded dolomite occurs above and below the massive dolomite unit. It is medium hard, gray to buff, and argillaceous with frequent laminae of gypsum, anhydrite, and shale.

Foundations for Class I station structures consist of mat or strip footings bearing on bedrock, till deposit, or compacted granular fill and pier footings socketed into bedrock.

The factor of safety of mat strip footing foundations for Class I structures against a bearing capacity failure (expressed as a ratio between the ultimate bearing capacity of material beneath the footing and the maximum contact stress beneath the footing) is greater than 5. Total settlement of Class I structures founded on bedrock will be less than 1/10 in., and total settlement of Class I structures founded on till deposit and granular fill will be less than 1/4 in. Settlement of structures will be elastic within the range of footing contact stresses anticipated. Consequently, settlement will occur upon application of footing stresses and no long-term settlement of structures is expected.

The bedrock beneath foundations in the station area was free of significant solutions activity and thus was considered to have maximum design net bearing pressure of 100 k/sq. ft. A value of 600k/sq. ft was established as a conservative value for the ultimate bearing capacity of the bedrock free of significant solution activity.

A portion of the Auxiliary Building (Area 6) is supported on socketed pier footings foundations. The maximum load expected on these footings is 1700 kips. The ultimate load that can be supported by the bedrock socket is 4900 kips. Settlement of individual piers, at the 1700 kip load, is expected to be less than 1/4 inch.

3.1.1.3 Major Structures

The major structures associated with the DBNPS include the Auxiliary Building, Containment Structures, Intake Structure and the Turbine Building.

3.1.1.3.1 Auxiliary Building

The Auxiliary Building is an L-shaped reinforced concrete structure that partially surrounds the Shield Building and is located next to the Turbine Building. It is a five story building with two levels below grade. The Auxiliary Building is subdivided into three separate structures. The Auxiliary Building houses the radwaste system, fuel storage and handling, auxiliary nuclear equipment, Control Room, switchgears, diesel generators and other operation facilities.

The northeast portion of the Auxiliary Building (Area 6) is 130 ft by 155 ft in plan and is 70 ft in height above the first floor slab. The first floor is at El. 585 and is supported on grade beams connected to 3 ft. diameter pier footings. Pier footings extend through compacted granular backfill beneath the floor slab and are socketed a minimum of 2.5 ft into bedrock.

The southeast portion of the Auxiliary Building (Area 7) is 130 ft by 140 ft in plan and is 120 ft in height above the foundation level. It is supported on a 3 ft thick mat foundation that bears on bedrock at El. 542.

The southwest portion of the Auxiliary Building (Area 8) is 140 ft by 140 ft in plan is 120 ft in height above foundation level. It is supported on a 2 ft thick mat foundation. The outside walls are supported on 3 ft thick strip footings. The bottom of the mat is at El. 563 and is underlain by an approximately 2 ft thick layer of concrete backfill over bedrock. For these conditions, the mat can be considered to be supported on bedrock. The bottom of the strip footings bear on the bedrock at El. 557.

3.1.1.3.2 Containment Structures

Nuclear Steam Supply System

The nuclear steam supply system (NSSS) consist of the reactor vessel, two vertical once-through steam generators, four shaft-sealed coolant circulation pumps, electrically heated pressurizer, and interconnecting piping. The system is arranged as two heat transport loops, each with two circulating pumps and one steam generator. The NSSS is designed by Babcock and Wilcox Company. The NSSS is designed for a warranted power output of 2772 MWt, with a corresponding gross electrical output of approximately 906 MWe.

The containment for the NSSS at DBNPS consists of three basic structures: A steel Containment Vessel, a reinforced concrete Shield Building, and the internal structures.

The Containment Vessel

The Containment Vessel is a cylindrical steel pressure vessel with hemispherical dome and ellipsoidal bottom which houses the reactor vessel, reactor coolant piping, pressurizer, pressurizer quench tank and coolers, reactor coolant pumps, steam generators, core flood tanks, letdown coolers, and normal ventilating system. It is completely enclosed by a reinforced concrete Shield Building having a cylindrical shape with a shallow dome roof. An annular space is provided between the wall of the Containment Vessel and the Shield Building, and clearance is also provided between the Containment Vessel and the dome of the Shield Building. The Containment Vessel and Shield Building are supported on a concrete ındation founded on a firm rock structure. With the exception of the concrete under the Containment Vessel there are no structural ties between the Containment Vessel and the Shield Building above the foundation slab. Above this there is unlimited freedom of differential movement between the Containment Vessel and the Shield Building.

The Containment Vessel was constructed in a two-stage operation and in a manner that conformed to the ASME Boiler and Pressure Vessel Code, Article 14, N-1411. The vessel inside diameter is 130 feet and the net free volume is approximately 2,834,000 ft^3. The cylindrical shell and bottom head thickness, exclusive of reinforced areas, is 1 1/2" with a dome thickness of 13/16". The 180 ton polar crane is supported from the cylindrical vessel shell by a 14'-6 1/2" deep by 5'-11" wide circular crane girder. Access to the containment is provided by an equipment hatch, a personnel air lock and an emergency air lock.

The Shield Building protects the containment vessel from external missiles. Protection from internal missiles is provided by the primary and secondary shield walls of the containment internal structure.

The Shield Building

The Shield Building is a reinforced concrete structure of right cylinder configuration with a shallow dome roof. An annular space is provided between the steel containment vessel and the interior face of the concrete shield building of approximately 4.5 feet to permit construction, operations, and periodic visual inspection of the steel containment vessel.

The Shield Building completely encloses the Containment Vessel, the personnel access openings, the equipment hatch, and that portion of all penetrations that are associated with primary containment. The design of the Shield Building provides for 1) biological shielding, 2) controlled release of the annulus atmosphere under accident condition, and 3) environmental protection of the Containment Vessel.

The Shield Building has a height of 279.5 ft. measured from the top of the foundation ring to the top of the dome and a diameter of 144 ft. The thickness of the wall and the dome are approximately 2.5 ft. and 2 ft. respectively. The Containment Building is founded on a bowl shaped mat foundation bearing on bedrock. The minimum thickness of the mat is 4.5 ft.

The Containment Vessel Internal Structures

The containment internal structures are comprised of the reactor cavity, the primary shield wall, the secondary shield wall, the refueling pool, the operating floors miscellaneous equipment supports, stairs, and service missile shields. The primary coolant system, including the reactor, steam generators, pressurizer, and reactor coolant pumps, is supported by these structures. Shield walls and floors are constructed of reinforced concrete. Structural steel frames and columns support the floors and transmit loads to the foundations.

Steam generator stability is ensured by a system of upper and lower supports which are designed LOCA loads concurrent with maximum seismic loads while allowing unrestricted movement due to thermal expansion of the reactor coolant system.

The pressurizer support, which is anchored on the steam generator compartment wall, is designed and analyzed as a steel truss.

The reactor vessel is supported by welded plate girders located on each of the four cold legs. Each support is composed of two steel plate girders. These girders are designed utilizing all possible combinations of dead, live, thermal and seismic loading.

3.1.1.3.3 Intake Structure

The Intake Structure is a reinforced concrete structure 58 ft. by 62 ft. in plan and is 55 ft. in height above foundation level. The Intake Structure is founded on a 3 ft thick mat foundation bearing on bedrock at El. 543.

The following major items are located in the Intake Structure: service water pumps, cooling tower water makeup pumps, diesel driven fire water pump, water treatment makeup pumps, traveling screens, and the backup service water pumps.

3.1.1.3.4 Borated Water Storage Tank

The Borated Water Storage Tank is 47 ft. in diameter, 50 ft in height above foundation level and has a capacity of 550,000 gallons. It is located west of the Containment Building. The tank consists of a 1/4 in thick steel shell and roof. The tank shell is supported on a reinforced concrete mat foundation approximately 6 ft. deep and 49 ft. in diameter, bearing on Class I granular backfill at El. 585. The class I backfill extends from El. 585 to the top of the bedrock at approximately El. 560. 48 anchor bolts 2 1/2 inches in diameter are used to fasten the tank to the foundation. These bolts have an embedment depth of 55 inches.

3.1.1.3.5 Intake Forebay Canal Dike

The Intake forebay canal dike impounds the station's ultimate heat sink. The dike consists of compacted glaciolacustrine and till deposits obtained from on-site borrow areas.

3.1.1.4 Screening Criteria

DBNPS performed a reduced-scope review by following the guidance identified in EPRI NP-6041-SL for the selection of equipment to be evaluated and the SQUG's Generic Implementation Procedure (GIP) to assess the seismic adequacy of equipment. Components having relatively high seismic capacities were determined to need no further review when using the methodology for seismic margin assessment per Tables 2-3 and 2-4 of EPRI NP-6041. This screening was performed for a seismic margin earthquake having a peak 5 percent damped peak spectral acceleration of <0.8 g. This screening level is considered to be nservative for DBNPS.

Prescreening is used to exclude from further review those items that are determined to be smically rugged based on experience and judgment. Prescreening allows more effort to be expended on those items that may be a concern with seismic ruggedness.

The IPEEE seismic walkdown program was combined with the USI A-46 (SQUG) walkdown program since many of the items were common to both sets of lists. The SQUG methodology, as defined by the GIP was then used to determine the seismic adequacy of the equipment. The results of these walkdowns were documented on the GIP's Screening Evaluation Work Sheets (SEWS). By using the SQUG methodology, the objectives of the IPEEE and the requirements of GL 87-02 were achieved without duplication of effort.

The following is the listing of the structures identified in EPRI NP-6041 Table 2-3 along with the justification for screening these items from the program.

1. Concrete Containment - This is not applicable to DBNPS
2. Free standing steel containment - The base mat is an integral part of the pressure boundary, therefore, no evaluation is required.
3. Containment internal structures - DBNPS has a pga of 0.15g which is > 0.10g, therefore, no evaluation is required.
4. Shear walls, footings, and containment shield walls - See item 3 above.
5. Diaphragms - See item 3 above.
6. Category I concrete frame structures - See item 3 above. Seismic Category I buildings at DBNPS include the Auxiliary Building, Containment, Intake Structure, and the Shield Building.
7. Category I Steel frame structures - See items 3 and 6 above.
8. Masonry Walls - Since DBNPS is performing a Reduced Scope evaluation with a Review Level Earthquake of 0.15g, the evaluation performed for I.E. Bulletin 80-11 is acceptable.
9. Control Room Ceiling - This will be included in the program
10. Impact between structures - Not required.
11. Category II Structures with Safety-Related equipment or with potential to fail Category I structures - There are no IPEEE Seismic Review Safe Shutdown Equipment List (SSEL) equipment located in a Category II structure. In addition, Section 3.8.1.1.6 of the DBNPS USAR states "there is no significant influence of any Class II structures on the Class I structures." Therefore, further evaluation of non-seismic Category I structure is not required.
12. Dams, levees, dikes - DBNPS does not have any dams or levees. A seismic Class I dike does impound the ultimate heat sink which is adjacent to the intake structure. Based on the above, no further evaluation is required.
13. Soil failure modes - Evaluation is not required for reduced scope plants.

The GIP methodology identifies several additional reviews to be performed before the seismic equacy of the equipment could be established. The following identifies those classes of equipment where additional evaluations were performed to meet the GIP requirements but were not required by the SMA methodology.

EPRI NP-6041 Table 2.4 requires no specific evaluation for active valves. All active valves included on the DBNPS Seismic Review Safe Shutdown Equipment List (SSEL) (Appendix A) were screened during the walkdown using the GIP requirements to ensure they are within the experience database as described in the GIP.

The SMA methodology specifies that for fans, air handlers, chillers, and air compressors that only units supported on vibration isolators require an evaluation of anchorage. The anchorage of all such equipment included in the DBNPS Seismic Review SSEL were evaluated against the GIP criteria, whether or not the equipment was on vibration isolators.

The SMA methodology specifies that for electrical power distribution panels, cabinets, switchgear, motor control centers (MCC) and instrumentation and control panels and racks require a limited walkdown to verify that such equipment is securely anchored to the floor or wall and that the instruments are properly attached to the cabinets. A full walkdown of the anchorage of all electrical equipment on the SSEL was performed, including appropriate anchorage calculations, to satisfy the GIP requirements.

The anchorage calculations used to determine the adequacy of the anchorage for the screening of uipment followed the GIP methodology. If the Seismic Review Team (SRT) was able to judge the minimum lowest natural frequency of the component, it was noted. To determine the seismic demand, the maximum spectral acceleration at that frequency was used, otherwise, the peak spectral acceleration values were used.

Approximate equipment weights and centers of gravity were based on Appendix C of the GIP unless specific equipment data was available. The anchorage capacity was based on the EPRI NP-5228 document "Seismic Verification of Nuclear Plant Equipment Anchorage."

Because DBNPS performed a reduced-scope evaluation and is a USI A-46 plant, the evaluation of relay chatter is not required. However, since "Bad Actor" relays were identified during the USI A-46 review, an USI A-46 relay evaluation for Bad Actors was performed for all equipment requiring a relay evaluation as identified on the IPEEE Seismic Review SSEL. The results of this review are identified in Section 3.1.4.2

3.1.1.5 Seismic Walkdown

NUREG 1407 indicates "well conducted, detailed walkdowns have been demonstrated to be the most important tool for identifying seismic weak links whose correction is highly cost effective."

This section identifies the Seismic Capability Engineers (SCEs) and the approach used to perform e walkdown.

3.1.1.5.1 Seismic Capability Engineers

Two major elements of the Reduced-Scope evaluation are the plant walkdown and the qualification of the engineers performing the walkdown. The SCEs are the individuals responsible for implementing the seismic evaluations for the equipment on the Seismic Review SSEL. These SCEs have acquired many years of formal and practical experience in the field of structural and seismic design and analysis. These individuals easily exceed the minimum qualification requirements established for both USI A-46 and IPEEE. All SCEs have successfully completed the SQUG Walkdown Screening and Seismic Evaluation Training Course and the Add-On Seismic IPE Training Course.

The following is a list of the on-site Toledo Edison individuals that comprised the seismic walkdown teams:

> Jagdish C. Arora PE, Richard N. Bair PE, Thomas E. Dabrowiak PE, Jon G. Hook PE, Steven J. Osting PE, and Scott R. Saunders.

The following is a list of individuals supplied by EQE International that were used to supplement the in-house teams at various times during the 8th and 9th Refueling Outages:

> James R. Disser, John O. Dizon PE, Steven J. Eder PE, Gayle S. Johnson PE, Omar Khemici PE, and Basilio Sumodobila PE.

The SCEs were rotated amongst the different Seismic Review Teams (SRT) to take advantage of ːir individual expertise. Many items were walked down with as many as 4 or 5 SCEs. This was done early on in the walkdown phase to acclimate the team members as well as providing for good team interaction. A brief summary of the qualification of each SCE is given below.

Jagdish C. Arora

Mr. Arora is a senior engineer in the Civil Structural Design Unit at Toledo Edison. He has a B.S. degree in civil engineering from the Indian Institute of Technology and a M.S. degree in civil engineering from the University of Minnesota. Mr. Arora has more than 30 years of experience in structural analysis and design of structures for electric utilities. Mr. Arora has extensive experience in the design and seismic analysis of civil structures and subsystems associated with nuclear power. He is a registered Professional Engineer in the States of Michigan and Minnesota.

Richard N. Bair

Mr. Bair is a senior engineer in the Civil Structural Design Unit at Toledo Edison. He has a B.S. degree in civil engineering from the Michigan Technological University with graduate courses from the University of Michigan. Mr. Bair has over 20 years of experience in structural analysis, design, and construction experience for nuclear power stations. He has extensive knowledge of cable tray/ conduit and HVAC support design. He is a registered Professional Engineer in the State of Michigan.

Thomas Dabrowiak

Mr. Dabrowiak is a senior engineer in the Civil Structural Design Unit at Toledo Edison. He has a B.S. degree in civil engineering from Purdue University. Mr. Dabrowiak has over 20 years of experience in structural analysis, and design of structures as well as miscellaneous structural items. He has experience in performing dynamic analysis of vibrating equipment. He is a registered Professional Engineer in the State of California.

James R. Disser

Mr. Disser is a project engineer with EQE International. He has a B.S. degree in civil engineering from University of Michigan. Mr. Disser has over 12 years experience in design, analysis, and project management. He has experience in the design and construction of piping, cable tray, and conduit, HVAC and their associated supports. He has performed USI A-46 walkdowns at two other nuclear plant sites.

John O. Dizon

Mr. Dizon is a group manager/principal engineer with EQE International. He has a B.S. in civil engineering from Mapua Institute of Technology (Philippines), and a M.S. in structural engineering from Stanford University. Mr. Dizon has over 15 years of experience in seismic analyses and design assessments of primary structures and piping systems, and seismic qualification of mechanical and electrical systems associated with the nuclear industry. He has performed or coordinated the USI A-46 program at several nuclear facilities. In addition Mr. Dizon participated in the seismic evaluation of the High Flux Isotope Reactor at Oak Ridge National Laboratory. He has co-authored numerous publications on the design and analysis of earthquake engineering. Mr. Dizon is a registered Professional Engineer in the Philippines and in the State of California.

Stephen J. Eder

Mr. Eder is Vice President and Regional Manager for EQE Engineering Consultants. He has a B.S. degree in civil engineering from Clarkson College of Technology and a M.S. in structural engineering and structural mechanics from University of California, Berkeley. Mr. Eder experience includes structural dynamic analyses, seismic evaluation and margin assessments, post-earthquake reconnaissance studies, and shake table and other dynamic tests and qualification. He is a Subject Matter expert for the SQUG methodology and developed the SQUG raceway system seismic evaluation guidelines using earthquake experience data, test results and fatigue analysis. He has performed seismic adequacy walkdowns at dozens of nuclear facilities and has authored numerous technical papers on the design and analysis of earthquake engineering. He is a registered Professional Engineer in the State of California.

Jon G. Hook

Mr. Hook is a senior engineer in the Civil Structural Design Unit at Toledo Edison. He has a B.S. degree in civil engineering from Michigan Technological University with graduate courses from the University of Michigan. Mr. Hook has over 20 years experience in nuclear design and construction. Mr. Hook has extensive experience performing seismic analysis and well as review and approval of seismic qualification of equipment. He is Toledo Edison's representative to the Seismic Qualification Utility Group (SQUG). He is a registered Professional Engineer in the State of Michigan.

Gayle S. Johnson

Mr. Johnson is a principal engineer with EQE International. He has a B.S. degree in civil engineering from University of Minnesota and a M.S. in civil engineering from the University of California, Berkeley. Mr. Johnson has over ten years of experience in seismic evaluation, seismic criteria development, structural design, linear and nonlinear analysis, and software development. He has participated in and managed several seismic evaluation of industrial facilities, and seismic safety evaluations of nuclear facilities. He is a registered Professional Engineer in the State of California.

Omar Khemici

Dr. Khemici is a principal engineer, with EQE International. He has a B.S. degree in civil engineering from Polytechnic Institute of Algiers (Algeria), a M.S. degree in civil engineering from Stanford University, and a Ph.D. degree in civil engineering from Stanford University. Dr. Khemici has over eight years of extensive professional experience in earthquake engineering. He is registered Professional Engineer in the State of California.

Steven J. Osting

Mr. Osting is a senior engineer in the Civil Structural Design Unit at Toledo Edison. He has a B.S. in Mechanical Engineering Technology from the University of Dayton. Mr. Osting has over 16 years of experience in an operating nuclear facility. His areas of expertise include the design and analysis of process piping and HVAC duct and associated supports. He is a registered Professional Engineer in the State of Ohio.

Scott R. Saunders

Mr. Saunders is a senior engineer in the Civil Structural Design Unit at Toledo Edison. He has a B.S. degree in civil engineering from Ohio State University. Mr. Saunders has over 10 years of experience in the operating nuclear facility. His area of expertise include the design and analysis of process piping, cable tray and conduits and associated supports as well as the design of miscellaneous structural items and associated supports.

Basilio N. Sumodobila, Jr.

Mr. Sumodobila is a principal Engineer with EQE International. He has a B.S. degree in civil engineering from Mapua Institute of Technology (Philippines) and graduate courses from the University of California, Berkeley. Mr. Sumodobila has over 19 years of experience in seismic evaluation, structural dynamic analysis, seismic analysis, structural design, linear and nonlinear analysis and finite element software development. He has participated in seismic walkdown at numerous nuclear facilities. He is a registered Professional Engineer in the Philippines and in the State of California.

1.1.5.2 Walkdown Procedure

The key element of the seismic evaluation is a careful and thorough walkdown by experienced engineers. Prior to the walkdowns, a file was established for each component being walked down. This data was then reviewed by the SCEs before the component was evaluated. This gave the SCEs insight into the design, construction, mounting, and other pertinent information needed to evaluate the component.

The IPEEE Seismic Review SSEL was combined with the USI A-46 (SQUG) seismic walkdown list to form one walkdown list that encompasses both programs. This list was the basis for the items being evaluated. Of the 348 components on the IPEEE Seismic Review SSEL walkdown list, 290 were also on the SQUG walkdown list, exclusive of containment penetrations. Each item on the list was inspected, evaluated, and a walkdown data sheet was completed, all in accordance with the GIP requirements. The walkdown concentrated on the following areas: seismic capacity, screening caveats, anchorage, and seismic spatial interaction. The SEWS found in the GIP were used for documenting the results of the walkdown instead of the forms in the SMA methodology. The GIP forms were chosen because they were more efficient in enveloping both the SMA and SQUG programs and were preferred by the SCEs. If the components did not fall within the SQUG 20 Classes of equipment, they were then classified as outliers.

Equipment was evaluated by using the seismic capacity Bounding Spectrum, as shown in Figure 4-2 of the GIP when the equipment was below approximately 40 feet above grade and has the lowest natural frequency of approximately 8 Hz or higher. If the equipment was located above approximately 40 feet ɔve grade or had a natural frequency below 8 Hz, then the capacity was established by using either 1.5 x Bounding Spectrum or existing seismic qualification reports. Seismic demand was established by using the Ground Response Spectrum or by using the appropriate in-structure SSE response spectrum. This is consistent with Section 4.2 of the GIP.

EPRI NP-6041 indicates "The most common failure mode for equipment is anchorage failures. A margins review should concentrate heavily on the review of anchorage failure modes." Equipment anchorage was inspected and evaluated in more detail than was required by the SMA methodology in order to meet the more exacting requirements of the GIP. The anchorage section of the SEWS addresses the requirements of both programs. This included, but is not limited to: adequacy of anchorage check, expansion anchor bolt tightness check, bolt spacing and edge distance check, concrete soundness check, base stiffness and load path check. All accessible anchorages were checked in this manner.

Calculations were performed as required during the walkdown to assess the adequacy of equipment anchorage. These calculations were generally performed in a manner that would quickly show whether the seismic capacity was greater than the seismic demand for the equipment anchorage. These calculations were not intended to be rigorous, but were made solely to provide a basis for the SCEs to judge whether the anchorage was adequate.

The SMA methodology requires a "100-percent walk by" of all success path components which are reasonably accessible. This walk-by is not intended to mean a complete inspection of each component, nor es it mean requiring an electrician or other technician to de-energize and open cabinets or panels for detailed inspection of all components. If the SCEs are reasonably assured that a group of similar nponents are installed in the same manner then it is not necessary to inspect each and every component. However, a thorough inspection of each component was performed to comply with the GIP requirements. This included obtaining access to electrical cabinets to inspect anchorage and the internal construction. The walkdowns were coordinated with the outage schedule to ensure a complete and thorough inspection was performed for each component on the walkdown list.

3.1.2 Success Paths For The IPEEE Seismic Assessment

As noted previously, Toledo Edison elected to follow the SMA methodology developed by the EPRI as described in report NP-6041 (Ref. 3). The NRC has identified certain enhancements to that procedure to satisfy the intent of the IPEEE. These enhancements will be addressed in the SMA for Davis-Besse as well.

The EPRI SMA procedure entails defining minimum sets of systems and equipment needed to bring the plant to a safe shutdown state. This is accomplished through the construction of success path logic diagrams (SPLDs). These diagrams show basic success paths to safe conditions, and alternative paths where they may be needed. Once these success paths are defined at the level of plant systems, the components in each system that must have the margin to perform their functions following an earthquake can be identified. Depending on the nature of these components, they may be subject to various levels of scrutiny to ensure that they have sufficient seismic margin. For many types of components, this would volve a screening assessment and walkdown to verify their ruggedness. For others, a more detailed sessment to evaluate their seismic margins relative to a review level earthquake (RLE) would be required. The seismic margin in these cases would be expressed in terms of a value representing a high confidence of a low probability of failure (HCLPF).

The first major step in the SMA is to select the required shutdown equipment through the development of the SPLDs. The application of this process for Davis-Besse is outlined in the following discussion. The enhancements called for by NRC involve explicit consideration of some non-seismic failures and evaluation of containment. These aspects are addressed in the discussions that follow as well.

3.1.2.1 Selection of Success Paths

The procedure outlined in NP-6041 is applied by defining success paths that would lead to a stable condition in which core cooling would not be threatened. These success paths must include the systems needed to attain hot shutdown, assuming offsite power is not available, and to maintain a safe condition for at least 72 hour after the earthquake. As a minimum, one primary success path and an alternative must be defined. If only one set of success paths is chosen, the Seismic Review Team (SRT) must show that the potential for leakage from the reactor coolant system (RCS), such as due to the combined effects of failures of instrument connections in various portions of the system, is very small. Another option permitted by the SMA procedure is to define a second set of success paths, in which it is assumed that the combined leakage from various RCS sources constitutes a small LOCA with an equivalent diameter of one inch. cause of the difficulties that might be encountered in verifying that leakage would not occur, and especially the need to walk down many small lines connected to the RCS, the second option was pursued · Davis-Besse. Thus, two sets of success paths are defined. In one, the RCS is assumed to retain its integrity. In the other, the systems that would need to respond to a small LOCA must be identified and evaluated. In each case, it is necessary to define both a primary path and an alternative path that is as independent of the primary as possible.

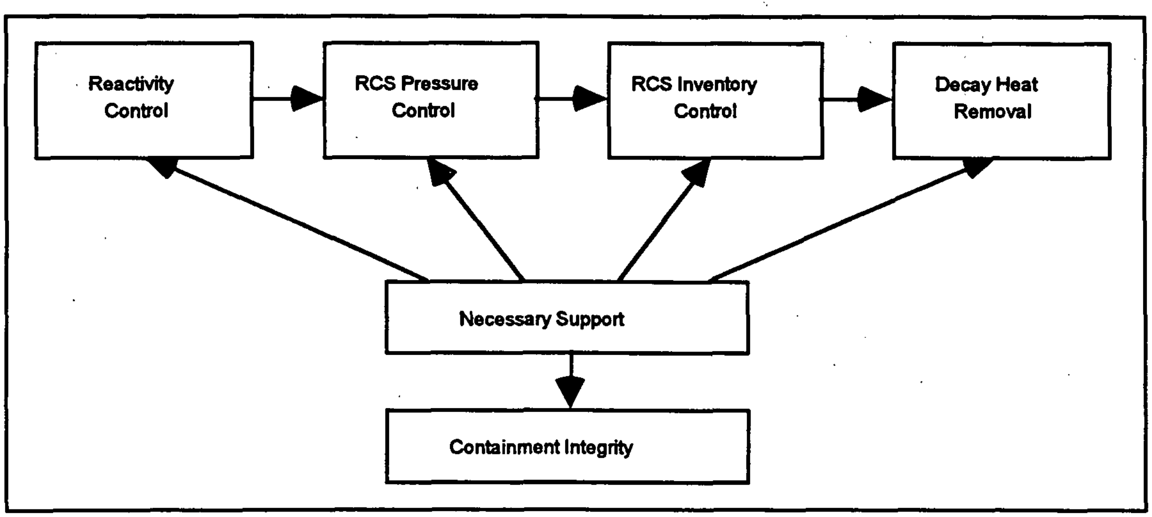
The purpose of defining both a primary path and an alternative path is to account for the potential that a problem might be encountered in showing that the systems and equipment needed for the primary path would have sufficient seismic margin. A balance must be struck between the amount of effort that might be required to walk down and evaluate the equipment in multiple pathways, and the possibility that an initial evaluation focusing only on the primary path might lead to the need to perform additional walkdowns late in the assessment when a problem was encountered. The NRC has requested that each utility initially take a broad view of the potential success paths, rather than focus too quickly on a single primary path and an alternative. In most cases, the number of alternatives is relatively limited; although additional paths may be defined, they typically become progressively less independent of earlier paths, and often involve more complex sets of systems and equipment.

In selecting success paths and constructing the corresponding SPLDs, an attempt has been made to reflect consideration of the projected seismic margins of critical equipment, the equipment already identified for evaluation for the A-46 effort at Davis-Besse, and the most logical primary and alternative paths. Based on the procedure laid out in NP-6041, the supplemental NRC guidance, and knowledge of ¬proaches being taken at other utilities, the following ground rules apply to the development of the .ccess paths:

1. The success paths must be adequate to maintain a safe condition for 72 hour following the earthquake.
2. For purposes of the IPEEE, the end state for the success paths may be hot shutdown. End states as they were defined in the IPE for internal events (Ref. 5) should be acceptable. A preliminary decision has been made to consider cold shutdown as the desired end state for both the intact RCS and the small LOCA cases.
3. In defining the paths, it must be assumed that the plant would be without offsite power for the entire 72 hour period. Note that any adverse effects that might result if offsite power were to remain available or be recovered at some point would need to be considered.
4. One set of success paths must be adequate to achieve success given the RCS remains intact. In the other set of success paths, it must be assumed that there is leakage from the RCS equivalent to a small LOCA with an equivalent diameter of 1 inch.
5. Redundant trains of a system are generally considered to be completely coupled with respect to seismic loadings. Therefore, if one train has sufficient seismic margin, the other should too. Likewise, if one train fails to have sufficient margin, the other cannot be relied upon as a backup.
6. The potential effects of relay chatter must be considered, but in a separate evaluation step.

7. Non-seismic failures of equipment must be addressed. This can have an effect on the success paths that are chosen for evaluation, if systems are included that rely on certain components or operator actions known to have high failure rates.
8. The integrity of small lines (i.e., less than 1 inch in diameter) connected to the RCS does not need to be considered, since one of the SPLDs will directly address the systems needed for a small LOCA.
9. The potential for flow diversion that could affect the function of a system needs to be considered only for cases in which the system function could be defeated. This should be addressed in the same manner as it was in the IPE. For example, failures of instrument lines attached to high pressure injection (HPI) piping should not constitute an adequate path for loss of flow to lead to core damage following a small LOCA.
10. The evaluation of seismic margin for the containment and associated systems is to be focused on the potential for early containment failure. For Davis-Besse, this is limited to consideration of the need for the containment vessel itself to maintain its integrity, and for the penetrations that could constitute leakage paths from containment to be isolated. Avoidance of other possible modes of early containment failure does not generally rely on particular system responses that would be impacted by the occurrence of an earthquake.

Four basic safety functions must be satisfied in each SPLD with respect to achieving and maintaining a stable condition for core cooling. They are illustrated in the figure below. In addition to ᵌse, the support necessary for the systems that perform these functions must be considered. Some aspects of containment integrity must also be assessed, irrespective of the response with respect to core cooling. The four basic safety functions are discussed below separately in the context of the two sets of boundary conditions (i.e., no LOCA and small LOCA). The support systems and containment issues are then discussed.



Basic Safety Functions for the SMA Success Paths

..1.2.1.1 Success Path Logic Diagram for the Intact RCS Case

In the first set of success paths, the challenge to the plant systems is functionally equivalent to a loss of offsite power. It is not necessary to postulate a loss of RCS integrity as a result of the earthquake. The selection of success paths for each of the safety functions is described below.

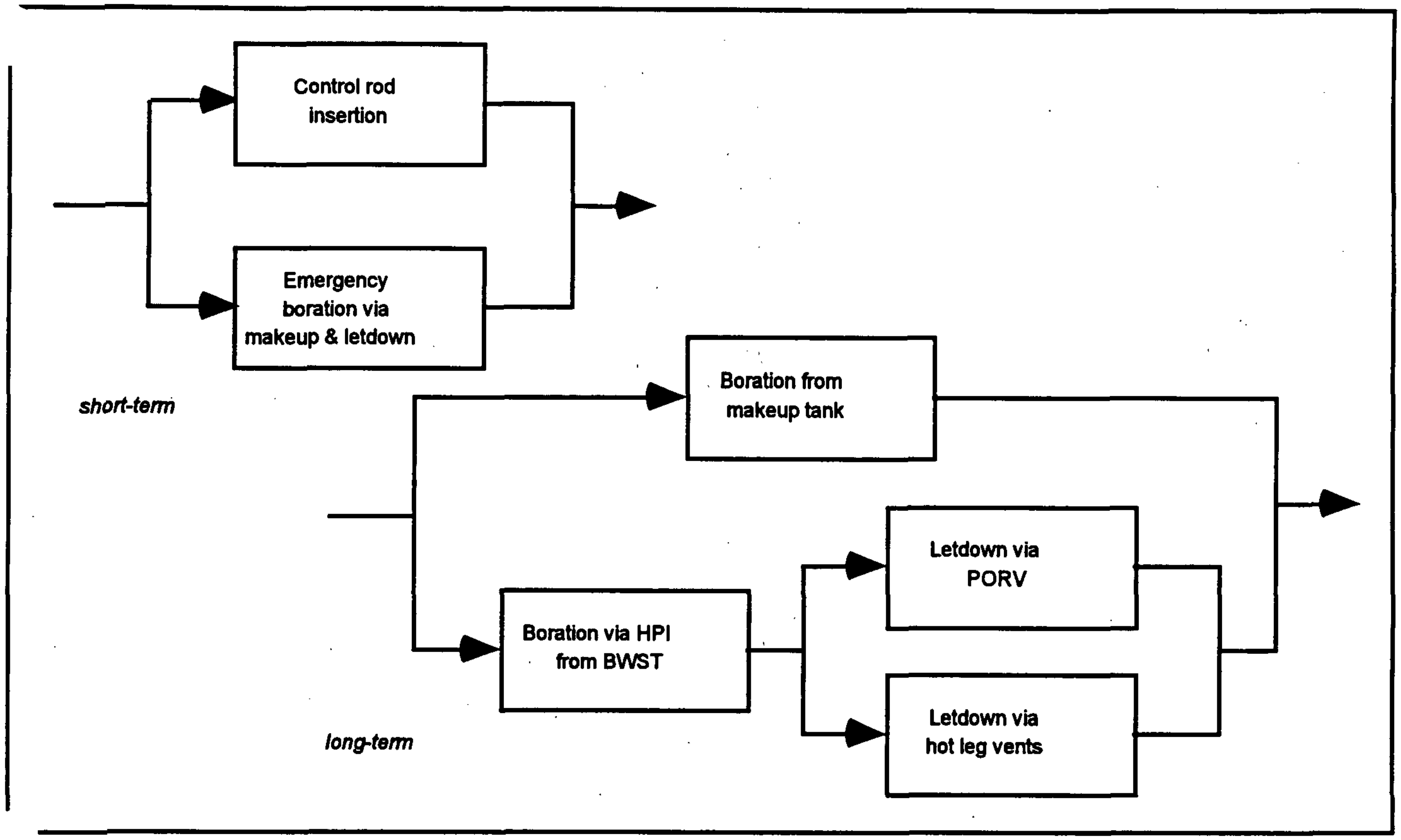
Reactivity Control

It is implicit that the earthquake will interrupt normal operation (e.g., by causing offsite power to be lost). Therefore, reactivity control would be required initially to reduce reactor power to decay heat levels. The primary path for short-term reactivity control for either of the two cases would be shutdown by insertion of the control rods. The alternative path would nominally entail emergency boration, although the primary path should be sufficiently reliable during an earthquake that the alternative path would not require detailed walkdown and evaluation.

To proceed successfully to cold shutdown, it could be necessary to provide additional insertion of negative reactivity. This would be needed to compensate for the positive feedback effects of lowering the temperature of the reactor coolant, and would have to be done by increasing the concentration of boron in the reactor coolant. Borated water can be made available for injection to the RCS from the borated water storage tank (BWST) or by addition of borated water to the makeup tank. If the BWST were used as the ~ource of highly borated water, it would be necessary to let down some RCS inventory to ensure that .equate boration could be achieved. Without removing some RCS inventory, a volume from the BWST roughly equivalent to only the shrinkage of reactor coolant caused by the cooldown could be added. This would not necessarily afford sufficient negative reactivity insertion to assure subcriticality. Therefore, either injection of water from the makeup tank with a higher boron concentration would be required, or a letdown path would be needed. The injection of water at high boron concentrations would require use of the boric addition pumps to transfer borated water from the boric acid addition tank. On the other hand, other non-safety equipment could be required if a letdown path were needed in conjunction with boration from the BWST.

The SPLD for the function of reactivity control is shown in the figure below for both short-term and long-term requirements. Two paths are shown for each case. Injection of borated water from the makeup tank by the makeup system was selected as one path for long-term reactivity control. As a second success path that would be as independent of the other path as possible, the injection of BWST water by the HPI system was selected. The pressurizer pilot-operated relief valve (PORV) or the hot leg vents could serve as a means to remove RCS inventory to permit injection of sufficient borated water. This would eliminate dependence on the makeup system for both elements of the function.

Note that the need for long-term reactivity control arises only because of the selection of cold shutdown as the required end state for the SPLD. If a decision is later made to consider remaining at hot shutdown conditions as adequate to prevent core damage, the long-term portion of the reactivity control nction could be eliminated from the SPLD.



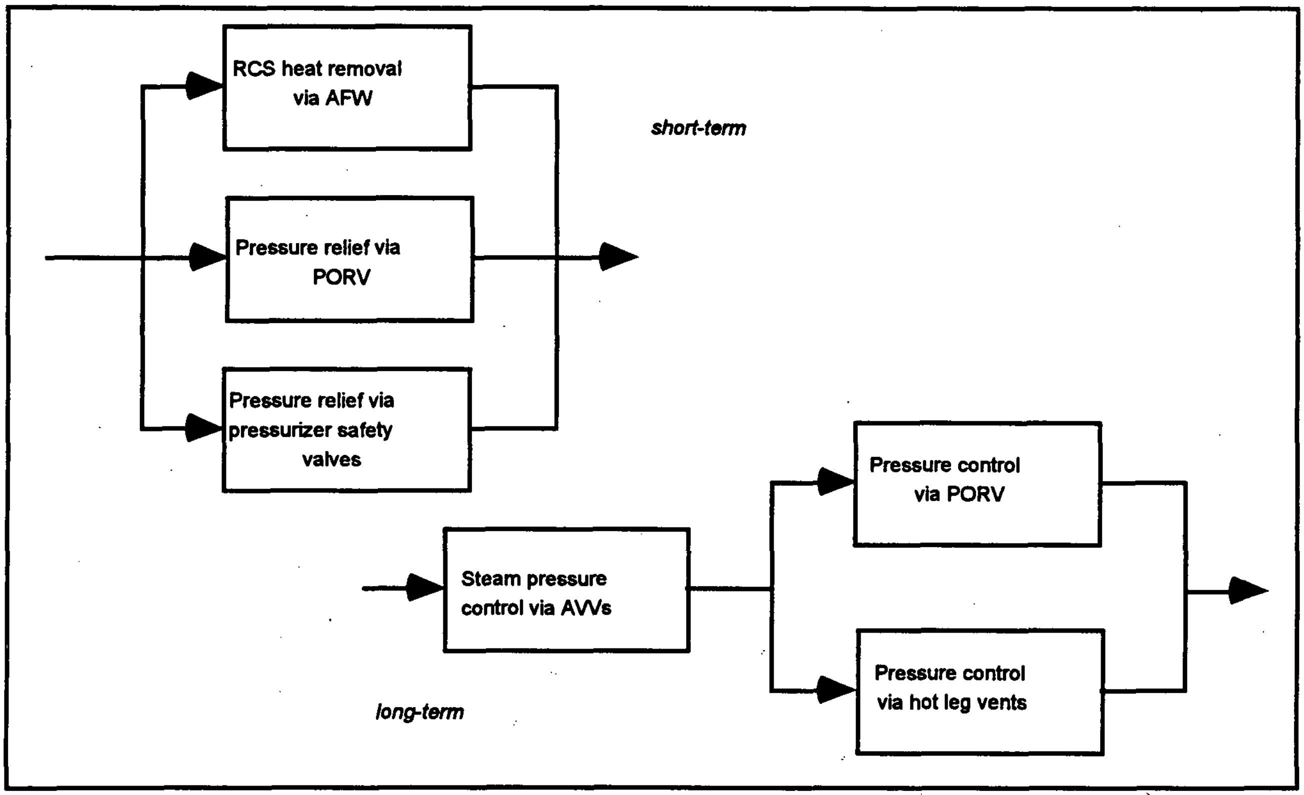
Success Paths for Reactivity Control (Intact RCS Case)

RCS Pressure Control

Control of RCS pressure depends largely on the mode of cooling that is considered in the SMA. For the intact RCS case, if auxiliary feedwater (AFW) were successful in providing RCS heat removal, there is no reason to believe that overpressure protection would be required to ensure that core cooling was preserved. If feedwater were not available, operation of the PORV and/or pressurizer safety-relief valves (PSVs) would be required to avoid excessive RCS pressures. Without feedwater, operation of some combination of these valves would be required for success of makeup/HPI cooling (i.e., as an alternative success path). Other failures (i.e., loss of pressurizer heaters) could cause operational problems, but should not cause core cooling to be lost.

As in the case of reactivity control, the need to achieve cold shutdown presents additional demands for pressure control in the longer term. To cool down the RCS, some combination of abilities to reduce steam pressure and RCS pressure would be required. Because the turbine bypass valves would be unavailable (due to loss of condenser vacuum, loss of instrument air, etc.), control of steam pressure would require reliance on the atmospheric vent valves (AVVs). The air-operated AVVs are equipped with accumulators that would permit them to be cycled for some period of time, after which local manual operation might be required (they have handwheels, and there are detailed instructions in the procedure : their control after loss of air). Because the reactor coolant pumps (RCPs) would not be available, it would not be possible to control RCS pressure using the pressurizer spray. Therefore, RCS pressure would ed to be controlled using the PORV and/or the hot leg vents.

The portion of the SPLD that would apply to control of RCS pressure is shown in the figure below. For short-term response, three alternative paths are shown; these do not require extensive investigation, and they are not redundant when viewed from the perspective of the decay heat removal function (as described below). For long-term pressure control, no clear alternative to the AVVs has been identified for reducing pressure in the steam generators.



Success Paths for RCS Pressure Control

Once again, it should be noted that the portion of the SPLD associated with long-term control of RCS pressure would be eliminated if the decision is made to consider long-term core cooling at hot shutdown conditions as acceptable.

RCS Inventory Control

For the case in which the RCS is assumed to be intact, several types of problems relating to RCS inventory could be postulated. If there were pre-existing leakage, if normal letdown could not be isolated, or if leakage developed from the RCP seals, it is possible that, over the 72 hour period following the earthquake, there would be insufficient inventory to permit cooldown and to achieve a long-term, stable mode of core cooling. Makeup flow is already required for success of the reactivity control function in the ıg term, to support reaching cold shutdown. No further addition of inventory should be required.

If a small LOCA were to occur due to failure of the RCP seals, the impact would be to shift from the act RCS case to the small LOCA case. There is no reason to develop the RCS inventory function relative for an induced small LOCA in the intact RCS SPLD, since that is directly the focus of the other SPLD.

It is possible that problems associated with excessive RCS inventory could arise following an earthquake. For example, failure to control makeup flow or in-leakage from RCP seal injection could cause pressurizer level to rise. Although this could create operational problems for the operators, it is not clear that it would lead to a threat to continued core cooling. It could impede an orderly cooldown, and if the pressurizer were filled, the RCS could be pressurized to the point that a relief valve would be challenged to open. There would be no immediate threat to RCS integrity, unless the relief valve opened and failed to reclose (which would constitute an additional, non-seismic failure). Thus, while a rigorous assessment might indicate that this could cause the function of RCS inventory control to be unsuccessful, it would not directly threaten core cooling. It should not be necessary to include this consideration in the success paths.

As described below, the alternative path for decay heat removal involves makeup/HPI cooling. For this alternative, additional makeup flow is required as part of the heat removal function, and the function of controlling RCS inventory is superseded.

Therefore, in light of these considerations, it is not necessary to construct a portion of the SPLD to cover considerations relating to RCS inventory control for the intact RCS case. If there is leakage from the RCS, it will be accommodated by the makeup required to accomplish other safety functions.

If the primary path were not required to proceed to cold shutdown, such that the need for injection of borated water was eliminated (thereby removing the implicit makeup function), it would be necessary to reconsider the control of RCS inventory further. In that case, the success path might need to consider the following: that letdown could be isolated; that the likelihood of pre-existing leakage sufficient to threaten the inventory needed for core cooling over the 72 hour period is small, and that RCP seal integrity (relative to leakage, not serious failure) would be preserved. Makeup would be an alternative to failure for any of these three potential threats to the inventory function. Further alternatives would also exist (e.g., partial depressurization sufficient to achieve at least some injection from the HPI system).

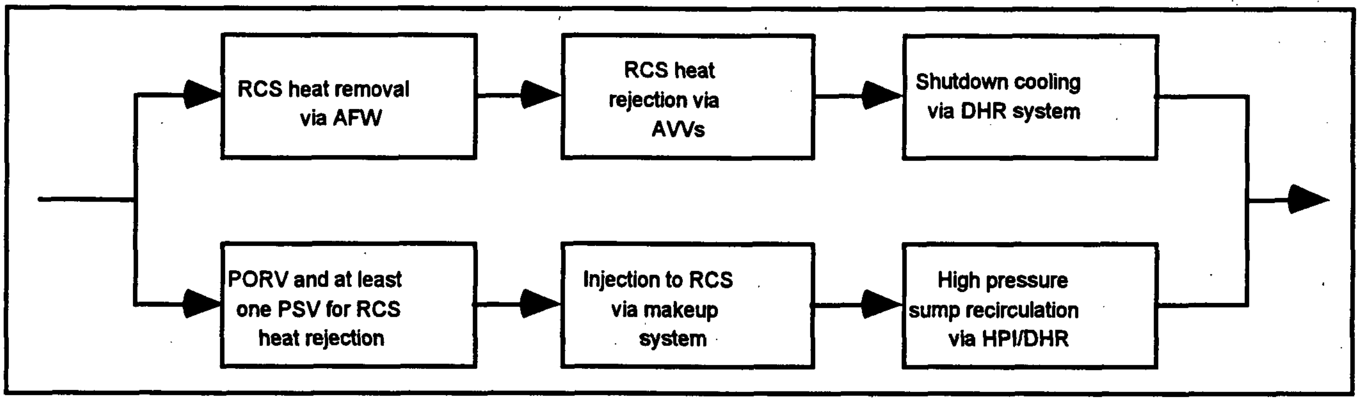
Decay Heat Removal

Decay heat removal would initially be accomplished via the steam generators, with AFW supplying flow and heat removed through the AVVs and/or the main steam safety valves (MSSVs), since the main condenser would not be available following the loss of offsite power. AFW would also be needed to remove sensible heat to permit cooldown to cold shutdown conditions. As noted under the function of RCS pressure control, control of steam pressure using the AVVs would be required as well to support the cooldown. If the cooldown is successful, the decay heat removal (DHR) system would be used for long-term cooling.

If AFW were not available, the principal alternative would be to remove decay heat via makeup/HPI cooling. This would initially involve use of the makeup system to provide injection of cold water, with a bleed path established by opening some combination of the PORV and PSVs. When the

BWST was depleted, it would be necessary to enter into high pressure recirculation from the containment mp. This would require use of the HPI system, since procedures prohibit using the makeup system in this mode. For successful injection by the HPI pumps, it would also be necessary to use the PORV, since the shutoff head for the pumps is too low to permit injection at the lift setpoint for the PSVs.

The portion of the SPLD that applies to the decay heat removal function is shown in the figure below. Note that, although decay heat could initially be rejected via the MSSVs, the AVVs would be required to support cooldown for long-term cooling. Hence, only the AVVs are shown in the diagram.



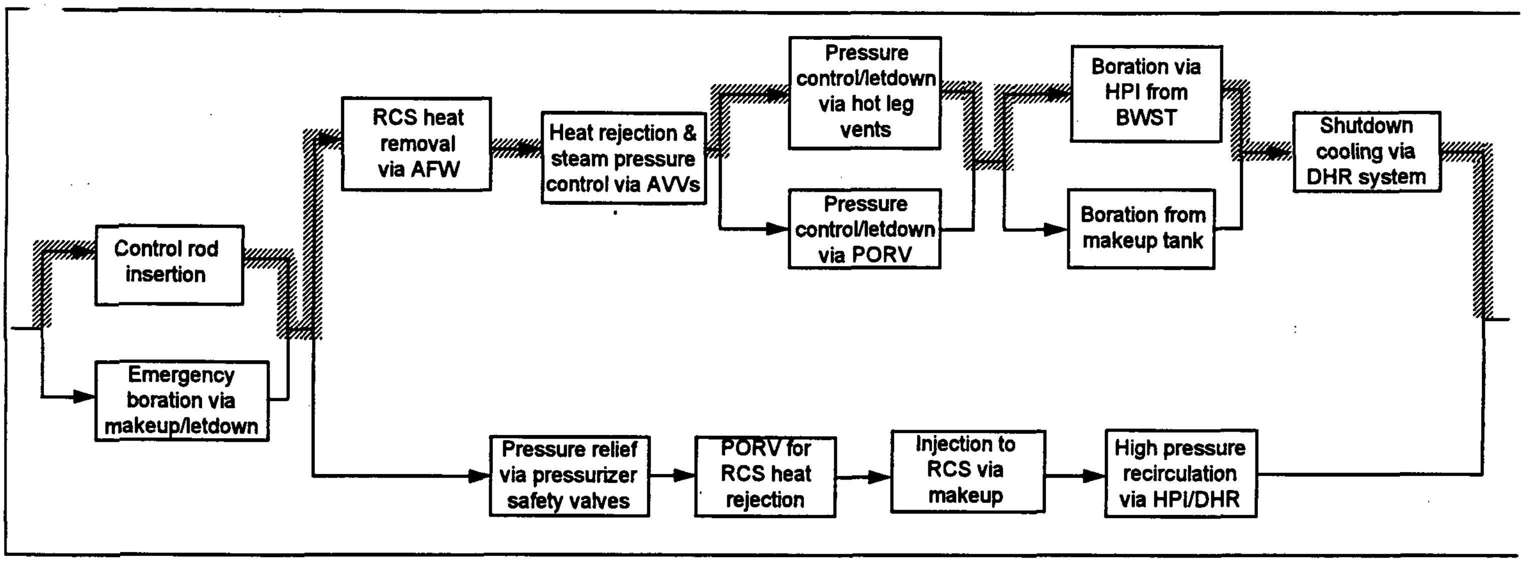
Success Paths for Decay Heat Removal (Intact RCS Case)

If the primary path were to terminate in hot shutdown, the block relating to shutdown cooling via the DHR system would be eliminated. The AVVs would not necessarily be needed; the MSSVs could be used for the rejection of RCS heat (the AVVs could be retained as a sub-alternative for that portion of the SPLD).

Integrated SPLD for the Intact RCS Case

The portions of the SPLD for the various safety functions were assembled to identify the overall success paths, as shown in the following figure. In the figure, the primary path is highlighted. As indicated, the primary path for short-term reactivity control would be via the insertion of control rods. Emergency boration is indicated as the alternative path for this function, although it is not necessary to evaluate the seismic capacity for this path.

The remaining elements of the safety functions are integrated. Heat removal via AFW would clearly be the primary path; if that were successful, it would provide for both RCS pressure control and decay heat removal in the short term. Rejection of heat via the AVVs would be part of both the removal of decay heat and the long-term control of RCS pressure. RCS pressure control for long-term cooldown could be accomplished via either the PORV or the hot leg vents. Because the PORV is an essential part of the alternative path, the hot leg vents were chosen as the primary means for this pressure control function. This could also provide sufficient bleed-off of reactor coolant to allow boration via the HPI system from the BWST for long-term reactivity control. Finally, the DHR system would be used for long-term decay heat noval.



Integrated SPLD (Intact RCS Case)

The alternative path accommodates all of the safety functions after initial reactivity control. The pressurizer relief valves would be needed initially to provide for adequate pressure relief and for some removal of heat during the early stages of makeup/HPI cooling. Although the use of the PORV would not be essential for short-term makeup/HPI cooling, it would be required for long-term cooling after the BWST is depleted. The makeup system would be needed to provide injection for some period, until RCS pressure was low enough that adequate flow could be provided by the HPI system for full decay heat removal. In the long term, recirculation from the containment emergency sump via the HPI system, piggybacked onto the DHR system, would be required.

The primary and alternative paths unavoidably share some elements, including the HPI and DHR systems. These systems should seismically rugged. There are, of course, numerous support systems that would be shared by systems in both paths. This is discussed further in Section 3.1.2.1.3.

3.1.2.1.2 Success Path Logic Diagram for the Small LOCA Case

A second SPLD was constructed for the case in which it was assumed that the earthquake would leave the plant with a small LOCA equivalent to 1 inch in diameter. The formulation of the elements of the success paths is discussed below for each of the safety functions in the context of this small LOCA.

Reactivity Control

It is assumed that short-term reactivity control for the small LOCA case would require insertion of control rods. It is recognized that response to the LOCA would necessitate the injection of borated water, and there would eventually be sufficient insertion of negative reactivity to ensure shutdown. Insertion of the control rods should be adequately rugged, however, that this is not a concern. Since the function is needed for the intact RCS case, the inclusion of control rod insertion here will not be a limiting factor with respect to seismic margin, and will not require additional walkdown or evaluation. The injection of borated water from the BWST, in conjunction with the letdown effectively provided by the break(s), will sure long-term reactivity control adequate to achieve and maintain cold shutdown. No additional systems need be represented in the SPLD for long-term reactivity control, irrespective of the end state finally chosen.

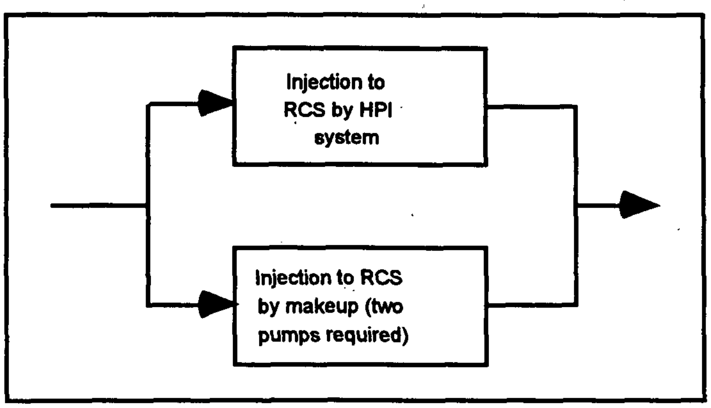
The portion of the SPLD associated with this function is identical to the short-term portion of reactivity control for the intact RCS case.

RCS Pressure Control

The initial break will afford a degree of depressurization of the RCS. If feedwater were unavailable, however, there would be a demand for the pressurizer relief valves to open. Thus, in the short term, the need for control of RCS pressure would be similar to the intact RCS case. In the longer term, it is not known if the break itself would provide adequate depressurization to permit attaining cold shutdown prior to depletion of the BWST inventory through injection to the RCS. Therefore, it is assumed that the PORV or hot leg vents would be required, as before. The AVVs would also be needed to permit adequate depressurization and corresponding removal of sensible heat from the RCS. Therefore, the success paths for this function are identical to those shown for the intact RCS case. If cooling at hot shutdown conditions is subsequently determined to be an acceptable means of long-term decay heat removal, no provisions for pressure control need to be included in the success path for the case in which AFW is available.

RCS Inventory Control

The small LOCA presents a demand for significant makeup to the RCS. This would generally be supplied by one of the HPI pumps. As an alternative, two makeup pumps could provide adequate injection. In the long term, cold shutdown would be achieved. For most leak locations, the leakage could eventually be terminated by lowering RCS pressure to essentially atmospheric levels. If the leak were low in the system (e.g., in the decay heat drop line), it would not be possible to stop the leakage altogether. At that point, however, given the very low rate of leakage and the success of the HPI and DHR systems, it would be appropriate to assume that sufficient inventory could be added to the RCS as necessary to ensure that the core was not uncovered. The success paths for the control of RCS inventory are illustrated in the below figure.



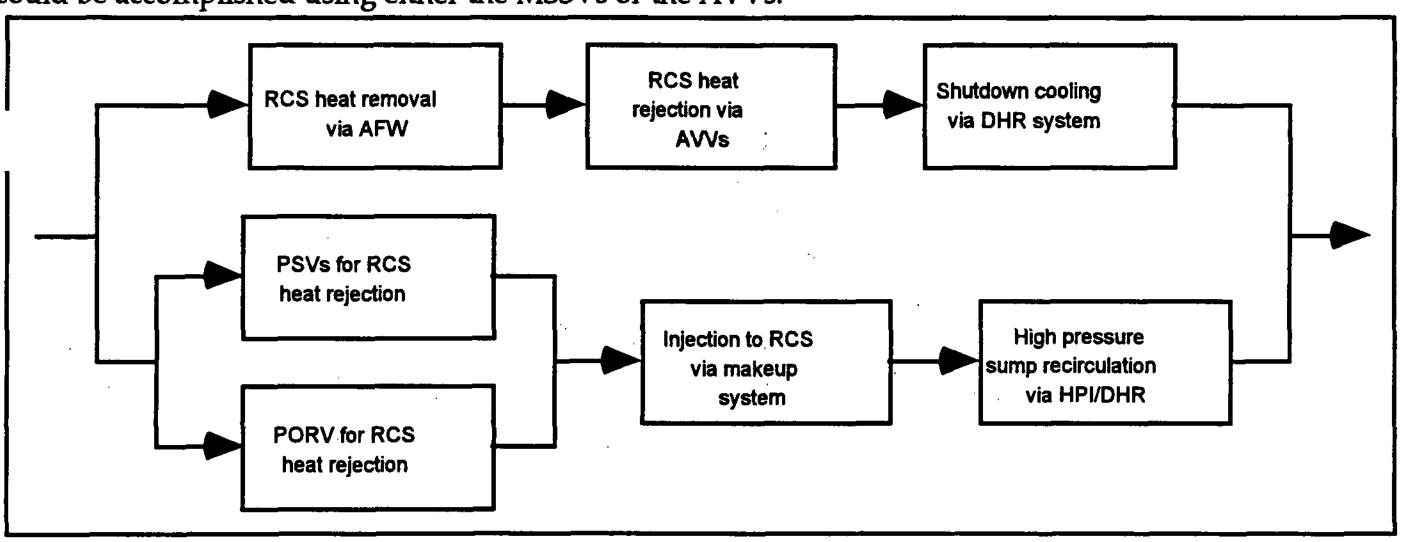
Success Paths for RCS Inventory Control (Small LOCA Case)

Decay Heat Removal

Because a 1-inch LOCA would be too small to remove decay heat fully for a significant period after the plant tripped, AFW would be needed to supply heat removal in the short term. As for the intact RCS case, heat removal by AFW would support cooldown of the RCS to cold shutdown conditions. After that, long-term decay heat removal would be accomplished by shutdown cooling via the DHR system. If feedwater were initially unavailable, makeup/HPI cooling would be required to provide adequate heat removal. The requirements for a bleed path would be somewhat diminished due to the presence of the LOCA. In the short term, it is expected that the PORV and PSVs could be redundant means to provide a bleed path, supplementing the break itself. In the long term, unlike the intact RCS case, the 1-inch LOCA would be adequate to remove decay heat before the BWST inventory was depleted, even with HPI as the injection source.

The portion of the SPLD associated with decay heat removal for the small LOCA case is shown in the below figure. This figure is very similar to that for the intact RCS case, except for the consideration of bleed paths for the makeup/HPI cooling pathway.

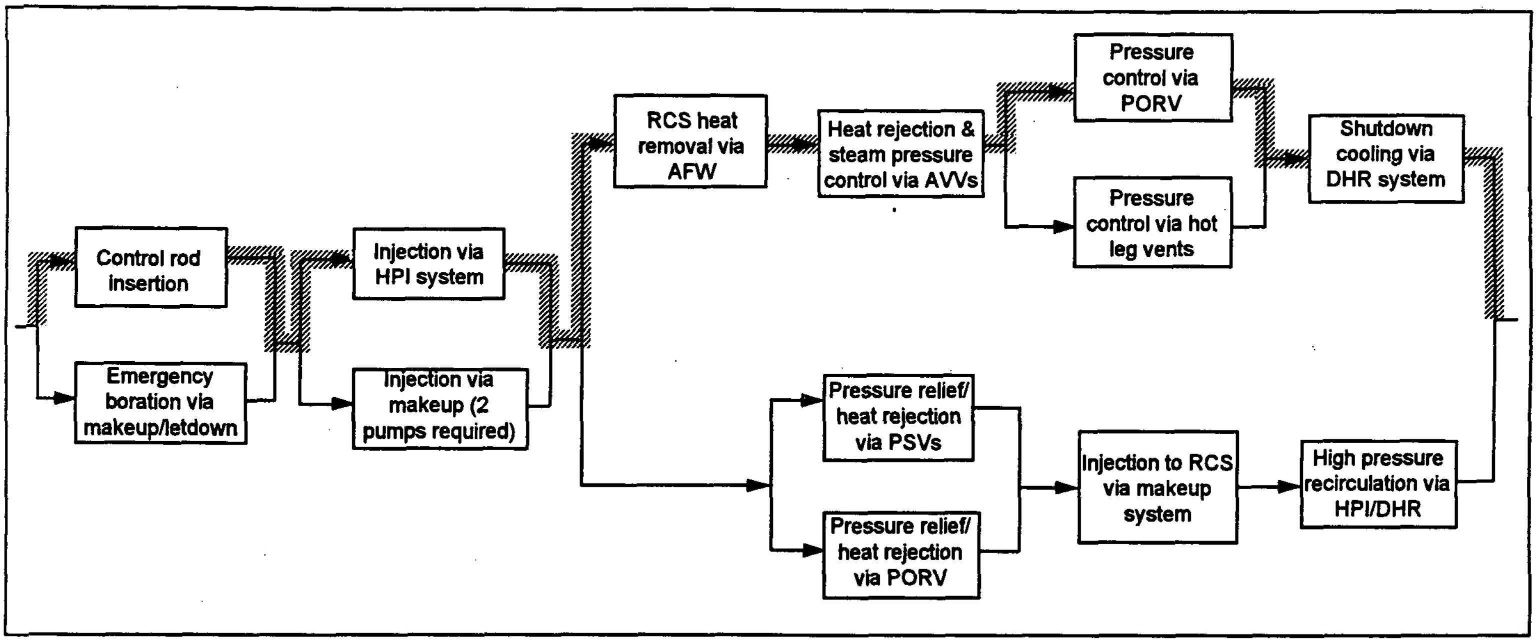
As in the intact RCS case, if it were concluded that cold shutdown did not need to be attained, the DHR system could be removed from the success path that includes heat removal via AFW. Steam relief could be accomplished using either the MSSVs or the AVVs.



Success Paths for Decay Heat Removal (Small LOCA Case)

Integrated SPLD for the Intact RCS Case

The integration of the small LOCA success paths for the individual safety functions results in the SPLD shown in following figure. As the descriptions for the individual functions would indicate, the overall pathways are very similar to those for the intact RCS case. The primary differences include the requirement for RCS makeup for inventory control, and the less significant differences relating to pressure control and bleed pathways for makeup/HPI cooling.



Integrated SPLD (Small LOCA Case)

3.1.2.1.3 Support Systems

Each of the systems identified in the SPLDs requires some level of support. This includes ac motive power, ac and dc control power, component cooling water, service water, and room cooling and ventilation. The PORV requires function of the containment air coolers (CACs) to ensure the containment environment will allow long-term operation. In each case, the required support must be specified and seismic margin verified in the same manner as for the systems associated with the safety functions. Support systems may be particularly important because their failures may, in many cases, affect both primary and alternative success paths. Furthermore, the consideration of non-seismic failures may be more critical with respect to the support systems. For example, the emergency diesel generators (EDGs) will be called upon due to the loss of offsite power that must be assumed in each case. The EDGs have relatively high unavailability compared to other components. This could be especially important with respect to the potential that failure of one EDG could affect the availability of both trains of AFW due to the possibility of overfeeding one of the steam generators when DC power was depleted. The support systems required for the systems associated with the success paths are detailed in dependency matrices and related documentation in the IPE report (Ref. 4).

3.1.2.2 Containment Integrity

The issue of containment systems is not addressed at all by NP-6041 (Ref. 3); it was identified as an enhancement to the NP-6041 procedure required by the Generic Letter. The guidance for containment considerations is, therefore, quite limited. It is primarily required that the systems needed to prevent an early containment failure be examined. For Davis-Besse, there are no systems that must function to prevent early containment failure by severe-accident phenomena. The only systems that need to function to preserve containment integrity are those that must isolate containment penetrations. There is reference NUREG-1407 (Ref. 4) to the desirability of considering the availability of containment heat removal, but there is no clear requirement to do so. Since the containment air coolers will be evaluated as a support function for the PORV, however, it would be appropriate to account for having considered them from a containment protection standpoint as well.

The containment structures themselves also require consideration, as outlined in NP-6041 (Ref. 3). This aspect is addressed in the next section.

3.1.2.3 Screening of Systems and Equipment

Unlike the Generic Implementation Procedure (GIP, Ref. 6), the SMA for the IPEEE does not generally exclude broad categories of equipment from consideration. The scope of the evaluation effort required for the SMA can, however, be substantially reduced through a screening process. Equipment that should have very high seismic capacity, based on the judgment of the Seismic Review Team, previous seismic evaluations (e.g., seismic PRAs), and past experience from actual earthquakes, can be "pre-screened". During the initial walkdowns, any such equipment must generally be inspected to confirm the pre-screening judgments. The rationale for screening out the components can then be formally documented, and no further consideration of that equipment is required.

The types of equipment that may be subject to screening and the rationale for expecting that they may be screened are described in NP-6041 (Ref. 3). Tables 2-3 and 2-4 of that report summarize this ormation, and it is discussed in more detail in Appendix A. Some of the systems and equipment that merit discussion at this point, especially with respect to the initial walkdowns inside containment, are discussed below.

Containment

Free-standing steel containment vessels that were designed for the combined loading of a design-basis accident plus the safe shutdown earthquake (SSE), with a dynamic analysis conducted for the SSE, have been found to have very high seismic capacities. Therefore, no further evaluation of such containments is necessary. The containment shield building is also screened.

NSSS Components

NP-6041 indicates that the NSSS components should have very high seismic capacities (with the possible exception of the pressurizer supports). Based on a review of design drawings, it should be possible to draw conclusions regarding support, anchorage, clearances, etc. for the NSSS components and major piping runs. Since one of the sets of success paths has been chosen based on assuming the existence of a small LOCA, it is not necessary to trace all instrument taps, small piping connections, etc., even on paper.

Appendix F of NP-6041 includes guidance for examining the NSSS components, including some aspects that could only be verified by an actual walkdown (e.g., assuring that all of the hold-down bolts in e anchorage systems are present). Worksheets for use during the walkdown are also provided.

Elsewhere in the document, however, it is pointed out that there may be limited opportunity or access to ,pect the NSSS components. Moreover, in Appendix D (Ref. 2, p. D-7), there is a discussion of steps to be taken to limit in-containment walkdowns to prevent undue radiation exposure. This would appear to support making a very limited inspection of NSSS components.

Therefore, it is recommended that the NSSS components be pre-screened prior to entry to containment, based on examination of design details. During the necessary walkdowns inside containment, any readily accessible NSSS supports should be "walked by" to verify, to the extent practical, the pre-screening conclusions. This would apply in particular to the pressurizer supports, since special concern has been raised regarding this component. Assuming satisfactory screening, detailed walkdowns and subsequent evaluations of NSSS components should not be required.

Valves

Table 2-4 provides screening criteria for valves. For passive valves (i.e., check valves and manual valves), the table indicates that only for cases in which the peak spectral acceleration would exceed 1.2 g would assessment for these valves be required. Even in that case, the assessment would be limited to consideration of potential system interactions (i.e., a heavy valve located in a very flexible run of piping, such that the valve could impact another component).

Later in the document, it is stated that passive valves and externally-operated valves that do not need to change state are not of concern and do not need to be included in the equipment list (Ref. 2, p. 3-‾‾). Thus, they are essentially excluded from the IPEEE review.

For active valves required to change state, pre-screening may also be performed. The screening conclusions should be verified by walkdown of a selected sampling of valves (that is, one valve in each category identified), with a walkby of critical valves to assure that the potential does not exist for system interactions.

Other Components

Many other types of components (e.g., horizontal pumps) can be pre-screened, with walkdowns to verify the screening assumptions. For these components, detailed evaluations of seismic capacities should not be required if the screening assumptions are verified. For components in large populations that are screened (e.g., electrical conduit), a sampling can be walked down, with the remainder walked by to the extent practical.

3.1.2.4 Consideration of Non-Seismic Failures

One of the enhancements to the EPRI SMA procedure identified in the NRC guidance is the need to consider more explicitly the role of non-seismic failures. These include relatively high-probability hardware failures and human interactions that could impact the success paths. No explicit procedure is identified for accounting for these failures in the success-oriented EPRI SMA approach. The method used in the NRC's SMA of Maine Yankee (NUREG/CR-4826, Ref. 7) was cited as one acceptable approach.

Because the NRC's SMA methodology is based on use of fault trees, however, that method does not lend elf directly to use in the EPRI approach.

The basic premise of the EPRI methodology is that the success paths should be selected so that they are reliable. NP-6041 suggests that the SRT should be confident that the overall unavailability of the success paths would be no higher than about 0.001. No means to assure that this is the case are identified; it is implicit in the EPRI procedure that only reliable paths would be credited.

To address the NRC concern, it appears that a more explicit treatment would be required. The most rigorous approach would be to construct a simplified fault tree that represents failure of the paths reflected in the SPLD. The fault tree would be based primarily on the IPE, although since some additional conditions must be taken into account (e.g., boration for long-term cooling at cold shutdown), some new logic development might be required. The fault trees could be pared to remove any basic faults that would affect only a single train of one system and that would have an unavailability of less than 0.01, or any faults that could affect multiple trains or multiple systems and that would have an unavailability of less than 0.001 (based on the screening performed in NUREG/CR-4826). The overall unavailability of the shutdown paths could then be estimated. The dominant contributors would have to be considered relative to the nominal criterion of 0.001 for the overall unavailability of the success paths. The steps that might be taken if this criterion were not met are not clear; that would depend on a qualitative assessment of the contributors.

Among the non-seismic failures that have been important in past seismic PRAs are the diesel nerators. These have relatively high "random" unavailabilities. This is exacerbated for the SMA by the requirement that recovery of offsite power must not be credited for at least 72 hour after the earthquake. In the IPE for internal events, recovery of offsite power is an important element. In addition to being relatively unreliable, failure of one or both EDGs would affect multiple systems.

This should not imply that the plant is inadequately designed for earthquakes. It may mean that the EDGs are the limiting factor with respect to non-seismic failures, and that they will provide a context for establishing a perspective on other non-seismic failures. Davis-Besse is not unique in this regard; every plant will depend on the availability of the EDGs for nearly any success path.

3.1.3 Analysis of Structure Response

This section identifies the various types of seismic analysis used to establish the seismic demand for evaluating components identified on the IPEEE Seismic Review SSEL.

Conservative design in-structure response spectra

DBNPS was designed using a modified Newmark spectrum with a horizontal pga of 0.15 g for the SSE and 0.08 g for the operating basis earthquake (OBE). The vertical pga for the SSE and OBE is taken as 2/3 of the corresponding horizontal acceleration.

The east-west component of the 1935 Helena earthquake was selected because many of its ıracteristics are similar to those which would be expected of the SSE at DBNPS.

The spectral amplification factors suggested by Professor Newmark were modified for two reasons. First, the maximum ground accelerations were not reduced by the factor 0.67 as it is suggested by Professor Newmark when the foundation conditions consist of competent rock. Also, consideration was made of the response spectrum of the Helena earthquake.

The original seismic analysis placed the free field seismic ground input motion at the foundation level of each building. This resulted in a conservative analysis as the seismic input motion was not deconvoluted by the soil.

The dynamic analysis performed on the structures consisted of reducing the structure into a mathematical model in terms of lumped masses and stiffness coefficients. The locations for lumped masses are chosen at floor levels and points considered to be of critical interest. Between mass points, the structural properties are reduced to uniform segments of cross-sectional area, effective shear area, and moment of inertia. The foundations of the Containment and Auxiliary Building Area 7 and 8 are located on competent rock and, consequently, are represented in the model as fixed bases. Investigations indicate that the influence of translation and rotation on the rock is small and, therefore, can be neglected. In the Auxiliary Building Area 6, the column foundations are coupled with the surrounding soil to form a soil-structure interaction model. With this information, a computerized analysis is used to form the stiffness matrix, [K], of the structure.

In addition to the original seismic analysis as described above, two additional seismic analyses were developed to establish seismic demand for the resolution of USI A-46 and IPEEE programs. A discussion of these two additional analysis follows.

Realistic, median-centered In-structure Response Spectra

EQE International was contracted by Toledo Edison to generate in-structure response spectra for use in the resolution of seismic IPEEE and USI A-46. The IPEEE in-structure response spectra were generated for a Review Level Earthquake with a NUREG CR-0098 median rock spectral shape anchored to a 0.30g pga. Throughout the analysis process, analysis parameters such as soil properties, structural properties and analysis methodologies were chosen to reflect a median centered analysis philosophy. To perform a median centered analysis, soil-structure interaction (SSI) was also considered.

The best estimate structural models used for this analysis were based on the mathematical models used in the licensing based seismic analysis. These original models represent the structures as sets of two planar (two dimensional) models, and neglected the effects of coupling between the two horizontal directions and eccentricities in the structures. In order to better represent a realistic, less conservative model of the dynamic behavior of the structures, three dimensional mathematical models of the structures were developed. These 3-D models were based on the licensing based 2-D models, supporting calculations to those models, and as-built drawings of the different structures. The raw spectra were then broadened

+/- 15 %. The broadened spectra for each mass point degree of freedom were then enveloped for all three l conditions.

The IPEEE in-structure response spectra were then scaled following the guidance of Section 4 of the GIP in order to create the USI A-46 in-structure response spectra. The ground motion, used to calculate the scale factor, was a NUREG CR-0098 84% non-exceedance probability (NEP) shape anchored to the site safe shutdown earthquake (SSE) peak ground acceleration of 0.15g. The value used for the scale factor was 0.697. The same scale factor was applied to all IPEEE spectral values in order to develop the USI A-46 spectra.

Based on the criteria identified in the GIP, the above analysis can be classified as a realistic, median-centered in-structure response spectra. Realistic, median-centered spectra were developed for all structures within the scope of the USI A-46 and IPEEE programs.

Reg. Guide 1.60 In-structure Response Spectra

The third spectra used is a Regulatory Guide 1.60 spectra that was developed specifically for the USI A-46 program and is applicable only for Area 7 and 8 of the Auxiliary Building.

As identified in Section 3 of the SSRAP Report, (Ref. 8) "SSRAP envisions that realistic (essentially median centered) in-structure spectra will be used for this comparison. Very conservative design spectra may be used, but their use is likely to introduce substantial conservatism". The realistic, median-centered in-structure response spectra developed for USI A-46 is a "scaled" IPEEE in-structure response spectra. e scaling process was performed in accordance with Section 4.2.4 of the GIP and introduced additional conservatism.

It was determined that a new conservative in-structure response spectra be developed using the criteria in the Standard Review Plan. This new in-structure response spectra would have less conservatism than the original seismic analysis yet still be a "conservative" spectra and not subjected to the additional safety factors as a realistic, median-centered spectra would. Only Area 7 and 8 of the Auxiliary Building were reanalyzed for this new spectra.

The structural models used for this analysis were based on the mathematical models documented in the original (licensing basis) analysis. These original models represent the structures as sets of two planar (two dimensional) models, and neglects the effects of coupling between the two horizontal directions and eccentricities in the structures. In order to better represent the dynamic behavior of the structures, three dimensional mathematical models of the structures were developed. These 3-D models were based on the 2-D models in the original analysis, supporting calculations to those models, and as built drawings of the different structures.

To create a three dimensional model from the 2-D model, the following properties were calculated in order to complete the 3-D model: the torsional stiffness of the wall system, the 3 rotational mass terms about the floor centers of gravity, the center of gravity of each floor, and the center of rigidity for each wall system. These properties were calculated based on information found in the original supporting .culations for the 2-D models, and structural plant drawings.

The result of these calculations were structural stick models that represent the 3-dimensional nature d eccentricities of the buildings studied. Dynamic eigenvalue extraction analyses were then performed on the stick models using the EQE program MODSAP. The resulting eigen systems were compared against the modal frequencies and participation factors calculated in the original analyses for the 2-D models. Where differences were found, they were determined to be attributable to the inclusion of the rotational mass moments of inertia and the inclusion of eccentricities. A modal damping ratio of 7% was assumed for all modes. This damping is considered appropriate for the reinforced concrete structures studied for the earthquake level considered, and is consistent with the recommendation given in Regulatory Guide 1.61 "Damping Values for Seismic Design of Nuclear Power Plants."

Frequency points were chosen for the calculation of the response spectra to ensure compliance with RG 1.122. These parameters are: 163 frequency points between 0.2 and 34 Hz. This results in a constant logarithmic increments between frequency points of 1.032.

A RG 1.60 shaped free field ground response spectrum anchored to the site SSE peak ground acceleration (0.15g) was used. The definition of the control point for the target freefield input motion was taken to be at the ground surface. This definition is considered to be appropriate and consistent with the intent of the SRP, as the till soil layer is of high stiffness.

The treatment of the calculation of the high strain soil properties and the deconvoluted motions for the CDSRA is also different from a median centered analysis. For a CDSRA, the low strain shear moduli are scaled by 1/2 for the lower bound and 2.0 for the upper bound soil conditions. Furthermore, the velope of the spectra of the deconvoluted motions from the three soil cases must envelop 60% of the free field ground surface target spectrum at the foundation level.

Using EQE's program, SSSIN, fixed base analyses were performed for both Areas 7 and 8 of the Auxiliary Building. Input to the analyses were the deconvoluted RG 1.60 pga of 0.15g motions for the three soil cases. The 3% and 5% damped response spectra of the output acceleration time histories were then calculated. The responses for each DOF were then enveloped for the three soil conditions, the resulting spectra broadened by +/- 15%.

3.1.4 Evaluation of Seismic Capacities of Components and Plant

3.1.4.1 Relay Evaluation

DBNPS performed a USI A-46 Relay Evaluation. The results of this evaluation were submitted by letter dated August 29, 1995 (Toledo Edison Serial Number 2316). The majority of IPEEE safe shutdown equipment requiring relay evaluations were previously evaluated for contact chatter during the USI A-46 Relay Evaluation. IPEEE safe shutdown equipment not previously evaluated by USI A-46 is denoted by a "T" in the 'IPEEE ONLY' column on the IPEEE SMA Safe Shutdown Equipment List (Appendix A). Only equipment unique to IPEEE seismic program are evaluated for bad actor (low seismic ruggedness) relays and described in this report. As used in this evaluation, low ruggedness relays are contact devices for which contact chatter is unacceptable and are listed in Appendix E of EPRI NP-7148-SL, "Procedure for aluating Nuclear Power Plant Relay Seismic Functionality".

Equipment requiring IPEEE low ruggedness relay reviews were grouped into broad 'circuit type' categories. These categories are: motor operated valve circuits, solenoid operated valve circuits, pump motor circuits, a neutron flux monitoring circuit, and level, pressure and flow circuits. In addition to the above 'broad circuit types', the IPEEE Review Level SSEL identifies three Safety Features Actuation System block switches and the three essential pressurizer heaters.

Motor Operated Valve Circuits

Low ruggedness relays are not used in the IPEEE motor operated valve circuits. These motor operated valves all receive power from Class 1E motor control centers (MCCs) which are backed by emergency diesel generators. The power flow path from the emergency diesel generators to the Class 1E MCCs was evaluated during USI A-46.

Contact devices used in the IPEEE motor operated valve circuits primarily consist of Westinghouse size 1 starters, Cutler Hammer and Rees momentary switches, G.E. SB-9 control switches and Limitorque limit and torque switches. Some valves use contacts from either the Safety Features Actuation System (SFAS) or the Steam and Feedwater Rupture Control System (SFRCS). Both of these systems were previously reviewed during the USI A-46 evaluation. Occasionally, Agastat or Deutsch auxiliary relay contacts are used in these circuits.

lenoid Operated Valve Circuits

Low ruggedness relays are not used in the IPEEE solenoid operated valve circuits. These solenoid operated valves receive Class 1E power from either the Safety Features Actuation System (SFAS) or from essential DC distribution panels. Both of these sources were previously reviewed during the USI A-46 relay evaluation. When equipped with auxiliary relay contacts, Agastat relays are used.

Pump Motor Circuits

Low ruggedness Westinghouse ITH overcurrent relays are used for ground fault detection in the 4kV High Pressure Injection and Makeup pump motor circuits. Contact chatter would result in tripping a running pump motor. The Westinghouse ITH relay does not initiate a lockout scheme; therefore, a tripped pump could be restarted from the Control Room. The remainder of the contacts comprising these pump motor circuits include: Agastat, Deutsch and General Electric auxiliary relay contacts, Westinghouse COM 5 overcurrent relay contacts, Allen Bradley and United Electric pressure switch contacts, Westinghouse and General Electric control switch contacts and Westinghouse breaker position switch contacts. Contacts like those listed above were evaluated during the USI A-46 relay evaluation and found to be seismically acceptable.

Neutron Flux Monitoring Circuit

Low ruggedness relays are not used in the IPEEE neutron flux monitoring circuit. This circuit is comprised of a Dixson indicator, a Gamma Metrics detector and Gamma Metrics amplifiers.

Level, Pressure and Flow Circuits

Low ruggedness relays are not used in the level, pressure and flow circuits. These circuits measure High Pressure Injection and Makeup flow and Core Flood Tank level and pressure. Components comprising these circuits are; Fisher-Rosemount and Motorola transmitters, Dixson indicators, Foxboro converters and square root extractors and Bailey converters, buffers, and indicators.

Safety Features Actuation System (SFAS) Block Switches

Three Safety Features Actuation System (SFAS) Block Switches are listed on the IPEEE Seismic Review SSEL. These switches are all Cutler-Hammer type E30DX, which do not appear on the low ruggedness relay list. Moreover, SFAS was evaluated during the USI A-46 Relay Evaluation and low ruggedness relays were not identified in this system.

Essential Pressurizer Heaters

Essential Pressurizer Heaters WMB1, WMB2 and WMB3 appear on the IPEEE Seismic Review SSEL equipment list. The control and power circuits for these heaters were evaluated during the USI A-46 relay evaluation. This evaluation showed that low ruggedness relays were not used in these circuits.

Relay Review Summary

The review performed for low seismic ruggedness relays (i.e., "bad actor" relays) for the equipment identified on the IPEEE Seismic Review SSEL, revealed there are no new relay chatter concerns that have not already been identified as part of the USI A-46 (SQUG) program.

3.1.4.2 Penetrations

A walkdown of containment penetrations from both the inside and outside of the Containment Vessel indicated that the penetrations were well supported off the containment steel shell. In addition, the applicable penetrations had sufficient flexibility between the containment vessel and the Containment Shield Building/Auxiliary Building to accommodate any differential movement between the two structures.

Inflatable seals or cooling systems are not used on the equipment hatch, personnel lock, emergency personnel lock, or other containment penetrations.

3.1.4.3 Subsystems

A walk-by was performed on the piping and HVAC subsystems as the SCEs were performing their inspections to identify any major concerns. When these items were encountered in the vicinity of equipment that were on the walkdown list, a more through inspection was performed for seismic interaction concerns. In general, these subsystems were found to be well supported with no interaction concerns.

1.4.4 Cable and Conduit Raceways

Table 2-4 of EPRI NP 6041 indicate cable trays and conduits have seismic capacities of at least 0.3g pga and a walkdown of these items is not required. However, cable and conduit raceways were walked down in detail in accordance with the GIP requirements. Only the buildings that house the walkdown equipment were included in the walkdown. In general, all rooms in these buildings were walked down unless it was fairly certain that no cable trays or conduits associated with the walkdown list were located or passed through the room. In a limited number of cases, access to rooms was limited due to radiation concerns. A total of 26 supports were selected for the Limited Analytical Review which represent the worst-case bounding samples of the raceway supports based upon a thorough walkdown of the areas.

In general the cable tray and conduit raceways are well supported. As a result of the Limited Analytical Review, two supports did not meet the GIP requirements and will be corrected under the USI A-46 program.

3.1.4.5 Control Room Ceiling

The Control Room and the cabinet room ceilings were inspected for seismic interaction concerns. These ceilings consist of a metal grid pattern with acoustic tiles inserted between the metal frame work and a light defuser panel in the Control Room. In general, the acoustic tiles are independently hung from the ceiling and the lighting fixtures and defuser panels are either independently supported from the ceiling or tened to the metal grid. The equipment located above the Control Room and cabinet room ceilings were inspected and was judged to be adequately supported such that there would be no interaction concerns. During the walkdown, the SRT identified some loose and cracked tiles along with missing clips and support wire. These deficiencies have since been corrected under the USI A-46 program.

3.1.4.6 Masonry Walls

DBNPS safety-related masonry walls were seismically qualified during the NRC IE Bulletin 80-11 Program. During the walkdown of Seismic Review SSEL equipment, masonry walls whose failure could impact the equipment were identified and compared with the list of walls covered by the IE 80-11 program. All masonry walls that could impact the equipment have been evaluated and are qualified for the site SSE.

3.1.4.7 Equipment

All components walked down that are part of the Seismic IPEEE program were either found to meet the GIP requirements or were previously identified as Outliers as part of the USI A-46 program. Those components identified as SQUG Outliers requiring fixes will be corrected in accordance with the USI A-46 program.

Appendix B contains the results of the walkdown for each component which is documented on the Screening Verification Data Sheets (SVDS). The SVDS summarizes the results of the seismic evaluation for ach component and indicates whether that component is included in the USI A-46 and/or IPEEE ograms.

3.1.4.8 Peer Review

Since the IPEEE Seismic program was combined into the USI A-46 program and the SQUG methodology was used for the walkdown, the third-party review made no distinction between the programs.

During the early stages of the walkdown program, two informal reviews were conducted by outside contractors. The purpose of these informal reviews was to assess the effectiveness of the walkdown teams in meeting the GIP requirements. In addition, two formal Third-Party Audits were conducted, the first was mid-way through the program and a final audit at the end of the program.

Dr. John D. Stevenson of Stevenson & Associates performed a formal audit midway through the program at the request of DBNPS Nuclear Assurance Department. Dr. Stevenson has approximately 30 years of experience in the seismic area, has been a contributor and reviewer of the SQUG program, and has performed SQUG walkdowns and Third-Party Audits at other nuclear facilities. Dr. Stevenson is an industry recognized expert in the seismic field.

Dr. Stevenson's activities included a review of completed SEWS packages, and the inspection of the corresponding equipment in the plant. A total of 15 items were reviewed. The following statements were documented by Dr. Stevenson in his report:

> "It is my opinion that none of my observations or recommendations made concerning SEWS Items 1-15 reviewed would result in the invalidation of the conclusions reached by the SRT's in their preparation of the SEWS."
>
> "My basic conclusion is that the USI A-46 resolution effort at Davis-Besse NPS is being performed using the SQUG developed Generic Implementation Procedure in a thoroughly competent and adequate manner. It should be understood that this program relies to a considerable extent on the judgment of the qualified SRT team consistent with the requirements of the GIP and therefore the judgment of a third part is not binding. . ."

Dr. James J. Johnson of EQE International performed his Third-Party Audit at the conclusion of the walkdown phase of the program. Dr. Johnson has over 20 years experience in the development, implementation and teaching of seismic issues. Dr. Johnson has played a significant role in the development of general and plant specific seismic evaluation procedures, including the SQUG program. Dr. Johnson has performed Third-Party Audits at other nuclear facilities. Dr. Johnson is an industry recognized expert in the seismic field.

Dr. Johnson's review of the SQUG program as implemented at Davis-Besse concluded that Toledo Edison implemented the GIP requirements "... in an appropriate and adequate fashion." In addition, Dr. Johnson agreed with the SRT's conclusion on the seismic adequacy of the equipment in which he reviewed.

3.1.5 Analysis of Containment Performance

NUREG-1407, Section 3.2.6 states "The primary purpose of the evaluation for a seismic event is to identify vulnerabilities that involve early failure of containment functions. These include containment integrity, containment isolation, prevention of bypass function, and some specific systems depending on a containment design".

As recommended in NUREG-1407, the DBNPS Independent Plant Examination (IPE) was used to determine the scope of the IPEEE containment performance evaluation. The DBNPS IPE evaluated the potential for containment integrity, containment isolation failure, and containment bypass.

As noted in the IPE (including update by Davis-Besse response to NRC request for additional information , Ref. 9), early containment failure represents only 0.28% of the potential containment failure modes. This is in comparison to 88% of overall containment failure possibilities which are characterized as "no failure".

Containment Integrity

Of the early failures analyzed, the plant damage states which contribute the most to the 0.28% involve sequences which involve a station blackout and the possibility of a reactor coolant system failure at high system pressure. Events of this type have the potential to involve a gross pressurization and failure of the containment vessel. As indicated in EPRI NP-6041 Table 2-3 and discussed in Section 3.1.1.4 the vessel is considered to be seismically rugged and requires no further evaluation.

Containment Isolation

Containment isolation was specifically noted in the IPE report as being "a negligible contributor to the potential for releases from the containment." A walkdown of containment penetrations from both the inside and outside of the Containment Vessel indicated that the penetrations were well supported off the containment steel shell. In addition, the applicable penetrations had sufficient flexibility between the containment vessel and the Containment Shield Building/Auxiliary Building to accommodate any differential movement between the two structures. Inflatable seals or cooling systems are not used on the equipment hatch, personnel lock, emergency personnel lock or any other containment penetration.

Valves required for containment isolation were included on the IPEEE SSEL and walked down and a relay evaluation performed where applicable. No concerns were identified as a result of the containment walkdowns.

Containment Bypass

Containment bypass failures made up 2.6% of the possible containment failure modes. Of this 2.6%, about one-half is from various steam generator tube failures, and the remainder is from potential interfacing-systems LOCAs (ISLOCAs). Evaluated ISLOCA events involve equipment which is normally a part of ECCS and DHR system piping and equipment. These components have been included in the IPEEE seismic SSEL, and require no further evaluation.

ntainment Cooling Systems

The containment heat removal systems consist of the Containment Air Coolers (CACs) and the Containment Spray (CS). The CACs are included on the IPEEE SSEL, and require no further evaluation. The in-containment portion of the CS consists entirely of Seismic Class I piping and spray headers/nozzles. These items are known to have high capacities and no further evaluation is required.

3.2 USI A-45, GI-131, And Other Seismic Safety Issues

NUREG-1407 lists the following programs related to seismic issues:

3.2.1 USI A-17 "Systems Interaction in Nuclear Power Plants"

DBNPS is an USI A-46 plant. The evaluation of spatial system interaction under seismic condition is subsumed by USI A-46. Therefore, this item need not be addressed further.

3.2.2 USI A-40 "Seismic Design Criteria"

The concern identified in USI A-40 have been subsumed by USI A-46. Therefore, this item need not be addressed further.

3.2.3 USI A-45 "Shutdown Decay Heat Removal Requirements"

The equipment comprising these systems were included in the SSEL and were evaluated for seismic adequacy as described in Section 3.1.4.7. No new issues separate from those previously ascertained as part of the site SQUG program were identified.

3.2.4 GI-131 "Potential Seismic Interaction Involving the Movable In-Core flux Mapping System Used in Westinghouse Plants"

Generic Issue 131 does not apply to DBNPS which has a Babcock & Wilcox NSSS.

3.2.5 The Eastern U.S. Seismicity Issue

These concerns are related to eight plant at five eastern U.S. sites, and as such, do not apply to DBNPS.

3.2.6 Seismic Induced Fires and Floods.

This issue is covered in Section 4.5 of this report as part of the Sandia Fire Risk Scoping Study evaluation.

3.3 Seismic Overview

During the design of DBNPS, conservative engineering practices were employed which resulted in a higher seismic capacity than the original design value of 0.15 pga would indicate. Some of these factors include: conservative modeling techniques which were based upon the limitations of the analysis performed and the "state of the art" at the time in computer technology; and applying the free field seismic input motion at the base of the foundation (bedrock) without using a reduction factor (i.e. deconvoluted).

The degree of conservatism in the original 0.15g SSE analysis is evident in subsequent seismic evaluations performed at DBNPS. These major undertakings include:

1. Re-evaluation of the seismic input motion from 0.15g to 0.20g
2. Generic Letter 87-02 "Verification of Seismic Adequacy of Equipment in Older Operating Nuclear Plants.

These item are addressed in further detail below.

Seismic Re-evaluation

Because of subsequent investigation of the relationship between earthquake intensity and ground acceleration. (Trifunac et. al. 1975) the NRC staff questioned whether the appropriate ground acceleration for a Modified Mercalli intensity earthquake of VII-VIII should be 0.20g instead of 0.15g. During the Advisory Committee of Reactor Safeguards (ACRS) hearings the ACRS pointed out that the Davis-Besse sign criteria were most likely more conservative than the current criteria (i.e., Regulatory Guides 1.60, _.o1, 1.92, etc.) and that possibly the Davis-Besse design for 0.15g was equivalent to a current design for 0.20g. A condition was placed in the Davis-Besse Operating License requiring the licensee to perform a seismic re-evaluation to demonstrate that the Davis-Besse design provided adequate margin for a 0.20g Maximum Possible Earthquake, using current criteria.

During the first fuel cycle, a seismic re-evaluation was performed following NRC staff guidelines. The re-evaluation, using a Maximum Possible Earthquake ground acceleration of 0.20g and current criteria, determined that there were still sufficient margins available in systems required for safe shutdown of the unit as well as systems required for continued shutdown heat removal. NUREG 1407 indicates that the ground motion should be considered at the surface in the free field, however, this analysis applied the 0.2g free field input ground motion directly at the bedrock elevation.

On May 31, 1983, the Commission issued a safety evaluation of the seismic re-evaluation which concluded that there is sufficient conservatism and margin in the piping systems, components and supports at Davis-Besse to ensure safe shutdown and continued heat removal in the event of an earthquake having a ground acceleration of 0.20g.

Generic Letter 87-02

DBNPS performed a seismic evaluation on active mechanical and electrical components using the methodology identified in the Generic Implementation Procedure (GIP) developed by SQUG and reviewed the NRC. The basis of this methodology is to provide a high degree of confidence that equipment included in the scope of USI A-46 is similar to the equipment identified in the SQUG Seismic Data Base. e equipment in the seismic data base has been subjected to actual earthquakes or shake table testing much larger then the licensing criteria for DBNPS. A conservative lower bound response spectrum (Bounding Spectrum) was developed that is bounded by these earthquakes. This Bounding Spectrum envelopes DBNPS ground response spectrum by a factor of 2.0. The Bounding Spectrum was used during the USI A-46 program in establishing the capacity of the equipment that was located below about 40 feet above grade and with a natural frequency of about 8 Hz or greater. The Bounding Spectrum has a pga value of 0.33g.

3.4 Summary of Seismic Analysis

The IPEEE Seismic program was combined with the USI A-46 (SQUG) seismic program since the majority of the components were common to both programs. The walkdown concentrated on the following areas: seismic capacity, screening caveats, anchorage, and seismic spatial interaction.

The following are brief synopses of the seismic evaluation discussed elsewhere in this report:

Cable and Conduit Raceways

In general, the cable tray and conduit raceways are well supported. As a result of the Limited Analytical Review, two supports did not meet the GIP requirements and will be corrected under the USI A-46 program.

ntrol Room Ceiling

The Control Room and the cabinet room ceilings were inspected for seismic interaction concerns as part of the USI A-46 (SQUG) program. During the walkdown, the SRT identified some loose and cracked tiles along with missing clips and support wire. These deficiencies have since been corrected under the USI A-46 program.

Equipment

All components walked down that are part of the Seismic IPEEE program were either found to meet the GIP requirements or were previously identified as Outliers as part of the USI A-46 program. Those components identified as SQUG Outliers requiring fixes will be corrected in accordance with the USI A-46 program.

Masonry Walls

All masonry walls that could impact the equipment have been evaluated and are qualified for the site SSE.

Penetrations

Containment penetrations were found to be well supported and did not pose as an interaction concern. Inflatable seals or cooling systems are not used on the equipment hatch, personnel lock, emergency personnel lock, or other containment penetrations.

Relay Review

The review performed for low seismic ruggedness relays (i.e. "Bad Actor" relays) for the equipment identified on the IPEEE Seismic Review SSEL, revealed there are no new relay chatter concerns that have not already been identified as part of the USI A-46 (SQUG) program.

Subsystems

In general, the subsystems were found to be well supported with no interaction concerns.

Analysis of Containment Performance

Containment Integrity

The vessel is considered to be seismically rugged and requires no further evaluation.

Containment Isolation

No concerns were identified as a result of the containment walkdowns.

Containment Bypass

These components have been included in the IPEEE seismic SSEL, and require no further evaluation.

'ntainment Cooling Systems

These items are known to have high capacities and no further evaluation is required.

Issues

The following is a listing of other issues that are now considered closed:

USI A-17 "Systems Interaction in Nuclear Power Plants"

USI A-40 "Seismic Design Criteria"

USI A-45 "Shutdown Decay Heat Removal Requirements"

GI-131 "Potential Seismic Interaction Involving the Movable In-Core flux Mapping System Used in Westinghouse Plants"

The Eastern U.S. Seismicity Issue

Vulnerabilities

No vulnerabilities were identified as a part of this process.

..EFERENCES FOR PART 3

1. Letter to NRC from John P. Stetz, Serial No. 2242, "Reevaluation of Seismic Scope for NRC Generic Letter Number 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," dated August 25, 1994.

2. Letter to NRC from John P. Stetz, Serial No. 2341, "Response to Generic Letter 88-20, Supplement 5, Individual Plant Examination of External Events for Severe Accident Vulnerabilities," dated November 6, 1996.

3. EPRI Report NP-6041-SL (Revision 1, Final Report), A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1), August 1991.

4. NUREG-1407, Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, June 1991.

5. "Individual Plant Examination for the Davis-Besse Nuclear Power Station," Toledo Edison Company, February 1993.

6. "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment," Seismic Qualification Utility Group Report, Revision 2, February 1992.

7. NUREG/CR-4826, Seismic Margin Review of the Maine Yankee Atomic Power Station, March 1987.

8. SSRAP Report "Use of Seismic Experience Data to Show Ruggedness of Equipment in Nuclear Power Plants," Senior Seismic Review and Advisory Panel, Rev. 4.0, February 28, 1991.

9. Serial No. 2322, "Response to Request for Additional Information on Individual Plant Examination Submittal, Davis-Besse Nuclear Power Station, Unit 1," dated September 11, 1995.

DAVIS-BESSE NUCLEAR POWER STATION

Individual Plant Examination for External Events

Seismic Evaluation Report

APPENDIX A

SEISMIC REVIEW

SAFE SHUTDOWN EQUIPMENT LIST (SSEL)

IPEEE SMA SAFE SHUTDOWN EQUIPMENT LIST

IPEEEONLY	Equip Class	Equipment ID Number	System/Equipment Description	Bldg	Elev	Room	Eval Cat.	Note	Normal State	Desired State	Pwr Reqd	Required Interconnections and Supporting Components	RC	IC	PC	DH	SU	CI
T			FUEL TRANSFER TUBE 1-1 AND FLANGE	CTM	565	UNK	S		CLS	CLS	N		0	0	0	0	0	1
T			FUEL TRANSFER TUBE 1-1 AND FLANGE	CTM	565	UNK	S		CLS	CLS	N		0	0	0	0	0	1
T			FUEL TRANSFER TUBE 1-1 AND FLANGE	CTM	603	426	S		CLS	CLS	N		0	0	0	0	0	1
T			FUEL TRANSFER TUBE 1-1 AND FLANGE	CTM	585	317A	S		CLS	CLS	N		0	0	0	0	0	1
T			FUEL TRANSFER TUBE 1-1 AND FLANGE	CTM	603	400	S		CLS	CLS	N		0	0	0	0	0	1
F	15	1N	STATION BATTERY 1N -125V dc	AUX	603	429	S		ON	ON	N		0	0	0	0	1	0
F	15	1P	STATION BATTERY 1P +125V dc	AUX	603	429	S		ON	ON	N		0	0	0	0	1	0
F	3	ABDC1	BUS C1, CUB 2 BRKR FRM BUS...	AUX	585	325	SR		OPN	OPN	N		0	0	0	0	1	0
F	3	AC-101	BREAKER FROM EDG 1	AUX	585	325	SR		OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	3	AC-105	BRKR, MUP MTR 1-1 MP37-1	AUX	585	325	SR		CLS	CLS	N		1	0	0	0	0	0
F	3	AC-105	BRKR, MUP MTR 1-1 MP37-1	AUX	585	325	SR		CLS	CLS	N		0	1	0	0	0	0
F	3	AC-107	BUS C1 CUB 7 FDR BRKR FR SWP1-1	AUX	585	325	SR		CLS	CLS	N		0	0	0	0	1	0
F	3	AC-110	BUS C1, CUB 10 BRKR TO 4.16...	AUX	585	325	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	0	1	0
F	3	AC-111	BREAKER FOR HPI PUMP 1-1	AUX	585	325	SR	6,68	OPN	CLS	Y	ELECTRICAL	0	1	0	0	0	0
F	3	AC-112	BUS C1 CBCL 12 FDR BRKR FOR DH PMP 1-1	AUX	585	325	SR		OPN	CLS	Y	ELECTRICAL	0	0	0	1	0	0
F	3	AC-112	BUS C1 CBCL 12 FDR BRKR FOR DH PMP 1-1	AUX	585	325	SR	52	OPN	CLS	Y	ELECTRICAL	0	1	0	0	0	0
F	3	AC-113	BREAKER, CC PMP MTR 1-1 MP431	AUX	585	325	SR		CLS	CLS	N		0	0	0	0	1	0
F	3	AC-1CE11	BUS C1, CUB 4-FEED BRKR FRM...	AUX	585	325	SR		CLS	CLS	N		0	0	0	0	1	0
F	3	AD-105	MUP 2 BKR	AUX	585	323	SR	57	OPN	CLS	Y	ELECTRICAL	0	1	0	0	0	0
F	8A	AF-3869	AFP 1-1 TO STEAM GEN 1-2 STOP VALVE	AUX	565	237	R		CLS	CLS	N		0	0	0	1	0	0
F	8A	AF-3870	AFP 1-1 TO STEAM GEN 1-1 STOP VALVE	AUX	565	237	R		OPN	OPN	N		0	0	0	1	0	0
F	8A	AF-608	AUX FEED TO STEAM GEN 1-1 LINE STOP VLV	AUX	585	303	R		OPN	OPN	N		0	0	0	1	0	0
F	8B	AF-6452	AFP 1-1 SOL CONTROL VALVE	AUX	565	237	SR		OPN	THR	Y	ELECTRICAL	0	0	0	1	0	0
F	2	BCE 11	BUS E1 NORM FEED BRKR FROM....	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-106	FEEDER BREAKER FOR MCC E12A	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-107	FEEDER BREAKER FOR MCC E11A	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-110	FEEDER BREAKER FOR MCC E14	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
T	2	BE-1103	BKR FOR HP-2C	AUX	565	209	SR	6	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	2	BE-1105	BKR FOR HP-2D	AUX	565	209	SR	6	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	2	BE-1106	BREAKER FOR LP INJ 1 VALVE, MVDH1B	AUX	565	209	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
T	2	BE-1108	FEEDER BREAKER FOR MCC E11B	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
F	2	BE-1109	BREAKER FOR MVMU400	AUX	565	209	SR		CLS	CLS	N		1	0	0	0	0	0
T	2	BE-1112	BKR FOR DH-9B	AUX	565	209	SR	52	OPN	CLS	Y	ELECTRICAL	0	1	0	0	0	0
F	2	BE-1120	FEEDER BREAKER FOR MCC E11B	AUX	565	209	SR		CLS	CLS	N		0	0	0	0	1	0

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F	2	BE-1121	BREAKER FOR DH PUMP 1 SUC VALVE FRM BWST	AUX	565	209	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	2	BE-1126	BREAKER FOR DH NORM SUC LINE 1 ISO VLV	AUX	565	227	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	2	BE-1127	BKR FOR MU 6405	AUX	565	227	SR		CLS	CLS	N		0	1	0	0	0	0
F	2	BE-1127	BKR FOR MU 6405	AUX	565	227	SR		CLS	CLS	N		1	0	0	0	0	0
T	2	BE-1135	BKR FOR MOV SW-5422	AUX	565	209	SR		CLS	CLS	N		0	0	0	0	1	0
T	2	BE-1136	BKR FOR MOV SW-5421	AUX	565	209	SR		CLS	CLS	N		0	0	0	0	1	0
T	2	BE-1137	BKR FOR CV-5070	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	BE-1138	BKR FOR CV-5071	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	BE-1139	BKR FOR CV-5072	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	BE-1140	BKR FOR CV-5073	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	BE-1141	BKR FOR CV-5074	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
F	2	BE-1144	BRKR, CTRM EMERG VNT FAN1..VLV	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	1	0
T	2	BE-1145	BKR FOR CV-645B	AUX	603	402	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	BE-1147	BKR FOR MU-6409	AUX	565	227	SR		CLS	CLS	N		0	1	0	0	0	0
F	2	BE-1148	BRKR, CTRM EMERG STND BYPAS...	AUX	603	402	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1150	FEEDER BREAKER TO MCC E11E	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1151	BREAKER FOR FEED FROM MCC E11C	AUX	603	402	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1154	BRKR, CC PMP RM VNT FAN 1...	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	1	0
T	2	BE-1158	FEEDER BREAKER FOR MCC E11B	AUX	585	304	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	2	BE-1159	BKR FOR FW-612	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
F	2	BE-1162	BKR FOR CFT 1 ISOLATION VALVE	AUX	585	304	S	40	OPN	OPN	N		0	1	0	0	0	0
F	2	BE-1162	BKR FOR CFT 1 ISOLATION VALVE	AUX	585	304	S	18	OPN	CLS	Y	MANUAL	0	0	0	1	0	0
F	2	BE-1166	BRKR FOR FEED TO MCC E11B	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	1	0
T	2	BE-1171	FEEDER BREAKER FOR MCC E11B	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
F	2	BE-1172	RC LETDOWN ISO VALVE	AUX	585	304	SR		CLS	CLS	N		0	1	0	0	0	0
T	2	BE-1173	FEEDER BREAKER FOR MCC E11B	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	BE-1174	FEEDER BREAKER FOR MCC E11B	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	BE-1175	FEEDER BREAKER FOR MCC E11B	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	BE-1176	FEEDER BREAKER FOR MCC E11B	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	BE-1177	FEEDER BREAKER FOR MCC E11B	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	BE-1178	FEEDER BREAKER FOR MCC E11B	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
F	2	BE-1180	BREAKER FOR XYE2 FDR TO MCCYE2	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1183	BREAKER FOR DH REMOVAL SUCTION LINE VLV	AUX	585	304	SR		OFF	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	2	BE-1183	BREAKER FOR DH REMOVAL SUCTION LINE VLV	AUX	585	304	SR		CLS	CLS	N		0	0	1	0	0	0

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F	2	BE-1185	BREAKER FOR BA PUMP 1 MP381	AUX	565	227	SR		CLS	CLS	N		1	0	0	0	0	0
T	2	BE-1187	BKR FOR DH-64	AUX	603	402	SR	52	CLS	CLS	N		0	1	0	0	0	0
F	2	BE-1191	BKR FOR MUP 1-1 MN OIL P-371B	AUX	565	227	SR		CLS	CLS	N		0	1	0	0	0	0
F	2	BE-1191	BKR FOR MUP 1-1 MN OIL P-371B	AUX	565	227	SR		CLS	CLS	N		1	0	0	0	0	0
F	2	BE-1192	BKR FOR P-371D	AUX	565	227	SR		CLS	CLS	N		0	1	0	0	0	0
F	2	BE-1192	BKR FOR P-371D	AUX	565	227	SR		CLS	CLS	N		1	0	0	0	0	0
F	2	BE-1194	BKR FOR MU-6421	AUX	565	227	SR	35	CLS	CLS	N		0	1	0	0	0	0
F	2	BE-1196	BREAKER FR FEEDER FRM MCC E11A	AUX	565	227	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1201	BRKR, CR EMERG SYS STANDBY...	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1202	FEEDER BREAKER TO MCC E12C	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1205	BRKR, SW PMP VENT FAN 2 MC99-2	ITK	575	051	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1208	BRKR, BAT RM 429B- ATM DAMP MO	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1209	BRKR, CTRL RM EMERG VENTILATN	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1212	BRKR, SW PMP VENT FAN 1 MC99-1	ITK	575	051	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1216	BRKR, CTRM EMERG COND UNT1 MTR	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1217	BRKR, VNT FN 1 MTR L.V.S.G. RM	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1218	BREAKER FOR AFP 1 SUCTION VALVE MV1382	AUX	603	429	SR		CLS	CLS	N		0	0	0	1	0	0
F	2	BE-1222	BREAKR FOR AFP ROOM VENT FAN 1 MOTOR	AUX	603	429	SR		CLS	CLS	N		0	0	0	1	0	0
F	2	BE-1223	FEEDER BRKR FOR PRZR HTRS CH 1	AUX	603	429	SR		CLS	CLS	N		0	0	1	0	0	0
F	2	BE-1226	BREAKER, CCW DISCH LN ISO VLV	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1233	BREAKER FOR BATT CHARGER DBC1P	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1234	FEEDER BREAKER FOR MCC E12E	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1235	BREAKER FOR BATT CHARGER DBC1N	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1240	BRKR, L.V.S.G. RM 429 VENT VALVE	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1241	BRKR, L.V.S.G. RM 429 VENT VALVE	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1255	BRKR, EDG RM 1 VENT FAN 1	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1256	BRKR, EDG RM 1 VENT FAN 2	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1258	EDG 1 IMMERSION HEATER BREAKER	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1259	BRKR, FDR TO 120VAC MCC YE1	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1261	BRKR, EDG SOAK PMP MP1471	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1273	FEEDER BREAKER FOR MCC E12F	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1274	BREAKER, SW PUMPSTRNR MF12-1	ITK	585	051	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1275	BRKR, SW PMP STRNR DRAIN VALVE	ITK	585	051	SR		CLS	CLS	N		0	0	0	0	1	0
T	2	BE-1277	BKR FOR SW ISOL VALVE - COOLING WATER	ITK	585	051	SR		CLS	CLS	N		0	0	0	0	1	0

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F	2	BE-1281	BREAKER, SW - INTAKE STRCT VLV	ITK	585	051	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1282	BREAKER, SW TO CLNG TWR MU VLV	ITK	585	051	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1284	BREAKER FR FEEDER FRM MCC E12A	ITK	585	051	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1285	BRKR, BAT RM VENT FAN 1-1	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
T	2	BE-1286	BKR FOR HP-32	AUX	545	101	SR	2	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	2	BE-1289	BRKR, EDG1 AC TURBO OIL PMP MO	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1291	BREAKER FOR FEEDER TO MCC E12A	AUX	545	101	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1292	BRKR FOR C31-4	AUX	545	101	SR		CLS	CLS	N		0	0	0	1	0	0
F	2	BE-1293	BRKR FOR C31-5	AUX	545	101	SR		CLS	CLS	N		0	0	0	1	0	0
F	2	BE-1295	BKR FOR MU-6419	AUX	545	101	SR	35	CLS	CLS	N		0	1	0	0	0	0
T	2	BE-1296	BKR FOR P197-1	AUX	545	101	SR	6	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	2	BE-1297	FEEDER BREAKER TO MCC E12F	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1298	BRKR, EDG FUEL OIL STRG & XFER	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	BE-1401	HI & LO SPD STARTER FR CTMT AIR CLR FAN1	AUX	603	429	SR		CLS	CLS	N		0	0	1	0	0	0
F	14	BE12	ESNTL PZR HTR BNK 1 SPLY PNL	AUX	603	429	SR	63	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	2	BF-1126	BREAKER FOR PRZR VAPOR SAMPLE LINE VALVE	AUX	603	427	SR	60	CLS	CLS	N		0	0	1	0	0	0
F	2	BF-1130	BREAKER FOR DH REMOVAL SUCTION LN VALVE	AUX	603	427	SR	86	CLS	CLS	N		0	0	1	0	0	0
F	2	BF-1285	BREAKER FOR PRZR SMPL LINE TO...HDR VLV	AUX	603	428	SR	60	CLS	CLS	N		0	0	1	0	0	0
F	2	BF-1617	BKR FOR MU 3971	AUX	603	428	SR	57	CLS	CLS	N		0	1	0	0	0	0
F	2	BRKR-C	CRDM TRIP BRKR-C C4612	AUX	603	428	SR		CLS	OPN	Y	ELECTRICAL	1	0	0	0	0	0
F	2	BRKR-D	CRDM TRIP BRKR-D C4806	AUX	603	402	SR		CLS	OPN	Y	ELECTRICAL	1	0	0	0	0	0
F	3	C1	4.16KV BUS	AUX	585	325	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	10	C1-1	CAC 1-1 (AIR SIDE FUNCTION)	CTM	565	217	SR	66	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	10	C21-1	CNTRL RM EMERG VENT SYS FAN1-1	AUX	638	603	SR	36	OFF	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	9	C25-1	SUPPLY FAN 1-1	AUX	585	318	SR	36	OFF	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	9	C25-2	SUPPLY FAN 1-2	AUX	585	318	SR	36	OFF	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	C3017	SW STRNR 1-1 DRAIN/BCKWASH VLV	ITK	585	052	SR		AUT	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	10	C31-4	ECCS RM CLR 1-4 FAN	AUX	545	105	SR	36	STB	O/O	Y	ELECTRICAL	0	0	0	1	0	0
F	10	C31-5	ECCS RM CLR 1-5 FAN	AUX	545	105	SR	36	STB	O/O	Y	ELECTRICAL	0	0	0	1	0	0
F	20	C3615	EDG 1 CONTROL PANEL	AUX	585	318	SR	102	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	C3617	EDG 1-1 STATIC EXCITER VOLT REG PANEL	AUX	585	318	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	C3621	EDG 1-1 ENGINE MNTD CTRL PNL	AUX	585	318	SR	46	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	C3621A	EDG 1-1 IDLE START/STOP CONTROL PANEL	AUX	585	318	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	C3630	SFRCS CONTROL CABINET	AUX	585	324	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0

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F	20	C3812	CABINET FOR RCS TEMP LOOP 1 (TI-5504)	AUX	589	303	S		CLS	OPN	N		1	0	1	1	0	0
F	20	C5755C	SFAS CHANNEL 2	AUX	623	502	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	C5756D	SFAS CHANNEL 4	AUX	623	502	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	C5761A	SFRCS ACTUATION CHANNEL 1	AUX	623	502	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	C5762A	SFRCS ACTUATION CHANNEL 1	AUX	623	502	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	C5762C	SFAS CHANNEL 1	AUX	623	502	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	C5762C	SFAS CHANNEL 1	AUX	623	502	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	C5763D	SFAS CHANNEL 3	AUX	623	502	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	9	C71-1	L.V.S.G. RM VENT FAN 1-1	AUX	603	429	SR	36	STB	O/O	Y	ELECTRICAL	0	0	0	0	1	0
F	9	C73-1	AFP ROOM EXHAUST FAN	AUX	565	237	SR	92	STB	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	9	C75-1	CC PMP RM VENT FAN 1-1	AUX	585	328	SR	36	O/O	O/O	Y	ELECTRICAL	0	0	0	0	1	0
F	9	C78-1	BATTERY ROOM VENT FAN 1-1	AUX	603	429	SR	36	STB	O/O	Y	ELECTRICAL	0	0	0	0	1	0
F	9	C99-1	EXHAUST FAN 1-1	ITK	585	52A	SR	36	O/O	O/O	Y	ELECTRICAL	0	0	0	0	1	0
F	9	C99-2	EXHAUST FAN 1-2	ITK	585	52A	SR	36	O/O	O/O	Y	ELECTRICAL	0	0	0	0	1	0
T	8A	CC-1407A	CCW RETURN ISO FROM LETDOWN COOLERS	CTM	585	315	SR	43	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	7	CC-1411A	CCW SUPPLY ISOLATION	CTM	585	315	SR	43	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
F	7	CC-1467	CCW FRM DH RMVL CLR 1-1...VLV	AUX	545	113	SR	33	CLS	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	7	CC-1471	CC FRM EDG 1-1 SOL OUTLET VLV	AUX	585	318	SR	33	CLS	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
T	7	CC-1567A	CCW SUPPLY ISOLATION FOR CRD COOLING	CTM	585	315	SR	43	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
F	8A	CC-5095	CC LN 1 DISCH ISO VALVE	AUX	585	328	SR		OPN	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	4	CE1-1	4.16kV-480V TRANSFORMER	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T		CF-1542	CFT VENT ISOLATION	AUX	585	314	R		CLS	CLS	N		0	0	0	0	0	1
T	8A	CF-1B	CORE FLOOD TANK 1-1 ISOLATION	CTM	565	214	S	18,40	OPN	OPN	N		0	1	0	0	0	0
T	8A	CF-1B	CORE FLOOD TANK 1-1 ISOLATION	CTM	565	214	S	18	OPN	CLS	Y	ELECTRICAL	0	0	0	1	0	0
T		CF-2A	CFT 1-2 DRAIN ISOLATION	CTM	565	217	R		CLS	CLS	N		0	0	0	0	0	1
T		CF-2B	CFT 1-1 DRAIN ISOLATION	CTM	565	214	R		CLS	CLS	N		0	0	0	0	0	1
T		CF-2B	CFT 1-1 DRAIN ISOLATION	CTM	565	214	R		CLS	CLS	N		0	1	0	0	0	0
T		CF-5A	CFT 1-2 VENT ISOLATION	CTM	585	317	R		CLS	CLS	N		0	0	0	0	0	1
T		CF-5B	CFT 1-1 VENT ISOLATION	CTM	585	316	R		CLS	CLS	N		0	0	0	0	0	1
T		CS-1530	CTMT SPRAY DISCH ISO VALVE TRN 1	AUX	585	303	R		CLS	CLS	N		0	0	0	0	0	1
T		CS-1530	CTMT SPRAY DISCH ISO VALVE TRN 1	AUX	585	303	S	16	CLS	CLS	N		0	1	0	0	0	0
T		CS-1531	CTMT SPRAY DISCH ISO VALVE TRN 2	AUX	585	314	R		CLS	CLS	N		0	0	0	0	0	1
F	8A	CV-2000B	RPS SFAS CH1 CTMT PRESS SWT CTMT ISO VLV	AUX	585	303	R		OPN	OPN	N		0	0	1	0	0	0
F	8A	CV-2002B	RPS SFAS CH3 CTMT PRESS SWT CTMT ISO VLV	AUX	603	402	R		OPN	OPN	N		0	0	1	0	0	0

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T		CV-5005	PURGE VALVE ISOLATION	AUX	643	600	R		CLS	CLS	N		0	0	0	0	0	1
T		CV-5006	PURGE VALVE ISOLATION	CTM	643	UNK	R	48	CLS	CLS	N		0	0	0	0	0	1
T		CV-5007	PURGE VALVE ISOLATION	CTM	603	UNK	R	48	CLS	CLS	N		0	0	0	0	0	1
T		CV-5008	PURGE VALVE ISOLATION	AUX	603	427	R		CLS	CLS	N		0	0	0	0	0	1
T	8	CV-5010A	P71B ISOLATION	CTM	585	316	SR	43	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	8	CV-5010C	P73B ISOLATION	CTM	603	410A	SR	43	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	8	CV-5011A	P71B ISOLATION	AUX	585	303	SR	43	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	8	CV-5011B	P68B ISOLATION	CTM	603	410	SR	43	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	8	CV-5011C	P73B ISOLATION	AUX	603	402	SR	43	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	8	CV-5011D	P74B ISOLATION	CTM	585	317	SR	43	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	8	CV-5011E	P43B ISOLATION	AUX	585	314	SR	43	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T		CV-5037	H2 PURGE VALVE ISOLATION	AUX	565	236	R	22	CLS	CLS	N		0	0	0	0	0	1
T		CV-5038	H2 PURGE VALVE ISOLATION	AUX	565	236	R		CLS	CLS	N		0	0	0	0	0	1
T	8	CV-5070	MOTOR OPERATED BUTTERFLY VALVE	CTM	623	ANN	S	42	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	8	CV-5071	MOTOR OPERATED BUTTERFLY VALVE	CTM	623	ANN	S	42	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	8	CV-5072	MOTOR OPERATED BUTTERFLY VALVE	CTM	623	ANN	S	42	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	8	CV-5073	MOTOR OPERATED BUTTERFLY VALVE	CTM	623	ANN	S	42	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	8	CV-5074	MOTOR OPERATED BUTTERFLY VALVE	CTM	623	ANN	S	42	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	8	CV-645B	DIFFERENTIAL PRESSURE ISOLATION VALVE	AUX	603	402	SR		OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
F	3	D1	4.16KV BUS	AUX	585	323	SR	57	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	2	D101	BRKR FOR +125VDC DIST PNL D1P	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	D102	BRKR FOR +125VDC DIST PNL D2P	AUX	603	429	SR	41	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	2	D103	BREAKER FOR + SUPPLY FRM DBC1P	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	D104	BREAKER FOR STATION BATT 1P	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	D111	BRKR FR EMERG LIGHT XFER SWT 1	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	D112	BRKR FR EMERG LIGHT XFER SWT 3	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	D116	BREAKER FOR INVERTER YVA	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	D117	BKR FOR MUP 1-1 DC OIL PMP P-371C	AUX	603	429	SR		CLS	CLS	N		0	1	0	0	0	0
F	2	D117	BKR FOR MUP 1-1 DC OIL PMP P-371C	AUX	603	429	SR		CLS	CLS	N		1	0	0	0	0	0
F	2	D131	BREAKER FOR STATION BATT 1N	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	D132	BRKR FOR -125VDC DIST PNL	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	D133	BRKR FOR -125VDC DIST PNL D2N	AUX	603	429	SR	41	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	2	D134	BRKR FOR - SUPPLY FROM DBC1N	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	D135	BREAKER FOR AFP TURB 1 MS INLT ISO VALVE	AUX	603	429	SR		CLS	CLS	N		0	0	0	1	0	0

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F	2	D145	BRKR FOR D1NA	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	14	D1N	ESSEN DIST PNL "D1N"	AUX	603	429A	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	2	D1N 01	BREAKER FOR INCOMING DC MCC 1	AUX	603	429A	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	D1N 03	BREAKER FOR INVERTER YV3	AUX	603	429A	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	D1NA	ESSENTIAL -125VDC DIST PNL CH1	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	14	D1P	ESSEN DIST PNL "D1P"	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	2	D1P 01	BREAKER FOR DC MCC 1	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	D1P 03	BREAKER FOR INVERTER YV1	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	20	D1P09	DISC SW FOR EDG 1-1 FUNCTION C3615	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	D1P20	CIRCUIT D1P20	AUX	603	429	SR		CLS	CLS	N		0	0	0	1	0	0
F	14	D2N	ESSEN DIST PNL "D2N"	AUX	603	428	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	2	D2N 02	BREAKER FOR DC MCC D2	AUX	603	428	SR	41	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	2	D2N 03	BREAKER FOR INVERTER YV4	AUX	603	428	SR		CLS	CLS	N		0	0	0	0	1	0
F	14	D2P	ESSNTL +125VDC DISTBTN PNL CH2	AUX	603	428	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	2	D2P 02	BREAKER FOR DC MCC 1 TO D2P	AUX	603	428	SR	41	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	2	D2P 03	BREAKER FOR INVERTER YV2	AUX	603	428	SR		CLS	CLS	N		0	0	0	0	1	0
F	8B	DA-3783	EDG AIR RCVR 1-1-1 TO AIR..VLV	AUX	585	318	SR		OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	8B	DA-3784	EDG AIR RCVR 1-1-2 TO AIR..VLV	AUX	585	318	SR		OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	16	DBC1N	BATTERY CHARGER -125V dc	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	16	DBC1P	BATT CHARGER FOR BATT 1P +125V	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	14	DC MCC-1	DC BUS TRAIN 1	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	8A	DH-11	RCS TO DH SYSTEM ISO VALVE	CTM	565	290	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	1	0	0
F	8A	DH-11	RCS TO DH SYSTEM ISO VALVE	CTM	565	290	SR	64	CLS	OPN	Y	ELECTRICAL	0	0	1	0	0	0
F	8A	DH-11	RCS TO DH SYSTEM ISO VALVE	CTM	565	290	SR		CLS	CLS	N		1	0	0	0	0	0
F	8A	DH-11	RCS TO DH SYSTEM ISO VALVE	CTM	565	290	SR		CLS	CLS	N		0	1	0	0	0	0
F	8A	DH-12	RCS TO DH SYSTEM ISO VALVE	CTM	565	290	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	1	0	0
F	8A	DH-12	RCS TO DH SYSTEM ISO VALVE	CTM	565	290	SR		CLS	OPN	Y	ELECTRICAL	0	0	1	0	0	0
F	8A	DH-12	RCS TO DH SYSTEM ISO VALVE	CTM	565	290	SR		CLS	CLS	N		0	1	0	0	0	0
F	7	DH-13B	DH COOLER 1-1 BYPASS FLOW CTRL VALVE	AUX	545	113	R		CLS	CLS	N		0	0	0	1	0	0
F	7	DH-14B	DH COOLER 1-1 OUTLET FLOW CTRL VALVE	AUX	545	113	R		OPN	OPN	N		0	0	0	1	0	0
F		DH-14B	DH COOLER 1-1 OUTLET FLOW CTRL VALVE	AUX	545	113	R	16	OPN	OPN	N		0	1	0	0	0	0
F		DH-14B	DH COOLER 1-1 OUTLET FLOW CTRL VALVE	AUX	545	113		16	OPN	OPN	N		1	0	0	0	0	0
F	8A	DH-1517	DH PUMP 1-1 SUCTION FROM RCS VALVE	AUX	565	236	SR	88	CLS	OP/CL	Y	ELECTRICAL	0	0	0	1	0	0
F	8A	DH-1517	DH PUMP 1-1 SUCTION FROM RCS VALVE	AUX	565	236	S	16	CLS	CLS	N		0	1	0	0	0	0

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F	8A	DH-1517	DH PUMP 1-1 SUCTION FROM RCS VALVE	AUX	565	236	S	16	CLS	CLS	N		1	0	0	0	0	0
F	8A	DH-1518	DH PUMP 1-2 SUCTION FROM RCS	AUX	565	236	SR		CLS	OP/CL	Y	ELECTRICAL	0	0	0	1	0	0
F	8A	DH-1B	DH COOLER 1-1 DISCH TO RCS ISO VALVE	AUX	565	208	SR	87	OPN	THR	Y	ELECTRICAL	0	0	0	1	0	0
F	8A	DH-1B	DH COOLER 1-1 DISCH TO RCS ISO VALVE	AUX	565	208	S	16	OPN	OPN	N		0	1	0	0	0	0
F	8A	DH-1B	DH COOLER 1-1 DISCH TO RCS ISO VALVE	AUX	565	208	S	16	OPN	OPN	N		1	0	0	0	0	0
F	R	DH-21	MANUAL VALVE	CTM	565	220		80	CLS	OP/CL	Y	MANUAL	0	0	0	1	0	0
F	R	DH-23	MANUAL VALVE	CTM	565	220		80	CLS	OP/CL	Y	MANUAL	0	0	0	1	0	0
F	R	DH-26	DH PMP 1-2 MIN COOLDOWN ISO VALVE	AUX	565	236		83	OPN	OP/CL	Y	MANUAL	0	0	0	1	0	0
F	8A	DH-2733	DH PMP 1-1 SUC (BWST OR EMER SUMP) VLV	AUX	545	105	SR		OPN	CLS	Y	ELECTRICAL	0	0	0	1	0	0
F	8A	DH-2733	DH PMP 1-1 SUC (BWST OR EMER SUMP) VLV	AUX	545	105	R		OPN	OPN	N		0	1	0	0	0	0
F	8A	DH-2733	DH PMP 1-1 SUC (BWST OR EMER SUMP) VLV	AUX	545	105	S		OPN	OPN	N		1	0	0	0	0	0
T		DH-2735	DH AUX SPRAY ISOLATION	CTM	603	410	R		CLS	CLS	N		0	0	0	0	0	1
F		DH-2736	DH AUX SPRAY ISOLATION	AUX	484	314	R		CLS	CLS	N		0	0	0	0	0	1
F	8A	DH-2736	DH AUX SPRAY ISOLATION	AUX	484	314	R		CLS	CLS	N		1	0	0	0	0	0
F	8A	DH-2736	DH AUX SPRAY ISOLATION	AUX	484	314	R		CLS	CLS	N		0	0	0	1	0	0
F	7	DH-4849	DH COOLDOWN LN RELIEF FOR LTOP	CTM	565	220	S	68	CLS	OP/CL	N		0	0	1	0	0	0
F	8A	DH-64	DH PMP 1-1 DISCH TO HPI PMP 1-1 SUC VLV	AUX	545	105	R		CLS	CLS	N		0	0	0	1	0	0
F	8A	DH-64	DH PMP 1-1 DISCH TO HPI PMP 1-1 SUC VLV	AUX	545	105		16	CLS	CLS	N		1	0	0	0	0	0
F	8A	DH-64	DH PMP 1-1 DISCH TO HPI PMP 1-1 SUC VLV	AUX	545	105	SR	52	CLS	OPN	Y	ELECTRICAL	0	1	0	0	0	0
F	8A	DH-7B	BWST ISO VALVE (LN 1)	YRD	585	901	R		OPN	OPN	N		1	0	0	0	0	0
F		DH-7B	BWST ISO VALVE (LN 1)	AUX	585	901	R		OPN	OPN	N		0	1	0	0	0	0
F	8A	DH-831	DH COOLER 1-1/1-2 XCONNECTION VALVE	AUX	545	113	R		CLS	CLS	N		0	0	0	1	0	0
F	8A	DH-831	DH COOLER 1-1/1-2 XCONNECTION VALVE	AUX	545	113	R		CLS	CLS	N		1	0	0	0	0	0
F	8A	DH-9B	DHP1-1 SUCT FRM EMER SUMP VLV	AUX	545	105		18	CLS	CLS	N		1	0	0	0	0	0
F	8A	DH-9B	DHP1-1 SUCT FRM EMER SUMP VLV	AUX	545	105	SR	52	CLS	OPN	Y	ELECTRICAL	0	1	0	0	0	0
T	8	DR-2012A	NORMAL SUMP ISOLATION	CTM	565	292	SR	43	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	8	DW-6831A	DEMINERALIZED WATER ISOLATION	CTM	585	316	SR	48	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
F	2	E1	480V ESSENTIAL UNIT SUBSTATION	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	21	E10-1	EMERG DIESEL GEN JCKT HT XCHNG	AUX	585	318	S		ON	ON	N		0	0	0	0	1	0
F	10	E106-1	COOLING COIL 1-1	AUX	638	603	S	34	ON	ON	N		0	0	0	0	1	0
F	1	E11A	480V ESSENTIAL MCC	AUX	565	209	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	1	E11B	480V ESSENTIAL MCC	AUX	585	304	SR	86	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	1	E11C	480V ESSENTIAL MCC	AUX	585	304	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	1	E11D	480V ESSENTIAL MCC	AUX	565	227	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0

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F	1	E11E	480V ESSENTIAL MCC	AUX	603	402	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	1	E12A	480V ESSENTIAL MCC	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	1	E12B	480V ESSENTIAL MCC	AUX	585	318	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	1	E12C	480V ESSENTIAL MCC	ITK	576	51	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	1	E12E	480V ESSENTIAL MCC	AUX	545	101	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	1	E12F	480V ESSENTIAL MCC	AUX	585	318	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	1	E14	480V ESSENTIAL MCC	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T	5	E188-1	MAKEUP GEAR LUBE OIL COOLER	AUX	565	225	S	47	ON	ON	N		1	0	0	0	0	0
T	5	E188-1	MAKEUP GEAR LUBE OIL COOLER	AUX	565	225	S	47	ON	ON	N		0	1	0	0	0	0
F	R	E197-1	CONT GAS ANAL SYS HT XCHNGR1-1	AUX	585	304			ON	ON	N		0	0	0	0	1	0
F	21	E198-1	BEARING OIL COOLER FOR HPI PUMP 1-1	AUX	545	105	S	6	ON	ON	N		0	0	0	0	1	0
T	5	E212-1	MAKEUP LUBE OIL COOLER	AUX	565	225	S	47	ON	ON	N		1	0	0	0	0	0
T	5	E212-1	MAKEUP LUBE OIL COOLER	AUX	565	225	S	47	ON	ON	N		0	1	0	0	0	0
F	21	E22-1	COMP. COOLING HEAT EXCHANGER 1-1	AUX	585	328	S		ON	ON	N		0	0	0	0	1	0
F	21	E22-3	COMP. COOLING HEAT EXCHANGER 1-3	AUX	585	328	S		ON	N/A	N		0	0	0	0	1	0
T	R	E24-1	STEAM GENERATOR 1-1	CTM	565	216			ON	ON	N		0	1	0	0	0	0
T	R	E24-1	STEAM GENERATOR 1-1	CTM	565	216			ON	ON	N		0	0	0	1	0	0
T	R	E24-2	STEAM GENERATOR 1-2	CTM	565	218			ON	ON	N		0	1	0	0	0	0
F	21	E26-1	SEAL RETURN COOLER	AUX	565	208	S		OFF	OFF	N		0	1	0	0	0	0
F	21	E26-1	SEAL RETURN COOLER	AUX	565	208	S		OFF	OFF	N		1	0	0	0	0	0
F	21	E26-2	SEAL RETURN COOLER	AUX	565	208	S		ON	ON	N		0	1	0	0	0	0
F	21	E26-2	SEAL RETURN COOLER	AUX	565	208	S		ON	N/A	N		1	0	0	0	0	0
F	21	E27-1	DECAY HEAT REMOVAL COOLER 1-1	AUX	545	113	S	79	ON	ON	N		0	0	0	1	0	0
F	21	E27-1	DECAY HEAT REMOVAL COOLER 1-1	AUX	545	113	S		ON	N/A	N		1	0	0	0	0	0
F	21	E27-1	DECAY HEAT REMOVAL COOLER 1-1	AUX	545	113	S		ON	ON	N		0	1	0	0	0	0
F	21	E34	BWST HEATER (HX)	AUX	565	209	S	99	ON	N/A	N		1	0	0	0	0	0
F	21	E34	BWST HEATER (HX)	AUX	565	209	S	99	ON	N/A	N		0	1	0	0	0	0
F	10	E37-1	CTMT AIR COOLER 1-1	CTM	585	317	S		ON	ON	N		0	0	1	0	0	0
F	10	E37-1	CTMT AIR COOLER 1-1	CTM	585	317	S	79	ON	ON	N		0	0	0	0	1	0
F	10	E37-1	CTMT AIR COOLER 1-1	CTM	585	317	S		ON	ON	N		0	0	1	0	0	0
F	10	E37-3	CAC COIL 1-3 (SW SIDE)	CTM	585	317	S		OFF	OFF	N		0	0	0	0	1	0
F	10	E42-4	ECCS ROOM COOLER COIL 1-4	AUX	545	105	S		ON	ON	N		0	0	0	0	1	0
F	10	E42-5	ECCS ROOM COOLER COIL 1-5	AUX	545	105	S		ON	ON	N		0	0	0	0	1	0
F	20	EI-4553	BUS Y1A VOLTMETER	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0

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F	20	EI-4554	BUS Y2A VOLTMETER	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	EI-6271	BUS D1P VOLTMETER	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	EI-6272	BUS D2N VOLTMETER	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	EI-6273	BUS 1P-BUS INCOMING VOLTMETER	AUX	623	502	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	EI-6275	BUS D1N VOLTMETER	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	EI-6276	BUS D2P VOLTMETER	AUX	623	502	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	EI-6277	BUS Y1 VOLTMETER	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	EI-6278	BUS Y4 VOLTMETER	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	EI-6281	BUS Y3 VOLTMETER	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	EI-6282	BUS Y2 VOLTMETER	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	EI-6297	BUS YAU VOLTMETER	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T	0	F108-1	EDG 1-1 INTAKE FILTER	AUX	585	318	S		OPN	OPN	N		0	0	0	0	1	0
F	1	F11A	480V ESSENTIAL MCC	AUX	427	427	SR	60	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	1	F12A	480V ESSENTIAL MCC	AUX	603	428	SR	60	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	0	F15-1	SERVICE WATER STRAINER 1-1	ITK	585	052	SR		STB	O/O	Y	ELECTRICAL	0	0	0	0	1	0
F.	1	F16A	480V ESSENTIAL MCC	AUX	603	428	SR	57	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T	20	FI 6425	MU FLOW INDICATION FOR INJ LINE C	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	FI MU31	MU FLOW INDICATION FOR INJ LINE A	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	18	FIS 1422C	CC PMP 1-1 DISCH FLOW INDIC SW	AUX	585	328	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T	18	FT 6425	MU FLOW TRANSMITTER FOR INJ LINE C	AUX	545	105	SR	1	ON	ONY			0	1	0	0	0	0
F	18	FT DH2B	LP INJ LINE 1 FLOW TRANSMITTER	AUX	545	105	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
T	18	FT HP3C	FLOW TRANSMITTER FOR HP-2C	AUX	565	208	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	18	FT HP3D	FLOW TRANSMITTER FOR HP-2D	AUX	565	208	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	18	FT MU34	MU FLOW TRANSMITTER FOR INJ LINE A	AUX	565	225	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	FYI-DH2B	LP INJECTION LINE 1 FLOW RELAY INDICATOR	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
T	20	FYI-HP3C	FLOW INDICATOR FOR HP-2C	AUX	623	505	SR	6	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	FYI-HP3D	FLOW INDICATOR FOR HP-2D	AUX	623	505	SR	6	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
f	R	HA-15	AIR COOLED COND UNIT 1 OUTLET ISOL VLV	AUX	638	603			CLS	OPN	Y	MANUAL	0	0	0	0	1	0
f	R	HA-17	AIR COOLED COND UNIT 1 INLET ISOL VLV	AUX	638	603			CLS	OPN	Y	MANUAL	0	0	0	0	1	0
F	20	HIS 100B	SFRCS CH 2 BLOCK SW	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS 100C	SFRCS CH 4 BLOCK SW	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS 101	HS FOR MSIV 101	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS 101B	SFRCS CH 1 BLOCK SW	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS 101C	SFRCS CH 3 BLOCK SW	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0

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F	20	HIS 106A	AFP TURB 1-1 MS ISO VALVE SG 1-1 HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS 1356	CTMT CLR 1 SW OUTLET VALVE HIS IN C5716	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 1370	SW PMP 1 HAND INDIC SWITCH	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 1382	HIS FOR ISO VALVE SW1382	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS 1382B	HIS FOR ISO VALVE SW1382 LOC IN C3630	AUX	585	324	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS 1399	SW TO CLNG WTR HDR HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T	20	HIS 1407A	HS FOR CC-1407A	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS 1411A	HS FOR CC-1411A	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
F	20	HIS 1414	CCW PMP 1 HAND INDIC SWITCH	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 1424	CC HX 1 SW OUT ISO VLV HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 1467	DH RMVL CLR 1 CCW OUT HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 1471	EDG 1 CCW OUT HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 1517	DH PMP 1-1 NORM SUC ISO VLV HIS IN C5704	AUX	623	505	SR	88	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
T	20	HIS 1524	HS FOR HPI PUMP 1-1	AUX	623	505	SR	6	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	HIS 1567A	HS FOR CC-1567A	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS 1719A	HS FOR RC-1719A	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS 1773A	HS FOR RC-1773A	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
F	20	HIS 200A	PRZR VNT VLV TO CTMT VNT VLV HIS	AUX	623	505	SR	60	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
T	20	HIS 2012A	HS FOR DR-2012A	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS 229A	HS FOR RC-229A	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS 236	HS FOR NN-236	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
F	20	HIS 239A	PRZR VAPOR SAMPLE ISO VALVE HIS	AUX	623	505	SR	60	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	HIS 2733	DH PMP 1-1 SUCT FRM LP INJ LINE HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
T	20	HIS 2733A	SFAS LEVEL 3 BLOCK SWITCH	AUX	623	505	SR	54	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS 2927	CTRM EMERG COND 1 SW OUTLT VLV	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 2929	SW TO INTAKE STRUCTURE VLV HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 2931	SW TO CLNG TWR MU VLV HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 3971	HS FOR MU 3971	AUX	623	505	SR	57	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS 4823	CTRL RM EMER SYS COND 1 IN HIS (C6708)	AUX	643	603	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 4823A	CTRL RM EMER SYS COND 1 IN HIS	AUX	643	603	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 4824	CTRL RM EMER SYS COND 1 OUT HIS (C6708)	AUX	643	603	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T	20	HIS 5011A	HS FOR CV-5011A	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS 5011B	HS FOR CV-5011B	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS 5011C	HS FOR CV-5011C	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1

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T	20	HIS 5011D	HS FOR CV-5011D	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS 5011E	HS FOR CV-5011E	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
F	20	HIS 5031	CTMT COOLER FAN 1 HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
T	20	HIS 5070	HS FOR CV-5070	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS 5071	HS FOR CV-5071	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS 5072	HS FOR CV-5072	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS 5073	HS FOR CV-5073	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS 5074	HS FOR CV-5074	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
F	20	HIS 5095	CCW LN 1 TO NON-ESSEN HDR HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 520A	AFP 1-1 GOV CTRL HIS, LOC IN C5709	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS 5261	EMERG VENT FAN 1-1 HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 5261A	HIS FR EMERG VENT FAN INLT VLV	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 5301	HIS FOR AUX BLDG CTRM DMPR AIR	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS 5889A	AFP TURB 1-1 STEAM INLET VALVE	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS 607	HS FOR SS-607 (SG 1-1 SAMPLE ISO VALVE)	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS 607	HS FOR SS-607 (SG 1-1 SAMPLE ISO VALVE)	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS 612	HS FOR FW-612	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
F	20	HIS 6403	SFRCS/AFW MANUAL INITIATION SWITCH TRN 1	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS 6405	HS FOR MU 6405	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS 6405	HS FOR MU 6405	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
T	20	HIS 6406	HS FOR MU-6406	AUX	623	505	SR	57	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	HIS 6407	HS FOR MU-6407	AUX	623	505	SR	58	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	HIS 6409	HS FOR MU-6409	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS 6419	HS FOR MU-6419	AUX	623	505	SR	35	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS 6421	HS FOR MU-6421	AUX	623	505	SR	35	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	HIS 6831A	HS FOR DW-6831A	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
F	20	HIS 7528	SFAS CHANNEL 1 BLOCK SW	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	HIS 7529	SFAS CHANNEL 2 BLOCK SW	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	HIS 7530	SFAS CHANNEL 3 BLOCK SW	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	HIS 7531	SFAS CHANNEL 4 BLOCK SW	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
T	20	HIS CF1B	HS FOR NORM CF ISO VALVE CF-1B	AUX	623	505	S	18,40	OFF	OFF	N		0	1	0	0	0	0
T	20	HIS CF1B	HS FOR NORM CF ISO VALVE CF-1B	AUX	623	505	S	18	OFF	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS DH11	NORMAL DH SUCTION ISO VLV DH 11 HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS DH11	NORMAL DH SUCTION ISO VLV DH 11 HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0

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F	20	HIS DH11A	NORMAL DH SUCTION VALVE HIS IN C5704	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS DH11A	NORMAL DH SUCTION VALVE HIS IN C5704	AUX	623	505	SR		OFF	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	HIS DH12	NORMAL DH SUCT ISO VLV DH12 HIS IN C5704	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS DH12	NORMAL DH SUCT ISO VLV DH12 HIS IN C5704	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	HIS DH12A	NORMAL DH SUCT ISO VLV HIS IN C5704	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS DH12A	NORMAL DH SUCT ISO VLV HIS IN C5704	AUX	623	505	SR		OFF	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	HIS DH1B	HIS FOR DH1B	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS DH1B-2	ISO VLV HVDH1B DISCONN HIS IN C5716	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
T	20	HIS DH64	HS FOR DH-64	AUX	623	505	SR	52	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS DH6B	DH PUMP 1-1 HAND INDICATING SWITCH	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS DH6B	DH PUMP 1-1 HAND INDICATING SWITCH	AUX	623	505	SR	52	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	HIS DH9B	HS FOR DH-9B	AUX	623	505	SR	52	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	HIS HP2C	HS FOR HP-2C	AUX	623	505	SR	6	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	HIS HP2C1	SFAS LEVEL 2 BLOCK FOR HP-2C	AUX	623	505	SR	6	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	HIS HP2D	HS FOR HP-2D	AUX	623	505	SR	6	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	HIS HP2D1	SFAS LEVEL 2 BLOCK FOR HP-2D	AUX	623	505	SR	6	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	HIS HP32	HS FOR HP-32	AUX	623	505	SR	2	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS ICS11B	HS FOR ICS11B	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS MU24A	HS FOR MUP 1	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS MU24A	HS FOR MUP 1	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	20	HIS MU24A1	HS FOR MUP 1-1 MN OIL PMP P-371B	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS MU24A1	HS FOR MUP 1-1 MN OIL PMP P-371B	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	20	HIS MU24A2	HS FOR MUP 1-1 DC OIL PMP P-371C	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	20	HIS MU24A2	HS FOR MUP 1-1 DC OIL PMP P-371C	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS MU24A3	HS FOR MUP1 GEAR OIL PMP	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	20	HIS MU24A3	HS FOR MUP1 GEAR OIL PMP	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS MU24B	HS FOR MUP 2	AUX	623	505	SR	57	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	HIS MU2A	HS FOR MU-2A	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
F	20	HIS MU2B	RC LETDOWN COOLERS INLET VALVE HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS MU38	RCP SEAL RETURN ISO VALVE HIS IN C5717	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS MU40	BA BATCH STOP VLV HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	20	HIS MU50A	BA PMP 1-1 HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
T	20	HIS MU59A	HS FOR MU59A	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS MU59B	HS FOR MU59B	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1

IPEEE SMA SAFE SHUTDOWN EQUIPMENT LIST

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T	20	HIS MU59C	HS FOR MU59C	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
T	20	HIS MU59D	HS FOR MU59D	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
F	20	HIS MU66A	RCP SEAL INJECTION MU66A HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS MU66B	RCP SEAL INJECTION MU66B HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS MU66C	RCP SEAL INJECTION MU66C HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS MU66D	RCP SEAL INJECTION MU66D HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HIS NC251	DG RM SUPPLY FAN 1-1	AUX	585	318	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS NC252	DG RM SUPPLY FAN 1-2	AUX	585	318	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS NC314	ECCS RM CLR FAN 1-4 SW	AUX	545	105	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS NC315	ECCS RM CLR FAN 1-5 SW	AUX	545	105	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HIS NC711	LOW VOLT SWGR RM VENT FAN 1-1	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS NC751	CCW PMP RM VNT FAN 1-1 LOC....	AUX	585	328	SR		AUT	AUT	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS NC781	BATTERY RM VENT FAN 1-1	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS NP1951	EDG FUEL OIL ST TK 1-1 PUMP HIS	YRD	585	603	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HIS NP1951A	EDG FUEL OIL ST TK 1-1 PUMP HIS	AUX	585	320	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T	20	HIS NP1971	HS FOR P197-1	AUX	545	105	SR	6	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	HIS NP1972	HS FOR P197-2	AUX	545	105	SR	6	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	20	HIS NV0645	HS FOR CV-645B	AUX	603	402	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	0	1
F	20	HIS RC2-6	RC PRZR AUTO VENT TO QUENCH TANK HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	HIS RC2-7	PRESSURIZER HEATER CTRL SELECT HIS	AUX	585	324	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	HIS RC2-A	RC PRESSURIZER ESSEN BNK 1 HTR CTRL HIS	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	8A	HP-2B	HP1 LN2-2 ISO VALVE	AUX	565	236	R		CLS	CLS	N		1	0	0	0	0	0
F	8A	HP-2C	HPI LINE 1-1 VALVE	AUX	565	208	SR	6	CLS	OPN	Y	ELECTRICAL	0	1	0	0	0	0
F	8A	HP-2C	HPI LINE 1-1 VALVE	AUX	565	208	R		CLS	CLS	N		1	0	0	0	0	0
F	8A	HP-2D	HPI LN1-2 ISO VALVE	AUX	565	208	R		CLS	CLS	N		1	0	0	0	0	0
F	8A	HP-2D	HPI LN1-2 ISO VALVE	AUX	565	208	SR	6	CLS	OPN	Y	ELECTRICAL	0	1	0	0	0	0
T	8A	HP-32	HPI PUMP 1-1 MINI RECIRC ISOL VALVE	AUX	545	105	SR	2	OPN	OP/CL	Y	ELECTRICAL	0	1	0	0	0	0
F	20	HS-4627	INCORE TEMP HAND SWITCH	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HS-4688	HAND SWITCH FR XHAUST FAN 1-1	ITK	585	053	SR		AUT	AUT	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HS-4698	HAND SWITCH FR XHAUST FN C99-2	ITK	585	053	SR		AUT	AUT	Y	ELECTRICAL	0	0	0	0	1	0
F	20	HS-5902	HAND SWITCH FOR AFP ROOM 1 VENT FAN	AUX	565	237	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HS-6453A	SG LVL/TEST SLCT HS FOR AFP 1-1 DISCH	AUX	585	324	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HS-ICS38B	AFP TURB 1-1 CTRL SELECT HIS, IN C3630	AUX	585	324	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	HS-NI45	MANUAL TRIP SWITCH	AUX	623	505	SR		ON	ON	Y	ELECTRICAL	1	0	0	0	0	0

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F	0	HV-4906	CREVS COND 1 OUT MOTOR DRIVEN OPERATOR	AUX	638	603	SR		CLS	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	8A	HV-5261	CTRM EMERG VENT FAN 1 INLT MDO	AUX	638	603	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	0	1	0
F	7	HV-5301A	CTRM COMPUT CONFER&COMPT SUP..	AUX	638	603	SR	33	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	7	HV-5301B	CTRM CTRL CABNET RM Q PNEU OP	AUX	638	603	SR	33	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	7	HV-5301C	CTRM CABLE SPRDNG RM Q PNEU OP	AUX	638	603	SR	33	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	7	HV-5301D	CTRM I&C SHOP&KTCHN Q PNEU OP	AUX	638	603	SR	33	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	7	HV-5301E	CTRM RTRN AIR FANS IN PNEU..OP	AUX	638	603	SR	33	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	7	HV-5301F	CTRM TOILET 2 EXH FAN PNEU OP	AUX	638	603	SR	33	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	7	HV-5301G	CTRM TOILET EXH FAN PNEU OP	AUX	638	603	SR	33	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	7	HV-5301H	CTRM KITCHEN EXH FAN PNEU OP	AUX	638	603	SR	33	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	0	HV-5305	L.V.S.G. RM 429 VENT DAMP OPER	AUX	603	429	SR		OP/CL	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	0	HV-5305A	L.V.S.G. RM 429 INTK A DAMP OP	AUX	603	429	SR	36	OP/CL	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	0	HV-5305B	L.V.S.G. RM INTK B DAMP OPER	AUX	603	429	SR	36	OP/CL	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	0	HV-5329A	EDG RM 318 AIR DAMP OPERATOR	AUX	585	318	SR		OP/CL	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	0	HV-5329B	EDG RM 318 AIR DAMP OPERATOR	AUX	585	318	SR		OP/CL	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	0	HV-5329C	EDG RM 318 AIR DAMP OPERATOR	AUX	585	318	SR	36	OP/CL	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	7	HV-5361A	CABLE SPRDNG RM DMPR INLT OPER	AUX	623	506	SR	33	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	7	HV-5361B	CABLE SPRDND RM INLT DMPR OPER	AUX	623	501	SR	33	OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	0	HV-5443A	CCP RM VNT FN 1 RM OUT DAMP OP	AUX	585	328	SR		OP/CL	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	0	HV-5443B	CCP RM VNT FN 1 RM IN DAMP OP	AUX	585	328	SR		OP/CL	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	0	HV-5443C	CCP RM VNT FN1-1 RM IN DAMP OP	AUX	585	328	SR	36	OP/CL	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	0	HV-5597	BAT RM A VENT TO ATM DAMP OPER	AUX	603	429	SR		OP/CL	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
F	7	IA-630	IA PCV FOR MU66D	AUX	565	208	S		THR	THR	N		0	1	0	0	0	0
F	7	IA-636	IA PCV FOR MU66A	AUX	565	208	S		THR	THR	N		0	1	0	0	0	0
F	7	IA-648	IA PCV FOR MU38	AUX	565	208	S		THR	THR	N		0	1	0	0	0	0
F	7	IA-654	IA PCV FOR MU66B	AUX	565	208	S		THR	THR	N		0	1	0	0	0	0
F	7	IA-660	IA PCV FOR MU66C	AUX	565	208	S		THR	THR	N		0	1	0	0	0	0
F	7	ICS-11B	MS LINE 1 ATMOSPHERIC VENT VALVE	AUX	643	601	SR	93	CLS	OP/CL	Y	ELECTRICAL/MANUAL	0	0	0	1	0	0
T	R	ICS-11BB	MANUAL VALVE	AUX	623	500			CLS	OPN	Y	MANUAL	0	0	0	1	0	0
F	7	ICS-11BD	AIR CONT VLV FOR ICS 11B	AUX	643	601	S		CLS	OPN	N		0	0	0	1	0	0
F	20	II-6283	DBC1P AMMETER	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	II-6285	DBC1N AMMETER	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	II-6289	BATT 1P TO BUS 1P AMMMETER	AUX	623	502	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	II-6291	BATT 1N TO BUS 1N AMMETER	AUX	623	502	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0

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F	20	IN LMT ZRL	IN LIMIT ZONE REFERENCE LIGHTS	AUX	603	402	SR		ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	5	K3-1	AUXILIARY FEED PMP TURBINE 1-1	AUX	565	237	SR	91	STB	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	17	K5-1	EDG 1-1 (ALL SKIDMOUNTED)	AUX	585	318	SR	102	STB	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	LC-6452	STEAM GEN 1/2 LVL CTRL FR AFP 1 CTRL VLV	AUX	585	325	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	LI-1402	CC SRG TNK SIDE 1 LV INDIC	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	LI-1525A	BWST LEVEL INDICATOR SFAS CH1	AUX	623	502	SR	1	ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	20	LI-1525A	BWST LEVEL INDICATOR SFAS CH1	AUX	623	502	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	LI-2787B	EDG DAY TANK 1-1 LV INDICATOR	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T	18	LI-CF3B1	CFT 1-1 LEVEL INDICATION	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	LI-RC14-3	RC PRESSURIZER CH 1 LEVEL INDICATOR	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	LI-RC14-3	RC PRESSURIZER CH 1 LEVEL INDICATOR	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	LI-RC14-3	RC PRESSURIZER CH 1 LEVEL INDICATOR	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	20	LI-RC14-3	RC PRESSURIZER CH 1 LEVEL INDICATOR	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	20	LI-SP9B1	STEAM GEN 1 STARTUP LEVEL INDICATOR	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	LIC 6452	STEAM GEN 1/2 SU LEVEL	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	LR-MU16	RC MAKEUP TANK LEVEL RECORDER	AUX	623	505	SR	1,7	ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	20	LR-MU16	RC MAKEUP TANK LEVEL RECORDER	AUX	623	505	SR	1,7	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	18	LSH 1128	EDG DAY TANK 1-1 LVL SWITCH HI	AUX	603	321A	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	LSL 1128	EDG DAY TANK 1-1 LVL SWITCH LO	AUX	603	318	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	LT-1402	CC SRG TNK 1-1 SIDE 1 LV TRANS	AUX	623	501	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	LT-2787	EDG DAY TANK 1-1 LVL TRANSMITT	AUX	585	321A	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T	18	LT-CF3B1	CFT 1-1 LEVEL TRANSMITTER	CTM	565	214	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	18	LT-MU16-1	RC MU TANK LVL TRANSMITTER	AUX	565	A83	SR	1	ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	18	LT-MU16-1	RC MU TANK LVL TRANSMITTER	AUX	565	A83	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	18	LT-RC14-3	RC PRESSURIZER LEVEL TRANSMITTER	CTM	585	317	SR	1	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	18	LT-RC14-3	RC PRESSURIZER LEVEL TRANSMITTER	CTM	585	317	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	LT-RC14-3	RC PRESSURIZER LEVEL TRANSMITTER	CTM	585	317	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	18	LT-RC14-3	RC PRESSURIZER LEVEL TRANSMITTER	CTM	585	317	SR	1	ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	18	LT-SP9B3	STEAM GEN 1-1 STARTUP LEVEL TRANSMITTER	CTM	565	285	SR	90	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	7	MS-101	MS LINE 1 ISO VALVE	AUX	643	601	SR		OPN	CLS	Y	ELECTRICAL	0	0	0	1	0	0
F	7	MS-101-1	MS LINE 1 MSIV BYPASS VALVE	AUX	643	601	R		CLS	CLS	N		0	0	0	1	0	0
F	8B	MS-106	MS LINE 1 TO AFP TURB 1-1 ISO VALVE	AUX	623	500	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	1	0	0
	8A	MS-106A	MS LINE 2 TO AFP TURB 1-1 ISO VALVE	AUX	623	501	R		OPN	OPN	N		0	0	0	1	0	0
F	7	MS-394	MS LINE 1 WARMUP DRAIN VALVE	AUX	643	601	R		CLS	CLS	N		0	0	0	1	0	0

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F	7	MS-5889A	AFP TURB 1-1 STEAM ADMISSION VALVE	AUX	565	237	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	1	0	0
F	7	MS-5889B	AFP TURB 1-2 STEAM ADMISSION VALVE	AUX	565	238	SR	56	CLS	OPN	Y	ELECTRICAL	0	0	0	1	0	0
F	8A	MS-611	SG 1-1 DRAIN LINE ISO VALVE	AUX	565	236	R		CLS	CLS	N		0	0	0	1	0	0
F	8A	MU-11	MX BED1 OUT TO ION BED LN VLV	AUX	565	211	R		MUT	MUT	N		1	0	0	0	0	0
F	8A	MU-12A	MU FILTER 1 INLET ISO VALVE	AUX	565	211	R		OPN	OPN	N		1	0	0	0	0	0
F	8A	MU-12B	MIXED BED 1-2 INLET ISO VALVE	AUX	565	211	R		CLS	CLS	N		1	0	0	0	0	0
F	7	MU-19	SEAL INJ CONT VLV	AUX	585	303	SR	10	THR	OPN	N		0	1	0	0	0	0
F	7	MU-19	SEAL INJ CONT VLV	AUX	585	303	S	3	THR	N/A	N		1	0	0	0	0	0
F	R	MU-216	BYPASS VLV FOR MU-19	AUX	585	303		20	CLS	THR	Y	MANUAL	0	1	0	0	0	0
F	7	MU-23	BA PMP PNEUMATC DISCH CTRL VLV	AUX	565	240	SR		CLS	OPN	Y	ELECTRICAL	1	0	0	0	0	0
T	8	MU-2A	LETDOWN ISOLATION	CTM	565	214	SR	43	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
F	8A	MU-2B	RC LETDOWN ISO VALVE	CTM	565	216	SR		OPN	CLS	Y	ELECTRICAL	0	1	0	0	0	0
F	7	MU-32	MU FLOW CTRL VALVE	AUX	565	225	SR		THR	OPN	N	PNEUMATIC	0	1	0	0	0	0
F	7	MU-32	MU FLOW CTRL VALVE	AUX	565	225	SR		THR	OPN	N	PNEUMATIC	1	0	0	0	0	0
F	R	MU-338	BAAT 1 & 2 DISCH XCONN VALVE	AUX	565	240		37	CLS	OPN	Y	MANUAL	1	0	0	0	0	0
F	R	MU-348	BORIC ACID PMP 1-1 DISCH VALVE	AUX	565	240		38	OPN	THR	Y	MANUAL	1	0	0	0	0	0
F	7	MU-38	RCP SEAL RETURN ISO VALVE	AUX	565	208	SR	51	OPN	OPN	Y	ELECTRICAL/MANUAL	0	1	0	0	0	0
F	8A	MU-3971	MUP 2 SUCT 3-WAY VLV	AUX	565	225	SR	9,35	OPN	OPN	Y	ELECTRICAL	0	1	0	0	0	0
F	8A	MU-40	BATCH FEED LINE STOP ISO VLV	AUX	565	211	SR		CLS	OP/CL	Y	ELECTRICAL	1	0	0	0	0	0
F	8A	MU-59A	RCP SEAL RETURN 2-1	CTM	565	214	R		OPN	OPN	N		0	1	0	0	0	0
F	8A	MU-59A	RCP SEAL RETURN 2-1	CTM	565	214	SR	53	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
F	8A	MU-59B	RCP SEAL RETURN 2-2	CTM	565	214	R		OPN	OPN	N		0	1	0	0	0	0
F	8A	MU-59B	RCP SEAL RETURN 2-2	CTM	565	214	SR	53	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
F	8A	MU-59C	RCP SEAL RETURN 1-1	CTM	565	214	R		OPN	OPN	N		0	1	0	0	0	0
F	8A	MU-59C	RCP SEAL RETURN 1-1	CTM	565	214	SR	53	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
F	8A	MU-59D	RCP SEAL RETURN 1-2	CTM	565	214	R		OPN	OPN	N		0	1	0	0	0	0
F	8A	MU-59D	RCP SEAL RETURN 1-2	CTM	565	214	SR	53	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
F	8A	MU-6405	MUP 1 SUCT 3-WAY VLV	AUX	565	225	SR	9	OPN	OPN	Y	ELECTRICAL	1	0	0	0	0	0
F	8A	MU-6405	MUP 1 SUCT 3-WAY VLV	AUX	565	225	SR	9	OPN	OPN	Y	ELECTRICAL	0	1	0	0	0	0
F	8B	MU-6406	MUP 2 RECIRC	AUX	565	225	SR	57	OPN	CLS	Y	ELECTRICAL	0	1	0	0	0	0
F	8B	MU-6407	MUP 1 RECIRC	AUX	565	225	SR	58	OPN	CLS	Y	ELECTRICAL	0	1	0	0	0	0
F	8B	MU-6407	MUP 1 RECIRC	AUX	565	225	R		OPN	OPN	N		1	0	0	0	0	0
F	8A	MU-6408	NORM MU TO SL INJ LN...ISO VLV	AUX	565	225	R		OPN	OPN	N		1	0	0	0	0	0
F	8A	MU-6409	MUP 1 DISCH XCONN	AUX	565	225	SR	11	OPN	CLS	Y	ELECTRICAL	0	1	0	0	0	0

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F	8A	MU-6409	MUP 1 DISCH XCONN	AUX	565	225	R		OPN	OPN	N		1	0	0	0	0	0
F	8A	MU-6419	MU INJ LINE THR VLV	AUX	565	208	SR	35	CLS	THR	Y	ELECTRICAL	0	1	0	0	0	0
F	8A	MU-6419	MU INJ LINE THR VLV	AUX	565	208	R		CLS	CLS	N		1	0	0	0	0	0
F	8A	MU-6420	BYPASS VLV FOR MU-32	AUX	565	225	R		CLS	CLS	N		0	1	0	0	0	0
F	8A	MU-6420	BYPASS VLV FOR MU-32	AUX	565	225	R		CLS	CLS	N		1	0	0	0	0	0
F	8A	MU-6421	MU ALT INJ LINE CTMT ISO VLV	AUX	565	208	SR	35	CLS	OPN	Y	ELECTRICAL	0	1	0	0	0	0
F	8A	MU-6422	NORM MU LINE CTMT ISO VLV	AUX	565	225	SR	12	OPN	THR	Y	ELECTRICAL	0	1	0	0	0	0
F	8A	MU-6422	NORM MU LINE CTMT ISO VLV	AUX	565	225	SR	12	OPN	THR	Y	ELECTRICAL	1	0	0	0	0	0
F	7	MU-66A	SEAL INJ FOR RCP 2-1	AUX	565	208	SR	51	OPN	OPN	Y	ELECTRICAL/MANUAL	0	1	0	0	0	0
F	7	MU-66A	SEAL INJ FOR RCP 2-1	AUX	565	208		3	OPN	N/A	N		1	0	0	0	0	0
F	7	MU-66B	SEAL INJ FOR RCP 2-2	AUX	565	208	SR	51	OPN	OPN	Y	ELECTRICAL/MANUAL	0	1	0	0	0	0
F	7	MU-66B	SEAL INJ FOR RCP 2-2	AUX	565	208		3	OPN	N/A	N		1	0	0	0	0	0
F	7	MU-66C	SEAL INJ FOR RCP 1-1	AUX	565	208	SR	51	OPN	OPN	Y	ELECTRICAL/MANUAL	0	1	0	0	0	0
F	7	MU-66C	SEAL INJ FOR RCP 1-1	AUX	565	208		3	OPN	N/A	N		1	0	0	0	0	0
F	7	MU-66D	SEAL INJ FOR RCP 1-2	AUX	565	208	SR	51	OPN	OPN	Y	ELECTRICAL/MANUAL	0	1	0	0	0	0
F	7	MU-66D	SEAL INJ FOR RCP 1-2	AUX	565	208		3	OPN	N/A	N		1	0	0	0	0	0
T	20	NI 5874A	CH 1 NUCLEAR INSTRUMENTATION	AUX	623	505	SR	45	ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
T	7	NN-236	NITROGEN VALVE ISOLATION	AUX	565	236	SR	48	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
F	20	NP 1473	EDG 1-1 AC TURBO OIL PMP CTRL BOX CH A	AUX	585	318	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	NV-5305A	L.V.S.G. RM DAMP CTRL STATION	AUX	603	429	SR		AUT	AUT	Y	ELECTRICAL	0	0	0	0	1	0
F	20	NV-5305B	L.V.S.G. RM DAMP CTRL STATION	AUX	603	429	SR		AUT	AUT	Y	ELECTRICAL	0	0	0	0	1	0
F	20	NV-55970	BATT RM 429B DISCH DMPR LOC SW	AUX	603	429	SR		AUT	AUT	Y	ELECTRICAL	0	0	0	0	1	0
F	0	NY-5874B	NEUTRON FLUX MONITORING CH 1 AMPLIFIER	AUX	603	402	SR		ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	0	NY-5874C	NEUTRON FLUX SIGNAL PROCESSOR CH 1	AUX	603	402	SR		ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
T		P1	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P10	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	1	0	0	0
T		P11	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	1	0	0	0
T		P12	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P13	CTM PENETRATION	CTM	565	236	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P14	CTM PENETRATION	CTM	565	208	S	5	N/A	N/A	N		0	0	0	0	0	1
F	5	P14-1	AUXILIARY FEEDWATER PUMP 1-1	AUX	565	237	S		STB	ON	Y	STEAM	0	0	0	1	0	0
T		P15	CTM PENETRATION (SPARE)	CTM	565	208	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P16	CTM PENETRATION	CTM	565	236	S	5	N/A	N/A	N		0	0	0	0	0	1

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T		P17	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P18	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P19	CTM PENETRATION	CTM	565	236	S	5	N/A	N/A	N		0	1	0	0	0	0
F	6	P195-1	EDG FUEL OIL TRANSFER PUMP 1-1	YRD	585	N/A	SR		O/O	O/O	Y	ELECTRICAL	0	0	0	0	1	0
T	5	P197-1	HPI 1-1 AC OIL PUMP	AUX	545	105	SR	6	OFF	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	5	P197-2	HPI 1-1 DC OIL PUMP	AUX	545	105	SR	6	OFF	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	0	P1C5SI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P1FI	ELECTRICAL PENETRATION (SPARE)	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P1L1LI	ELECTRICAL PENETRATION	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P1L2LI	ELECTRICAL PENETRATION	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P1L5WI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P1MI	ELECTRICAL PENETRATION (SPARE)	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P1P2MI	ELECTRICAL PENETRATION	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P1P3BI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P2	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P20	CTM PENETRATION	CTM	565	236	S	5	N/A	N/A	N		0	1	0	0	0	0
T		P21	CTM PENETRATION	CTM	565	208	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P22	CTM PENETRATION	CTM	565	208	S	5	N/A	N/A	N		0	1	0	0	0	0
T		P23	CTM PENETRATION	CTM	565	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P24	CTM PENETRATION	CTM	565	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P25	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P26	CTM PENETRATION	CTM	585	303	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P27	CTM PENETRATION	CTM	565	236	S	5	N/A	N/A	N		0	0	0	1	0	0
T		P28	CTM PENETRATION	CTM	565	208	S	5	N/A	N/A	N		0	0	0	1	0	0
T		P29	CTM PENETRATION	CTM	565	236	S	5	N/A	N/A	N		0	0	0	1	0	0
T	0	P2C5CI	ELECTRICAL PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P2C5G1	ELECTRICAL PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P2L2CI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P2L4GI	ELECTRICAL PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P2P5FI	ELECTRICAL PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P2QI	ELECTRICAL PENETRATION (SPARE)	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P2RI	ELECTRICAL PENETRATION (SPARE)	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P3	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
F	6	P3-1	SERVICE WATER PUMP 1-1	ITK	585	052	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0

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T		P30	CTM PENETRATION	CTM	545	113	S	49	N/A	N/A	N		0	1	0	0	0	0
T		P31	CTM PENETRATION	CTM	545	105	S	49	N/A	N/A	N		0	1	0	0	0	0
T		P32	CTM PENETRATION	CTM	565	225	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P33	CTM PENETRATION	CTM	643	601	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P34	CTM PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P35	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	1	0	0
T		P36	CTM PENETRATION	CTM	585	308	S	5	N/A	N/A	N		0	0	0	1	0	0
T	R	P36-1	RC PUMP 1-1-1	CTM	565	216			ON	O/O	N		0	1	0	0	0	0
T	R	P36-1	RC PUMP 1-1-1	CTM	565	216			ON	ON	N		0	1	0	0	0	0
T	R	P36-2	RC PUMP 1-1-2	CTM	565	283			ON	O/O	N		0	1	0	0	0	0
T	R	P36-2	RC PUMP 1-1-2	CTM	565	283			ON	ON	N		0	1	0	0	0	0
T	R	P36-3	RC PUMP 1-2-1	CTM	565	288			ON	O/O	N		0	1	0	0	0	0
T	R	P36-3	RC PUMP 1-2-1	CTM	565	288			ON	ON	N		0	1	0	0	0	0
T	R	P36-4	RC PUMP 1-2-2	CTM	565	284			ON	O/O	N		0	1	0	0	0	0
T	R	P36-4	RC PUMP 1-2-2	CTM	565	284			ON	ON	N		0	1	0	0	0	0
T		P37	CTM PENETRATION	CTM	585	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
F	5	P37-1	MAKEUP PUMP 1-1	AUX	565	225	SR	8	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	5	P37-1	MAKEUP PUMP 1-1	AUX	565	225	SR	8	ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	5	P37-2	MAKEUP PUMP 1-2	AUX	565	225	SR	8,35	OFF	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	5	P371A	MAKEUP GEAR DRIVEN LUBE OIL PUMP	AUX	565	225	S	47	ON	ON	N		1	0	0	0	0	0
T	5	P371A	MAKEUP GEAR DRIVEN LUBE OIL PUMP	AUX	565	225	S	47	ON	ON	N		0	1	0	0	0	0
T	5	P371B	MAKEUP MAIN LUBE OIL PUMP	AUX	565	225	SR	47	ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
T	5	P371B	MAKEUP MAIN LUBE OIL PUMP	AUX	565	225	SR	47	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	5	P371C	MAKEUP AUX LUBE OIL PUMP	AUX	565	225	SR	47	OFF	ON	Y	ELECTRICAL	1	0	0	0	0	0
T	5	P371C	MAKEUP AUX LUBE OIL PUMP	AUX	565	225	SR	47	OFF	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	5	P371D	MAKEUP AUX GEAR LUBE OIL PUMP	AUX	565	225	SR	47	OFF	ON	Y	ELECTRICAL	1	0	0	0	0	0
T	5	P371D	MAKEUP AUX GEAR LUBE OIL PUMP	AUX	565	225	SR	47	OFF	ON	Y	ELECTRICAL	0	1	0	0	0	0
T		P38	CTM PENETRATION	CTM	585	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
F	5	P38-1	BORIC ACID PUMP 1-1	AUX	565	240	SR		OFF	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	5	P38-2	BORIC ACID PUMP 1-2	AUX	565	240	S		OFF	OFF	N		1	0	0	0	0	0
T		P39	CTM PENETRATION	CTM	643	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P3CI	ELECTRICAL PENETRATION (SPARE)	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P3GI	ELECTRICAL PENETRATION (SPARE)	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P3L4SI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1

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T	0	P3NI	ELECTRICAL PENETRATION (SPARE)	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P3P4CI	ELECTRICAL PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P4	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P40	CTM PENETRATION	CTM	643	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P41	CTM PENETRATION	CTM	565	236	S	5	N/A	N/A	N		0	0	0	0	0	1
F	5	P42-1	DECAY HEAT PUMP 1-1	AUX	545	105	SR	79	OFF	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	5	P42-1	DECAY HEAT PUMP 1-1	AUX	545	105	SR	15,52	OFF	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	5	P42-1	DECAY HEAT PUMP 1-1	AUX	545	105	SR	15	OFF	OFF	N		1	0	0	0	0	0
T		P42A	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P42B	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
F	5	P43-1	COMP COOLING PUMP 1-1	AUX	585	328	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	5	P43-3	CC PUMP 1-3	AUX	585	328	S		OFF	N/A	N		0	0	0	0	1	0
T		P43A	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P43B	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P44A	CTM PENETRATION	CTM	565	236	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P44B	CTM PENETRATION	CTM	565	236	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P45	CTM PENETRATION (SPARE)	CTM	565	208	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P46	CTM PENETRATION (SPARE)	CTM	565	236	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P47A	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P47B	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P48	CTM PENETRATION	CTM	565	225	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P49	CTM PENETRATION	CTM	565	208	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P4FI	ELECTRICAL PENETRATION (SPARE)	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P4L1GI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P4RI	ELECTRICAL PENETRATION (SPARE)	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P5	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	1	0	0	0
T		P50	CTM PENETRATION	CTM	565	208	S	5	N/A	N/A	N		0	1	0	0	0	0
T		P51	CTM PENETRATION	CTM	565	236	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P52	CTM PENETRATION	CTM	565	208	S	5	N/A	N/A	N		0	1	0	0	0	0
T		P53	CTM PENETRATION	CTM	565	208	S	5	N/A	N/A	N		0	1	0	0	0	0
T		P54	CTM PENETRATION	CTM	565	208	S	5	N/A	N/A	N		0	1	0	0	0	0
T		P55	CTM PENETRATION	CTM	565	208	S	5	N/A	N/A	N		0	1	0	0	0	0
T		P56	CTM PENETRATION	CTM	565	208	S	5	N/A	N/A	N		0	1	0	0	0	0
F	5	P56-1	CONTAINMENT SPRAY PUMP 1-1	AUX	545	105	SR	17	OFF	OFF	N		0	1	0	0	0	0

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F	5	P56-1	CONTAINMENT SPRAY PUMP 1-1	AUX	545	105	SR	17	OFF	OFF	N		1	0	0	0	0	0
F	5	P57	BWST RECIRC PMP	AUX	565	209	S	99	ON	N/A	N		0	1	0	0	0	0
F	5	P57	BWST RECIRC PMP	AUX	565	209	S	99	ON	N/A	N		1	0	0	0	0	0
F	5	P57	BWST RECIRC PMP	AUX	565	209	S	99	ON	N/A	N		0	1	0	0	0	0
F	5	P57	BWST RECIRC PMP	AUX	565	209	S	99	ON	N/A	N		0	1	0	0	0	0
F	5	P57	BWST RECIRC PMP	CTM	565	236	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P58	CTM PENETRATION (SPARE)	CTM	565	208	S	5	N/A	N/A	N		0	0	0	0	0	1
F	5	P58-1	HPI PUMP 1-1	AUX	545	105	SR	13	OFF	OFF	N		1	0	0	0	0	0
F	5	P58-1	HPI PUMP 1-1	AUX	545	105	SR	6	OFF	ON	Y	ELECTRICAL	0	1	0	0	0	0
T		P59	CTM PENETRATION	CTM	585	303	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P5EI	ELECTRICAL PENETRATION (SPARE)	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P5HI	ELECTRICAL PENETRATION (SPARE)	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	P5RI	ELECTRICAL PENETRATION (SPARE)	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P6	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	1	0	0	0
T		P60	CTM PENETRATION	CTM	565	236	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P61	CTM PENETRATION (SPARE)	CTM	565	208	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P62	CTM PENETRATION (SPARE)	CTM	565	208	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P63	CTM PENETRATION (SPARE)	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P64	CTM PENETRATION (SPARE)	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P65	CTM PENETRATION (SPARE)	CTM	565	236	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P66	CTM PENETRATION (SPARE)	CTM	585	303	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P67	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P68A	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P68B	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P69	CTM PENETRATION	CTM	565	208	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P7	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	1	0	0	0
T		P70	CTM PENETRATION (SPARE)	CTM	565	208	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P71A	CTM PENETRATION	CTM	585	303	S	5	N/A	N/A	N		0	0	1	0	0	0
T		P71B	CTM PENETRATION	CTM	585	303	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P71C	CTM PENETRATION	CTM	585	303	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P72A	CTM PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	1	0	0	0
T		P72B	CTM PENETRATION (SPARE)	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P72C	CTM PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P73A	CTM PENETRATION	CTM	603	402	S	5	N/A	N/A	N		0	0	1	0	0	0

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T		P73B	CTM PENETRATION	CTM	603	402	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P73C	CTM PENETRATION	CTM	603	402	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P74A	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	1	0	0	0
T		P74B	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P74C	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P75	CTM PENETRATION (SPARE)	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P76	CTM PENETRATION (SPARE)	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P77	CTM PENETRATION (SPARE)	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P78	CTM PENETRATION (SPARE)	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P80	CTM PENETRATION (EMERGENCY LOCK)	CTM	585	317A	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P81	CTM PENETRATION (PERSONNEL LOCK)	CTM	603	426	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P82	CTM PENETRATION (EQUIPMENT HATCH)	CTM	603	400	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P8A	CTM PENETRATION	CTM	623	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P8B	CTM PENETRATION	CTM	623	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P8C	CTM PENETRATION	CTM	623	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P8D	CTM PENETRATION	CTM	623	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P8E	CTM PENETRATION	CTM	623	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P8F	CTM PENETRATION	CTM	623	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P8G	CTM PENETRATION	CTM	623	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P8H	CTM PENETRATION	CTM	623	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P8I	CTM PENETRATION	CTM	623	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P8J	CTM PENETRATION	CTM	623	UNK	S	5	N/A	N/A	N		0	0	0	0	0	1
T		P9	CTM PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	1	0	0	0
T	0	PAC1NI	ELECTRICAL PENETRATION	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PAC3EI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PAL2NI	ELECTRICAL PENETRATION	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
F	0	PAL2NX	ELECTRICAL PEN TERMINAL BOX (EXTERNAL)	AUX	603	402	S		CLS	OPN	Y	ELECTRICAL	0	0	0	1	0	0
T	0	PAL3DI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PAP1BI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PAP1PI	ELECTRICAL PENETRATION	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PAP2PI	ELECTRICAL PENETRATION	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PAP3FI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PAP4BI	ELECTRICAL PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PAP5BI	ELECTRICAL PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1

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T	0	PBC2DI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PBC3PI	ELECTRICAL PENETRATION	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PBC4DI	ELECTRICAL PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PBL1EI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PBL3QI	ELECTRICAL PENETRATION	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PBL4EI	ELECTRICAL PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PBP1CI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PBP1DI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PBP1RI	ELECTRICAL PENETRATION	CTM	585	316	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PBP4AI	ELECTRICAL PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PBP5A1	ELECTRICAL PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PBP5DI	ELECTRICAL PENETRATION	CTM	603	427	S	5	N/A	N/A	N		0	0	0	0	0	1
F	20	PC-5898	CREVS STBY COND 1 DAMPER CONTROL (C6714)	AUX	643	603	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	0	1	0
T	0	PCC4TI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PCC4UI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PCC4VI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PCC5TI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PCC5UI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PCC5VI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PCL2CI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PCL2EI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PCL2FI	ELECTRICAL PENETRATION	CTM	585	314	S	5	N/A	N/A	N		0	0	0	0	0	1
F	0	PCL2GX	ELECTRICAL PEN TERMINAL BOX (EXTERNAL)	AUX	603	427	S		CLS	OPN	Y	ELECTRICAL	0	0	0	1	0	0
T	0	PCL4WI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PCP4NI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PCP4PI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PCP5NI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PCP5PI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
T	0	PCP5QI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
F	18	PDIS 1379A	SW STRNR 1-1 PRESS DIFF IND SW	ITK	585	052	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T	18	PDS 4957	DIFFERENTIAL PRESSURE SENSOR	AUX	545	105	SR	1,6	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	18	PDSH 3981	DG1 JKT CC OUT ISO VLV PDSH	AUX	585	318	SR	1,32	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	PI-2000	CTMT SFAS CH 1 PRESSURE INDICATOR	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
T	20	PI-CF4B1	CFT 1-1 PRESSURE INDICATOR	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0

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F	20	PI-MU52A	BA PMP 1-1 DISCH LN PRESS INDI	AUX	565	241	SR	1	ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	20	PI-RC2B4	RC LOOP 1 HLG WR SFAS CH 1	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	PI-RC2B4	RC LOOP 1 HLG WR SFAS CH 1	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	20	PI-SP12B	STEAM GEN 1 DISCH PRESSURE INDICATOR	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
T	0	PLP4QI	ELECTRICAL PENETRATION	CTM	603	407	S	5	N/A	N/A	N		0	0	0	0	0	1
F	18	PS 28020	CREVS COND 1 MTR UNLOADER PRESS SWITCH	AUX	643	603	SR				Y	ELECTRICAL	0	0	0	0	1	0
F	18	PS 28021	CREVS COND 1 MTR UNLOADER PRESS SWITCH	AUX	643	603	SR				Y	ELECTRICAL	0	0	0	0	1	0
F	18	PS 5900	CREVS CH 1 SWITCHOVER PRESSURE	AUX	638	603	SR				Y	ELECTRICAL	0	0	0	0	1	0
T	18	PS MU102B	MAKEUP LUBE OIL PRESSURE SENSOR	AUX	565	225	SR	1,47	ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
T	18	PS MU102B	MAKEUP LUBE OIL PRESSURE SENSOR	AUX	565	225	SR	1,47	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	18	PS2MU105B	MAKEUP LUBE OIL PRESSURE SENSOR	AUX	565	225	SR	1,47	ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
T	18	PS2MU105B	MAKEUP LUBE OIL PRESSURE SENSOR	AUX	565	225	SR	1,47	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
T	18	PS3MU105B	MAKEUP LUBE OIL PRESSURE SENSOR	AUX	565	225	SR	1,47	ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
T	18	PS3MU105B	MAKEUP LUBE OIL PRESSURE SENSOR	AUX	565	225	SR	1,47	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	0	PSE-226	PZR QUENCH TANK SAFETY VLV RUPTURE DISK	CTM	565	218	S		CLS	OP/CL	N		0	0	1	0	0	0
F	0	PSE-5463	PZR SAFTEY VALVE RUPTURE DISK	CTM	565	218	S		CLS	OP/CL	N		0	0	1	0	0	0
F	0	PSE-5464	PZR SAFETY VALVE RUPTURE DISK	CTM	565	218	S		CLS	OP/CL	N		0	0	1	0	0	0
F	18	PSH 5898	CREVS STBY COND 1 FAN START	AUX	643	603	SR				Y	ELECTRICAL	0	0	0	0	1	0
F	20	PSH 7528A	RC LOOP 1 HOT LEG SFAS CHANNEL 1	AUX	623	502	SR	1	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	PSH 7531A	RC LOOP 2 HOT LEG SFAS CHANNEL 4	AUX	623	502	SR	1	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	18	PSH RC2B4	RC LOOP 1 HLG NR, PRESS SWITCH, SFAS CH1	CTM	603	483	SR	1	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	18	PSHL 28019	CREVS UNIT 1 HIGH/LOW PRESS SWITCH	AUX	643	603	SR				Y	ELECTRICAL	0	0	0	0	1	0
F	18	PSL 106A	PRESS SWTCH LO FR AFP TURB 1-1 STM INLET	AUX	565	237	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	PSL 106B	PRESS SWITCH LOW AT AFP TURB 1-1 SUCTION	AUX	565	237	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	PSL 106C	PRESS SWITCH LOW FOR AFP TURB 1-1 INLET	AUX	565	237	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	PSL 106D	PRESS SWITCH LOW FOR AFP TURB 1-1 INLET	AUX	565	237	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
T	18	PSL 1376A	SW PMP 1-1 DISCH SRC TAP PRESS SWITCH LO	ITK	585	052	SR	1,101	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	PSL 28017	CREVS UNIT 1 LOW OIL PRESS PROT SWITCH	AUX	643	603	SR				Y	ELECTRICAL	0	0	0	0	1	0
F	18	PSL 3783	EDG STARTING AIR RECVR 1-1-1	AUX	585	318	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	PSL 3784	EDG STARTING AIR RECVR 1-1-2	AUX	585	318	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	PSL 4930A	AFP 1-1 SUCTION AFTER STRNR PRESS SWT LO	AUX	565	237	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	PSL 4930B	AFP 1-1 SUCTION AFTER STRNR PRESS SWT LO	AUX	565	237	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	PSL 5898	CREVS STANDBY COND 1 FAN STOP	AUX	643	603	SR				Y	ELECTRICAL	0	0	0	0	1	0
F	18	PSLL MU66A	PS FOR MU66A	AUX	565	208	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0

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F	18	PSLL MU66B	PS FOR MU66B	AUX	565	208	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	18	PSLL MU66C	PS FOR MU66C	AUX	565	208	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	18	PSLL MU66D	PS FOR MU66D	AUX	565	208	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	18	PT-2000	CTMT PRESSURE SFAS CH1 PRESSURE TRANSMIT	AUX	603	400	SR	1,62	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	18	PT-2002	CTMT PRESSURE SFAS CH3 PRESSURE TRANS	AUX	623	500	SR	1,62	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	18	PT-5898	CREVS CH 1 REFRIG HEAD PRESS TRANSMITTER	AUX	643	603	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T	18	PT-CF4B1	CFT 1-1 PRESSURE TRANSMITTER	CTM	565	214	SR	1	ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	18	PT-RC2B4	RCP LOOP 1 HLG WR PRESSURE TRANSMIT CH 1	CTM	603	483	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	PT-RC2B4	RCP LOOP 1 HLG WR PRESSURE TRANSMIT CH 1	CTM	603	483	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	18	PT-SP12B1	STEAM GEN 1-1 OUTLT STEAM PRESS TRANSMIT	CTM	585	317	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	7	PY-101A	MSIV--PNEUMATIC RELAY	AUX	643	601	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	1	0	0
F	7	PY-101B	MSIV--PNEUMATIC RELAYS	AUX	643	601	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	1	0	0
F	7	PY-101G	MSIV--PNEUMATIC RELAY	AUX	643	601	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	1	0	0
F	7	PY-101H	MSIV--PNEUMATIC RELAY	AUX	643	601	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	1	0	0
F	7	PY-101J	MSIV-- PNEUMATIC RELAY	AUX	643	601	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	1	0	0
T		RC-10	SPRAY BLOCK VALVE	CTM	623	580		50	OPN	OPN	N		0	1	0	0	0	0
F	8A	RC-11	PRESSURIZER POWER RELIEF ISO VALVE	CTM	623	580	R		OPN	OPN	N		0	0	1	0	0	0
F		RC-11	PRESSURIZER POWER RELIEF ISO VALVE	CTM	623	580		50	OPN	OPN	N		0	1	0	0	0	0
F	7	RC-13B	PRESSURIZER CODE SAFETY RELIEF VALVE	CTM	565	218	S		CLS	OP/CL	Y	ELECTRICAL	0	0	1	0	0	0
T	7	RC-1719A	RCS DRAIN ISOLATION	CTM	565	220	SR	48	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T	7	RC-1773A	RCS DRAIN ISOLATION	CTM	565	220	SR	48	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
F		RC-2	SPRAY ISOLATION VALVE	CTM	623	580	R	50	CLS	CLS	N		0	1	0	0	0	0
F	8A	RC-200	HIGH POINT VENT PZR TO QUENCH TANK	CTM	585	385	SR		CLS	OP/CL	Y	ELECTRICAL	0	0	1	0	0	0
F	7	RC-207	PZR QUENCH TANK RELIEF VALVE	CTM	585	218	S		CLS	OP/CL	N		0	0	1	0	0	0
T	7	RC-229A	PZR QUENCH TANK ISOLATION	AUX	565	225	SR	48	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
T		RC-229B	PZR QUENCH TANK RECIRC ISOLATION	CTM	565	220	SR	48	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
F	8A	RC-239A	RCS SAMPLE ISOLATION PZR VAPOR	CTM	585	385	SR		CLS	OP/CL	Y	ELECTRICAL	0	0	1	0	0	0
F	8A	RC-239A	RCS SAMPLE ISOLATION PZR VAPOR	CTM	585	385	R		CLS	CLS	N		0	1	0	0	0	0
F	8A	RC-239B	PRESSURIZER LIQUID PHASE SAMPLE VALVE	CTM	585	385	R		CLS	CLS	N		0	0	1	0	0	0
F	8A	RC-239B	PRESSURIZER LIQUID PHASE SAMPLE VALVE	CTM	585	385	R		CLS	CLS	N		0	1	0	0	0	0
F	8A	RC-240A	PZR VAPOR SAMPLE ISOL VALVE	CTM	585	385	R		CLS	CLS	N		0	0	1	0	0	0
F	8A	RC-240A	PZR VAPOR SAMPLE ISOL VALVE	CTM	585	385	R		CLS	CLS	N		0	0	0	0	0	1
F	0	RC-2A	PZR PWR RELIEF VALVE (SOL PILOT OP)	CTM	623	580	SR		CLS	OP/CL	Y	ELECTRICAL	0	0	1	0	0	0
F		RC-2A	PZR PWR RELIEF VALVE (SOL PILOT OP)	CTM	623	580	R		CLS	CLS	N		0	1	0	0	0	0

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F		RC-4608A	HIGH POINT VENT LOOP 1	CTM	565	216	R		CLS	CLS	N		0	1	0	0	0	0
F		RC-4610A	HIGH POINT VENT LOOP 2	CTM	565	216	R		CLS	CLS	N		0	1	0	0	0	0
F	8B	RC-4632	RC LOOP 2 COLD LEG SAMPLE VALVE	CTM	585	315	R		CLS	CLS	N		0	0	1	0	0	0
F	8B	RC-4632	RC LOOP 2 COLD LEG SAMPLE VALVE	CTM	585	315	R		CLS	CLS	N		0	1	0	0	0	0
F	10	S33-1	CREVS WATER COOLED COND 1	AUX	643	603	SR	34	ON	ON	N		0	0	0	0	1	0
F	10	S61-1	CREVS AIR COOLED CONDENSER 1	AUX	638	603	SR	34	OFF	O/O	Y	ELECTRICAL	0	0	0	0	1	0
T		SA-2010	SA HEADER ISOLATION	AUX	585	314	R		CLS	CLS	N		0	0	0	0	0	1
F	7	SP-17B1	MS LINE 1 CODE SAFETY VALVE (PSVSP17B1)	AUX	643	601	S		CLS	OP/CL	N		0	0	0	1	0	0
F	7	SP-17B2	MS LINE 1 CODE SAFETY VALVE (PSVSP17B2)	AUX	643	601	S		CLS	OP/CL	N		0	0	0	1	0	0
F	7	SP-17B3	MS LINE 1 CODE SAFETY VALVE (PSVSP17B3)	AUX	643	601	S		CLS	OP/CL	N		0	0	0	1	0	0
F	7	SP-17B4	MS LINE 1 CODE SAFETY VALVE (PSVSP17B4)	AUX	643	601	S		CLS	OP/CL	N		0	0	0	1	0	0
F	7	SP-17B5	MS LINE 1 CODE SAFETY VALVE (PSVSP17B5)	AUX	643	601	S		CLS	OP/CL	N		0	0	0	1	0	0
F	7	SP-17B6	MS LINE 1 CODE SAFETY VALVE (PSVSP17B6)	AUX	643	601	S		CLS	OP/CL	N		0	0	0	1	0	0
F	7	SP-17B7	MS LINE 1 CODE SAFETY VALVE (PSVSP17B7)	AUX	643	601	S		CLS	OP/CL	N		0	0	0	1	0	0
F	7	SP-17B8	MS LINE 1 CODE SAFETY VALVE (PSVSP17B8)	AUX	643	601	S		CLS	OP/CL	N		0	0	0	1	0	0
F	7	SP-17B9	MS LINE 1 CODE SAFETY VALVE (PSVSP17B9)	AUX	643	601	S		CLS	OP/CL	N		0	0	0	1	0	0
T		SS-235B	SAMPLE ISOLATION VALVE	CTM	585	UNK	SR	48	CLS	CLS	N		0	0	0	0	0	1
F	7	SS-607	STEAM GEN 1-1 SAMPLE LINE CTMT ISO VALVE	AUX	585	314	SR		OPN	CLS	Y	ELECTRICAL	0	0	0	1	0	0
F		SS-607	STEAM GEN 1-1 SAMPLE LINE CTMT ISO VALVE	AUX	585	314	SR	48	OPN	CLS	Y	ELECTRICAL	0	0	0	0	0	1
F	R	ST131	MAIN STEAM LINE 1 TO AFPT 1-2 STEAM TRAP	AUX	623	501		100	N/A	N/A	N		0	0	0	1	0	0
F	R	ST132	MAIN STEAM LINE 1 TO AFPT 1-2 STEAM TRAP	AUX	623	500		100	N/A	N/A	N		0	0	0	1	0	0
F	R	ST133	MAIN STEAM LINE 2 TO AFPT 1-1 STEAM TRAP	AUX	565	237		100	N/A	N/A	N		0	0	0	1	0	0
F	R	ST134	AFPT INLET HDR INLET XCONNECT STM TRAP	AUX	565	238		100	N/A	N/A	N		0	0	0	1	0	0
F	R	ST137	MAIN STEAM INLT HDR TO AFPT 1-2 STM TRAP	AUX	585	314		100	N/A	N/A	N		0	0	0	1	0	0
F	R	ST138	MAIN STEAM INLT HDR TO AFPT 1-1 STM TRAP	AUX	585	314		100	N/A	N/A	N		0	0	0	1	0	0
F	R	ST139	MAIN STEAM LINE TO AFPT 1-1 STEAM TRAP	AUX	623	501		100	N/A	N/A	N		0	0	0	1	0	0
F	R	ST148	STEAM TRAP	AUX	565	237		100	N/A	N/A	N		0	0	0	1	0	0
F	R	ST149	STEAM TRAP	AUX	545	125		100	N/A	N/A	N		0	0	0	1	0	0
F	R	ST39	MAIN STEAM LINE TO AFPT 1-1 STEAM TRAP	AUX	630	500		100	N/A	N/A	N		0	0	0	1	0	0
F	R	ST90	MAIN STEAM LINE 2 TO AFPT 1-1 STEAM TRAP	AUX	623	500		100	N/A	N/A	N		0	0	0	1	0	0
F	8B	SV-101A	MS LINE 1 ISO VALVE SOL VALVE	AUX	643	601	SR		ON	OFF	Y	ELECTRICAL	0	0	0	1	0	0
F	8B	SV-101B	MS LINE 1 ISO VALVE SOL VALVE	AUX	643	601	SR		ON	OFF	Y	ELECTRICAL	0	0	0	1	0	0
F	8B	SV-101F	MS LINE 1 ISO VALVE SOL VALVE	AUX	643	601	SR		ON	OFF	Y	ELECTRICAL	0	0	0	1	0	0
F	8B	SV-1356A	CAC 1-1 SW OUTLET ISO VALVE	AUX	585	314	SR	65	ON	OFF	Y	ELECTRICAL	0	0	0	0	1	0

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F	8B	SV-1356B	CAC 1-1 SW OUTLET ISO VALVE	AUX	585	314	SR	65	ON	OFF	Y	ELECTRICAL	0	0	0	0	1	0
F	8B	SV-1424	SOL VLV FR HX 1 SW OUT ISO VLV	AUX	585	328	SR	30	ON	OFF	Y	ELECTRICAL	0	0	0	0	1	0
F	8B	SV-1467	SOL VLV FOR HV-1467	AUX	545	113	SR	33	ON	O/O	Y	ELECTRICAL	0	0	0	0	1	0
F	8B	SV-1471	SOL VLV FOR HV-1471	AUX	585	318	SR	33	ON	O/O	Y	ELECTRICAL	0	0	0	0	1	0
F	8B	SV-5301	AUX BLDG CTRM DMPR AIR SOL VLV	AUX	638	603	SR	33	ON	OFF	Y	ELECTRICAL	0	0	0	0	1	0
F	8B	SV-5301A	CTRM COMP CONF RM&COMP..SOLVLV	AUX	638	603	SR	33	ON	OFF	Y	ELECTRICAL	0	0	0	0	1	0
F	8B	SV-5889A	AFP TURB 1-1 STM ADM BLD OFF SOL VALVE	AUX	565	237	SR		ON	OFF	Y	ELECTRICAL	0	0	0	1	0	0
F	8B	SV-607	STEAM GEN 1-1 SAMPLE LINE CTMT ISO VALVE	AUX	585	314	SR		ON	OFF	Y	ELECTRICAL	0	0	0	1	0	0
F	8B	SV-ICS11B2	SV FOR ICS-11B	AUX	643	601	SR		ON	OFF	Y	ELECTRICAL	0	0	0	1	0	0
F	8B	SV-MU38	SOL VLV FOR MU-38	AUX	565	208	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	8B	SV-MU66A	SOL VLV FOR MU-66A	AUX	565	208	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	8B	SV-MU66B	SOL VLV FOR MU-66B	AUX	565	208	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	8B	SV-MU66C	SOL VLV FOR MU-66C	AUX	565	208	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	8B	SV-MU66D	SOL VLV FOR MU-66D	AUX	565	208	SR		ON	ON	Y	ELECTRICAL	0	1	0	0	0	0
F	R	SW-105	ECCS RM COOLER 1-2 BYPASS VLV	AUX	545	115		28	OPN	OP/CL	Y	MANUAL	0	0	0	0	1	0
F	7	SW-1356	CAC 1-1 OUTLET TEMP CTRL VALVE	AUX	585	314	R		THR	OPN	Y	ELECTRICAL	0	0	0	0	1	0
T		SW-1358	CAC 1-3 OUTLET TEMP CTRL VALVE	AUX	585	314		29	CLS	CLS	N		0	0	0	0	1	0
F	8A	SW-1366	CAC 1-1 INLET ISO VALVE	AUX	585	314	R		OPN	OPN	N		0	0	0	0	1	0
F	8A	SW-1368	CAC 1-3 INLET ISO VALVE	AUX	585	314		29	OPN	OPN	N		0	0	0	0	1	0
F	8A	SW-1379	SW STRNR 1-1 DRAIN VALVE	ITK	585	052	SR		OP/CL	OP/CL	Y	ELECTRICAL	0	0	0	0	1	0
T	8A	SW-1381	SW STRNR 1-3 DRAIN VALVE	ITK	585	052	R		CLS	CLS	N		0	0	0	0	1	0
F	8A	SW-1382	SW SUPPLY TO AFP 1-1 ISO VALVE	AUX	565	237	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	1	0	0
T	8A	SW-1399	SW LOOP 1 TO TPCW HX...ISO VLV	ITK	585	053	SR		OPN	CLS	Y	ELECTRICAL	0	0	0	0	1	0
F	7	SW-1424	CCW HT XCHANG 1-1 OUT CTRL VLV	AUX	585	328	SR	30	MOD	OPN	Y	ELECTRICAL	0	0	0	0	1	0
F	8A	SW-2927	CTRM EMERG COND 1-1 OUTLET TV	AUX	638	603	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	0	1	0
F	8A	SW-2929	SW DISCH TO IN STRUCTURE VALVE	ITK	585	053	SR	25	OP/CL	CLS	Y	ELECTRICAL/MANUAL	0	0	0	0	1	0
F	8A	SW-2930	SW DISCH TO IN FOREBAY VALVE	ITK	585	053	SR	26	OP/CL	OPN	Y	ELECTRICAL/MANUAL	0	0	0	0	1	0
F	8A	SW-2931	SW DISCH TO COOLING TWR MU VLV	ITK	585	053	SR	25	OP/CL	CLS	Y	ELECTRICAL/MANUAL	0	0	0	0	1	0
F	8A	SW-2932	SW DISCH TO COLLECT BASIN VLV	ITK	585	053	SR	26	OP/CL	CLS	Y	ELECTRICAL/MANUAL	0	0	0	0	1	0
F	7	SW-2944	STRNR BLWDN - COLLEC BASIN VLV	ITK	585	052	R		CLS	CLS	N		0	0	0	0	1	0
F	7	SW-2945	STRNR BLWDN - INTAKE..4BAY VLV	ITK	585	052	R		OPN	OPN	N		0	0	0	0	1	0
F	R	SW-335	CTMT AIR COOLER SW RETURN VLV	TUR	565	251		28	OPN	OP/CL	Y	MANUAL	0	0	0	0	1	0
F	R	SW-44	CCW HT XCHANG DISCH HEADER	TUR	565	251		28	OPN	OP/CL	Y	MANUAL	0	0	0	0	1	0
F	8A	SW-5067	H2 DILU SYS BLWR1-1 MOV IN VLV	AUX	585	314	R		CLS	CLS	N		0	0	0	0	1	0

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T	8A	SW-5421	SW OUTLET MOV FOR E42-5	AUX	545	105	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	0	1	0
T	8A	SW-5422	SW OUTLET MOV E42-4	AUX	545	105	SR		CLS	OPN	Y	ELECTRICAL	0	0	0	0	1	0
F	7	SW-5896	CTRM EMERG COND 1-1 SW VLV	AUX	638	603	S		OPN	THR	Y	ELECTRICAL	0	0	0	0	1	0
F	R	SW-8432	SW RTRN IN ISO VLV TO RAD MONT	ITK	585	053		24	OPN	CLS	Y	MANUAL	0	0	0	0	1	0
F	R	SW-89	ECCS RM COOLER 1-1 BYPASS VLV	AUX	545	115		28	OPN	OP/CL	Y	MANUAL	0	0	0	0	1	0
F	R	SW-95	ECCS RM CLR 1-3 OUTLET VALVE	AUX	545	113		28	OPN	OP/CL	Y	MANUAL	0	0	0	0	1	0
T	R	T1	REACTOR VESSEL 1-1	CTM	565	213		23	ON	ON	N		0	1	0	0	0	0
T	R	T1	REACTOR VESSEL 1-1	CTM	565	213	SR		OUT	IN	N		1	0	0	0	0	0
F	21	T10	BORATED WATER STORAGE TANK 1-1	YRD	585	901	S		ON	ON	N		1	0	0	0	0	0
F	21	T10	BORATED WATER STORAGE TANK 1-1	AUX	585	901	S		ON	ON	N		0	1	0	0	0	0
F	21	T12-I	COMPONENT COOLING SURGE TNK I	AUX	623	501	S		ON	ON	N		0	0	0	0	1	0
F	21	T153-1	EDG FUEL OIL STORAGE 1-1	YRD	585	N/A	S		ON	ON	N		0	0	0	0	1	0
F	21	T18	SFP DEMINERALIZER TANK 1-1	AUX	565	233	S		ON	N/A	N		1	0	0	0	0	0
F	21	T18	SFP DEMINERALIZER TANK 1-1	AUX	565	233	S		ON	N/A	N		0	1	0	0	0	0
T	21	T198-1	HEAD TANK FOR HPI 1-1	AUX	545	105	S	6	ON	ON	N		0	1	0	0	0	0
T	0	T199-1	LUBE OIL RESERVOIR FOR HPI 1-1	AUX	545	105	S	6	ON	ON	N		0	1	0	0	0	0
T	R	T2	RCS PRESSURIZER 1-1	CTM	603	218			ON	ON	N		0	1	0	0	0	0
T	0	T2	RCS PRESSURIZER 1-1	CTM	603	218	S		ON	ON	N		0	0	1	0	0	0
F	21	T4	MAKEUP TANK 1-1	AUX	565	205	S		ON	ON	N		1	0	0	0	0	0
F	21	T4	MAKEUP TANK 1-1	AUX	565	205	S		ON	ON	N		0	1	0	0	0	0
F	21	T46-1	EDG DAY TANK 1-1	AUX	585	321A	S		ON	ON	N		0	0	0	0	1	0
F	21	T7-1	BORIC ACID ADDITION TANK 1-1	AUX	565	240	S	39	ON	ON	N		1	0	0	0	0	0
F	21	T7-2	BORIC ACID ADDITION TANK 1-2	AUX	565	240	S	39	ON	ON	N		1	0	0	0	0	0
F	21	T86-1	EDG 1-1 AIR RECEIVER 1-1-1	AUX	585	318	S		ON	ON	N		0	0	0	0	1	0
F	21	T86-2	EDG 1-1 AIR RECEIVER 1-1-2	AUX	585	318	S		ON	ON	N		0	0	0	0	1	0
T	21	T9-1	CORE FLOOD TANK 1-1	CTM	585	316	S	35	ON	ON	N		0	1	0	0	0	0
F	18	TC-5329	EDG RM 1 TEMP CONTROLLER	AUX	585	318	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	20	TDI-4951	RCS MARGIN TO SAT INDICATOR (TSAT)	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	TDI-4951	RCS MARGIN TO SAT INDICATOR (TSAT)	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	19	TE-1356	CTMT COOLER FAN 1 SUCTION TEMP ELEMENT	CTM	585	317	SR	1	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	19	TE-5329	EDG RM 318 TEMP ELEMENT	AUX	585	318	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	19	TE-5443	CC PMP 1 RM TEMP ELEMENT	AUX	585	328	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	19	TE-IM07M	INCORE OUTLET M7 TEMP ELEMENT	CTM	578	315	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	19	TE-RC385	RC LOOP 1 HLG WR TEMP ELEMENT	CTM	565	216	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0

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F	19	TE-RC3B5	RC LOOP 1 HLG WR TEMP ELEMENT	CTM	565	216	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	19	TE-RC4B2	RCP 1-1 DISCH CLG WR TEMP ELEMENT	CTM	565	216	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	19	TE-RC4B2	RCP 1-1 DISCH CLG WR TEMP ELEMENT	CTM	565	216	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	19	TE-RC4B2	RCP 1-1 DISCH CLG WR TEMP ELEMENT	CTM	565	216	SR		ON	ON	Y	ELECTRICAL	1	0	0	0	0	0
F	20	TI-1356	CTMT COOLER FAN 1 SUCTION TEMP INDICATOR	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	20	TI-4627	INCORE TEMP INDICATOR	AUX	623	505	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	0	TI-5504	PORTABLE RC TEMP INDICATOR	AUX	585	303	SR		OFF	ON	Y	ELECTRICAL	1	0	1	1	0	0
F	18	TIC 5443	CC PMP 1 RM TEMP INDEX CONTROL	AUX	585	328	SR	1	AUT	AUT	Y	ELECTRICAL	0	0	0	0	1	0
F	18	TS-4688	TEMP SWT FR XHAUST FAN C99-1&2	ITK	585	053	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	TS-5135	TEMP SWITCH FOR AFP ROOM VENT FAN 1-1	AUX	565	237	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	TS-5261	CTRM EMERG VENT FAN 1 TEMP SWT	AUX	638	603	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	TS-5318	L.V.S.G. RM DAMP TEMP SWITCH	AUX	603	429	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	TS-5443	CC PMP RM VNT FN 1 TEMP SWITCH	AUX	585	328	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	TS-5597	TEMP SW FR BATT RM A THERMO	AUX	603	429	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	TSH 1483	CC HX CCW OUT TEMP SWICH HIGH	AUX	585	328	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	TSH 5421	ECCS RM CLR FAN 1-5 TEMP SW	AUX	545	105	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	TSH 5422	ECCS RM CLR FAN 1-4 TEMP SW	AUX	545	105	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	TSL 5421	ECCS RM CLR FAN 1-5 TEMP SW	AUX	545	105	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	TSL 5422	ECCS RM CLR FAN 1-4 TEMP SW	AUX	545	105	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	TT-1356	CTMT COOLER FAN 1 SUCTION TEMP TRANSMIT	AUX	585	303	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	18	TT-5329	EDG RM 1 TEMP TRANSMITTER	AUX	585	318	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	TT-5443	CC PMP 1 RM TEMP TRANSMITTER	AUX	585	328	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	TT-IM07M	INCORE OUTLET M7 TEMP TRANSMIT	AUX	623	502	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	TT-RC3B5	RC TEMP HLG WR CH 1 TSAT TEMP TRANSMIT	AUX	623	502	SR	1	ON	ON	Y	ELECTRICAL	0	0	0	1	0	0
F	18	TT-RC3B5	RC TEMP HLG WR CH 1 TSAT TEMP TRANSMIT	AUX	623	502	SR		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
T	R	WMB1	PZR ESSENTIAL HEATER BANK 1	CTM	565	218	R		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
T	R	WMB2	PZR ESSENTIAL HEATER BANK 1	CTM	565	218	R		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
T	R	WMB3	PZR ESSENTIAL HEATER BANK 1	CTM	565	218	R		ON	ON	Y	ELECTRICAL	0	0	1	0	0	0
F	14	Y1	ESSEN INSTR DIST PNL "Y1"	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	2	Y101	XFER SWITCH FOR INV YV1 &....	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	Y101A	XFER SWITCH FOR Y1A	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	20	Y104	CREVS DISC SWITCH FOR C6708 & C6714	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	20	Y105	EDG 1-1 DISCONNECT SWITCH FOR C3615	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	14	Y1A	120VAC ESSEN INST DIST PANEL	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0

IPEEE SMA SAFE SHUTDOWN EQUIPMENT LIST

IPEEEONLY	Equip Class	Equipment ID Number	System/Equipment Description	Bldg	Elev	Room	Eval Cat.	Note	Normal State	Desired State	Pwr Reqd	Required Interconnections and Supporting Components	RC	IC	PC	DH	SU	CI
F	14	Y2	ESSEN INSTR DIST PNL "Y2" 120V	AUX	603	428	SR	41	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	2	Y201	XFER SWT FOR YV2 ABD YBR BUS	AUX	603	428	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	Y201A	XFER SWT FOR INV YV2 AND YBR	AUX	603	428	SR		CLS	CLS	N		0	0	0	0	1	0
F	14	Y2A	120VAC ESSEN INST DIST PANEL	AUX	603	428	SR	41	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	14	Y3	ESSEN INSTR DIST PNL "Y3" 120V	AUX	603	429A	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	2	Y301	XFER SWITCH FOR Y3	AUX	603	429A	SR		CLS	CLS	N		0	0	0	0	1	0
F	20	Y305	EDG 1-1 DISCONNECT SWITCH FOR C3615	AUX	603	429A	SR		CLS	CLS	N		0	0	0	0	1	0
F	14	Y4	ESSEN INSTR DIST PNL "Y4" 120V	AUX	603	428	SR	41	ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	2	Y401	XFER SWT FR DIST PNL Y4 FRM...	AUX	603	428	SR		CLS	CLS	N		0	0	0	0	1	0
F	14	YAU	UPS INSTR DIST PNL "YAU"	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	2	YAU 01	MAIN DISC SWITCH	AUX	603	429	SR		CLS	CLS	N		0	0	0	0	1	0
F	1	YE1	MCC YE1	AUX	585	318	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
	4	YE1	MCC YE1	AUX	585	318	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	2	YE101	BRKR, LVSG RM VNT FN1-1 DAMPER	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	YE102	BRKR, EDG RM 1 SPLY FAN RECIRC	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	YE103	BRKR, EDG RM 1 SPLY FAN OUTLT	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	YE104	BRKR, L.V.S.G. RM VENT FAN 1-1	AUX	585	318	SR		CLS	CLS	N		0	0	0	0	1	0
F	1	YE2	MCC YE2	AUX	585	304	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T	2	YE201	BKR FOR CV-5011A	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	YE201	BKR FOR CV-5011A	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	1	0
T	2	YE202	FEEDER BREAKER FOR MCC YE2	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	YE203	BKR FOR CV-5011C	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	YE204	FEEDER BREAKER FOR MCC YE2	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	YE205	BKR FOR CV-5011E	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	0	1
F	2	YE208	BREAKER FOR TRANS 240-120 AC..	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	YE209	BRKR, CCP RM VNT FN 1 RM BYPSS	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	YE210	BRKR, CC PMP RM VNT FN 1 RM IN	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	1	0
F	2	YE212	BRKR, CC PMP RM O.A. LOUVER 1	AUX	585	304	SR		CLS	CLS	N		0	0	0	0	1	0
F	4	YE2A	480-240V TRANSFORMER	AUX	585	304	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	4	YE2B	240-120V TRANSFORMER	AUX	585	304	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
T	2	YF-201	FEEDER BREAKER FOR MCC YF2	AUX	603	427	SR		CLS	CLS	N		0	0	0	0	0	1
T	2	YF-203	FEEDER BREAKER FOR MCC YF2	AUX	603	427	SR		CLS	CLS	N		0	0	0	0	0	1
F	16	YV1	125VDC/120VAC INVERTER CH 1	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	16	YV2	125VDC/120VAC INVERTER CH 2	AUX	603	428	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0

IPEEE SMA SAFE SHUTDOWN EQUIPMENT LIST

IPEEEONLY	Equip Class	Equipment ID Number	System/Equipment Description	Bldg	Elev	Room	Eval Cat.	Note	Normal State	Desired State	Pwr Reqd	Required Interconnections and Supporting Components	RC	IC	PC	DH	SU	CI
F	16	YV3	125VDC 120VAC INVERTER CH 3	AUX	603	429A	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	16	YV4	125VDC/120VAC INVERTER CH 4	AUX	603	428	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	16	YVA	UPS "YVA" INVERTER	AUX	603	429	SR		ON	ON	Y	ELECTRICAL	0	0	0	0	1	0
F	18	ZC-6452	AFP 1-1 DISCH CTRL VLV POS CONTROLER	AUX	565	237	SR		ON	ON	Y	ELECTRICAL	0	0	0	1	0	0

IPEEE SMA Safe Shutdown Equipment List Notes

NOTE #

1. Instrument device requires and includes all components (e.g. power supply, electronic devices, signal processors, instrument sensing tubing, interconnecting wiring, etc.) traced from the instrument to the process or controlled device which are required to support its operability. Due to the variance in the available level of details, it is not possible to show this by use of highlighting on an operational schematic.

2. HP-32 is required to be shut in the recirculation mode from the emergency containment sump.

3. For the reactivity control function, the position of this valve is not important.

4. Instrumentation, components, or equipment required exclusively for the non-loss-of-coolant-accident (non-LOCA) case.

5. A penetration listing includes checking for spatial interactions in addition to evaluating the integrity of the penetration itself.

6. Instrumentation, components, or equipment required exclusively for the small LOCA case.

7. A. Makeup (MU) tank level is required.
   B. Control Room indication chosen has been chosen instead of local indication.
   C. The Operations Department prefers use of the recorder instead of meter indication.

8. A. Assume makeup pump 1-2 is off and makeup pump 1-1 is the running pump as the normal plant Seismic Margin Analysis (SMA) lineup.
   B. Includes component cooling water (CCW) cooling assembly and skid mounted lube oil systems.

9. "Desired state open" is for both the normal supply from the makeup tank and supply from the borated water storage tank (BWST).

10. The position of MU-19 may be as-is (throttled) or failed open.

11. MU-6409 will be shut to isolate both trains of makeup in the case of a small LOCA in which the makeup system is used when high pressure injection (HPI) is unavailable.

12. To control flow, manually throttle MU-6422 due to MU-32 failing fully open upon loss of air.

13. HPI is not necessary for reactivity control during a non-LOCA, but is included as a pressure boundary. The abrasive (cyclone) separators are considered as an integral part of the pumps.

14. Relief valve only serves as a boundary isolation for this function.

15. Decay heat removal (DHR) pump is not in operation for this function, but relay evaluation is included to prevent inadvertent actuation for the non-LOCA case.

16. For the non-LOCA case, no relay evaluation is required for this function. If valve spuriously closes, it will then serve as the boundary isolation valve. If valve spuriously opens, an isolation boundary will still be maintained.

17. Relay evaluation is required to prevent possible pump dead-head operation.

18. Since the breaker for the power supply is normally open, no relay review is required.

19. Valve is to be manually closed to preserve a flow boundary.

20. Manually open for the alternate flow path.

21. For the small LOCA case, all safety features actuation signals (SFAS) for channels 1 and 3 (actuation channel 1) are required to function properly for SFAS levels 1, 2, 3, and 5.

22. For the containment isolation function, CV-5037 becomes the pressure boundary if CV-5038 fails to provide adequate isolation.

23. The reactor vessel includes all attached reactor coolant system (RCS) nuclear steam supply system (NSSS) equipment and piping which provide a pressure boundary for each loop and its associated steam generator.

24. Manually close for boundary isolation.

25. In the absence of electrical power, SW-2929 and SW-2931 may have to be manually closed to ensure the service water (SW) return flow path is directed to the other end of the intake forebay for cooling.

26. For the preferred return flow path, SW-2932 may have to be manually closed and SW-2930 manually opened to ensure a SW return flow path to the forebay in the absence of electrical power.

27. The preferred recirculation flow path for the BWST includes flow through the spent fuel pool demineralizer which requires BW-16, SF-2656, and SF-98 to be initially open.

28. Manually shut this valve for isolation of SW train 2 in order to demonstrate the capability of providing a pressure boundary.

29. Valve position is of no consequence for SMA purposes.

30. Air line between solenoid valve and service water valves is not included as a part of the SMA.

31. This note intentionally left blank.

32. Includes interlock to emergency diesel generator (EDG) CCW outlet valve as described in note "CL-6" on OS-21 SH1.

33. Air line is not included as a part of the SMA.

34. The necessary Control Room emergency ventilation system (CREVS) equipment is skid mounted, which includes the condensing unit (S33-1), the air cooled condenser (S61-1), and the cooling coil (E106-1).

35. This equipment is used for the alternate scenario in which two makeup pumps are used for inventory control for a small LOCA in which HPI is unavailable.

36. The associated piping/ductwork for the intended air path is also included within the scope of the SMA.

37. Manually open to cross-connect both boric acid addition tanks (BAATs).

38. MU-23 will fail fully open upon loss of air, requiring MU-348 to be manually throttled to prevent a runout of boric acid pump 1-1.

39. Both BAATs are required because there is no assurance that one alone will contain the necessary amount of boric acid.

40. Core flood tanks (CFTs) may be used for inventory control, such as when HPI is unavailable, otherwise the CFTs will be isolated.

41. This panel is required to feed at least channel 2 steam and feedwater rupture control system (SFRCS) and SFAS cabinets to prevent trips to support channel 1 for the non-LOCA case.

42. This equipment is located inside the annulus between the containment vessel and shield building.

43. For the containment isolation function, a SFAS actuation signal is required to send a "close" signal to the valve/component.

44. This note intentionally left blank.

45. NI 5874A is required for monitoring reactor power along with its associated power and indication circuitry.

46. The 125 VDC shown at fuses FU-3P and FU-3N at C3621 includes power traced back to the source.

47. The necessary components of the gear and bearing lube oil systems are skid mounted with the makeup pumps.

48. This valve will require both a SFAS actuation signal as well as associated air piping/tubing, solenoid valves, and components required to place the valve in its desired position for containment isolation.

49. This penetration is required exclusively for the small LOCA case. In addition to checking for spatial interactions and evaluating the integrity of the penetration itself, this penetration also includes checking for possible flow path obstruction due to failure of screens, trash racks, etc.

50. For the inventory control function, position of this valve is unimportant. However, if offsite power is still available, the desired position for RC-2 is closed to prevent the inadvertent lowering of RCS pressure.

51. Local handwheel will be used for long-term operation to ensure desired valve position for the non-LOCA case. For the small LOCA case, the desired position for containment isolation valves MU-38, MU-66A, MU-66B, MU-66C, and MU-66D is shut for a SFAS level 3 actuation.

52. This equipment will be necessary for the small LOCA case in which recirculation from the emergency containment sump is required, such as during high pressure recirculation (HPR).

53. For the small LOCA case, the desired position of containment isolation valves MU-59A, MU-59B, MU-59C, and MU-59D is shut for a SFAS level 3 actuation.

54. For the small LOCA case, it will be necessary for DH-2733 to have its SFAS level 3 signal blocked in order for entry into DHR conditions.

55. This assumes that SW-1395 is initially shut and SW-1399 is open supplying cooling water to secondary heat loads.

56. For isolation from train 2 equipment, the desired state for MS-5889B would be shut instead of open.

57. For the small LOCA scenario in which HPI is unavailable, equipment used in train 2 of the makeup system will be considered similar in construction to train 1 components.

58. To maximize makeup flow to the RCS, the minimum recirculation line may be isolated.

59. The pilot operated relief valve (PORV) normally receives power from D2N, but may also be powered from DC train 1 equipment via the maintenance cross-tie.

60. Train 2 equipment which has support systems that are similar in construction and operation to equivalent train 1 systems.

61. If steam is available from steam generator 1-2, the desired position of MS-106A is open. Otherwise, if isolation from steam generator 1-2 is desired, check valve MS-734 will provide a boundary isolation for steam generator 1-1.

62. For pressure control, both channels 1 and 3 will be required to give a SFAS actuation signal for the small LOCA case.

63. Includes panel breakers 13, 14, and 15.

64. The system boundary located downstream of this point is shown in the decay heat removal drawings. For the RCS pressure control function, there are no active components past this point in the system and no relay evaluations required since this flow path is not established until several hours following the seismic event.

65. Solenoid valves are assumed to be deenergized for the associated service water valve to be fully open.

66. This note intentionally left blank.

67. Includes associated air handling ductwork in containment.

68. For the non-LOCA case, HPI pump breakers must be racked out for DH-4849 to be able to provide low-temperature overpressure protection (LTOP).

69. This note intentionally left blank.

70. This note intentionally left blank.

71. This note intentionally left blank.

72. This note intentionally left blank.

73. This note intentionally left blank.

74. This note intentionally left blank.

75. This note intentionally left blank.

76. This note intentionally left blank.

77. This note intentionally left blank.

78. This note intentionally left blank.

79. Additional equipment is shown on support systems drawings.

80. DH-21 and DH-23 may be manually opened for an alternate flow path.

81. This note intentionally left blank.

82. This note intentionally left blank.

83. Manually close for DHR operation with train 1.

84. This note intentionally left blank.

85. This note intentionally left blank.

86. For single train operation, MCC F11A would have to be powered via the cross tie from MCC E11B using breakers described in Note 15 of OS-059. This is for the decay heat removal and pressure control functions only.

87. Operators to control DHR flow by throttling valve DH-1B via HIS DH1B.

88. Operators to open DH-1517 via HIS 1517 for DHR operation with train 1.

89. This note intentionally left blank.

90. Includes control signal to LIC 6452 (OS-17A SH1).

91. A. Includes trip throttle valve and governor valve.
    B. Includes equipment for steam exhaust path to atmosphere.

92. Includes associated fan intake and discharge path.

93. Includes associated reach rod and air tubing to bleed off the closing air supply. For decay heat removal, ICS11B will be controlled by use of a manual handwheel.

94. The SMA boundary for this piping extends to the S/I boundary outside the Auxiliary Building. Pipe integrity is not required past this point for purposes of the SMA.

95. This note intentionally left blank.

96. This note intentionally left blank.

97. This note intentionally left blank.

98. This note intentionally left blank.

99. The BWST recirculation pump and heater are included only as a pressure boundary.

100. Steam traps are included in the decay heat removal function only as a pressure boundary. It is assumed that at the time of a seismic event, the steam traps have been functioning and any water drained from the steam lines. Therefore, the steam traps are not required to function following a seismic event because the auxiliary feed pump turbines (AFPTs) are expected to start soon after occurrence of the event.

101. Assuming SW-1399 was open and SW-1395 was shut, this includes the interlock to SW-1399 as described in Note "CL-6" on OS-020 SH2.

102. Only consider those control circuits on panel C3615 (EDG 1-1) which will allow the EDG to auto start, auto load on the bus, and continue to run and provide power--e.g., control power, protective relays, etc. It will also be necessary to include the entire skid mounted portions of the EDG coolant, lube oil, air start, and fuel oil systems, including the combustion air supply flow path from the intake to the exhaust.

DAVIS-BESSE NUCLEAR POWER STATION

Individual Plant Examination of External Events

Seismic Evaluation Report

APPENDIX B

SCREENING VERIFICATION DATA SHEETS (SVDS)

SCREENING VERIFICATION DATA SHEET (SVDS)
NV-55980 CERTIFICATION

All the information contained on the following Screening Verification Data Sheets (SVDS) is, to the best of our knowledge and belief, correct and accurate. "All information" includes each entry and conclusion (whether verified to be seismically adequate or not).

The "SQUGGER" number adjacent to each seismic Capability Engineer (SCE) corresponds to the SQUGGER column on the SVDS. It represents the cognizant SCEs who were responsible for performing the walkdown and evaluation for the specified equipment.

APPROVED: Signatures of all Seismic Capability Engineers on the Seismic Reviews Team (SRT) are required. There should be at least two SCE on each SRT with at least one signatory a licensed professional engineer. All signatories should agree with all of the entries and conclusions.

SQUGGER NO.	NAME	SIGNATURE	DATE
8	JAGDISH C. ~~Jack~~ Arora P.E. JCA 8/14/95		8/14/95
1	Richard N. Bair P.E.		8/23/95
2	Thomas E, Dabrowiak P.E.		8/14/95
11	James R. Disser ~~P.E.~~ 8/9/95		8/9/95
3	John O. Dizon P.E.		8-3-95
12	Steven J. Eder P.E.		8/4/95
4	Jon G. Hook P. E.		8/15/95
5	Gayle S. Johnson P.E.		8/4/95
10	Omar Khemici P.E.		8/4/95
9	Steven J. Osting P.E.		8/15/95
6	Scott R. Saunders		8/15/95
7	Basilio Sumodobila P.E.		8/4/95

SCREENING VERIFICATION DATA SHEET (SVDS)

LINE NO	A 46	IP EEE	EQ CLASS	EQ NO	EQUIPMENT DESCRIPTION	BLDG	ELEV	RM	SQUGGER	BASE ELEV	<40	SPECTRUM CAP	SPECTRUM DEMAND	CAP> DEMAND	CAVEATS WORD	CAVEATS INTEN	ANCH OK	INTER-ACT OK	OUTLIER Y/N	OUTLIER OK	EQUIP OK
1	Y	Y	15	1N	STATION BATTERY -125V dc	AUX	603	429	3 4	603	Y	BS	GRS	Y	Y		Y	Y			Y
2	Y	Y	15	1P	STATION BATTERY +125V dc	AUX	603	429	3 4	603	Y	BS	GRS	Y	Y		Y	Y			Y
3	Y	Y	8B	AF-6452	AFP 1-1 SOL CONTROL VALVE	AUX	565	237	1 7	565	Y	BS	GRS	Y	Y		N/A	Y			Y
4	Y	Y	3	C1	4.16 KV SWITCH GEAR	AUX	585	325	2 11	585	Y	ABS	RRS	Y	Y		Y	Y			Y
5	Y	Y	10	C1-1	CAC 1-1 (AIR SIDE FUNCTION)	CTM	585	217	1 4	585	Y	ABS	CRS	Y	Y		Y	Y			Y
6	Y	Y	9	C21-1	CNTRL RM EMERG VENT SYS FAN1-1	AUX	638	603	7 9	638	NO	DOC	RRS	Y	NO	NO	Y	Y	Y	NO	NO
7	Y	Y	9	C25-1	SUPPLY FAN 1-1	AUX	585	318	1 4	610	Y	BS	CRS	Y	Y		Y	Y			Y
8	Y	Y	9	C25-2	SUPPLY FAN 1-2	AUX	585	318	1 4	610	Y	BS	CRS	Y	NO	NO	NO	Y	Y	Y	Y
9	Y	Y	20	C3017	SW STRNR 1-1 DRAIN/BCKWASH VLV CABI	ITK	576	052	6 7	580	Y	BS	GRS	Y	Y		Y	Y			Y
10	Y	Y	10	C31-4	ECCS RM CLR 1-4 FAN	AUX	545	105	4 8	545	Y	ABS	CRS	Y	Y		Y	Y			Y
11	Y	Y	10	C31-5	ECCS RM CLR 1-5 FAN	AUX	545	105	8 9	545	Y	BS	GRS	Y	Y		Y	Y			Y
12	Y	Y	20	C3615	EDG 1 CONTROL PANEL	AUX	585	318	12347	585	Y	BS	GRS	Y	Y		Y	Y			Y
13	Y	Y	20	C3617	EDG 1-1 STATIC EXITER VOLTAGE REG PNL	AUX	585	318	4 8	585	Y	ABS	RRS	Y	Y		Y	Y			Y
14	Y	Y	20	C3621A	EDG 1-1 IDLE START/STOP CONTROL PNL	AUX	585	318	4 8 12	585	Y	BS	GRS	Y	Y		Y	Y			Y
15	Y	Y	20	C3630	AUXILIARY SHUTDOWN PANEL	AUX	585	324	6 7	585	Y	BS	GRS	Y	Y		Y	Y			Y
16	Y	Y	20	C3645	CONTROL PANEL (AUX FEEDWATER)	AUX	585	325	3 4	585	Y	BS	GRS	Y	NO	Y	Y	Y			Y
17	Y	Y	20	C3812	CABINET FOR PORTABLE RC TEMP IND TI5	AUX	585	303	4 710	587	Y	BS	GRS	Y	Y		Y	Y			Y
18	Y	Y	20	C4612	CRD SYS PRIMARY TRIP BRKR C	AUX	603	428	123	603	Y	BS	GRS	Y	Y		Y	Y			Y
19	Y	Y	20	C4806	CRD SYS PRIMARY TRIP BRKR D	AUX	603	402	123	603	Y	BS	GRS	Y	Y		Y	Y			Y
20	Y	Y	20	C4808	NEUTRON FLUX MONITOR CAB CH1	AUX	603	402	1 4	603	Y	BS	GRS	Y	Y		Y	Y			Y
21	Y	Y	20	C5702	CONTROL ROOM LEFT CONSOLE	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	NO	Y	NO	NO
22	Y	Y	20	C5703	CONTROL ROOM LEFT CONSOLE	AUX	623	505	2 3	623	NO	ABS	RRS	Y	NO	NO	Y	NO	Y	NO	NO
23	Y	Y	20	C5704	CONTROL ROOM LEFT CONSOLE	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	NO	Y	NO	NO

LINE NO	A 46	IP EEE	EQ CLASS	EQ NO	EQUIPMENT DESCRIPTION	BLDG	ELEV	RM	SQUGGER	BASE ELEV	<40	SPECTRUM CAP	SPECTRUM DEMAND	CAP> DEMAND	CAVEATS WORD	CAVEATS INTEN	ANCH OK	INTER-ACT OK	OUTLIER Y/N	OUTLIER OK	EQUIP OK
24	Y	Y	20	C5705	CONTROL ROOM LEFT CONSOLE	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	NO	Y	NO	NO
25	Y	Y	20	C5706	CONTROL ROOM CENTER CONSOLE	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	NO	Y	NO	NO
26	Y	Y	20	C5707	CONTROL ROOM CENTER CONSOLE	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	NO	Y	NO	NO
27	Y	Y	20	C5708	CONTROL ROOM CENTER CONSOLE	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	NO	Y	NO	NO
28	Y	Y	20	C5709	CONTROL ROOM CENTER CONSOLE	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	NO	Y	NO	NO
29	Y	Y	20	C5712	CONTROL ROOM RIGHT CONSOLE	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	NO	Y	NO	NO
30	Y	Y	20	C5715	CONTROL ROOM STATION ELEC. DIST. PAN	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	Y			Y
31	Y	Y	20	C5716	CONTROL ROOM ENG. SAFETY FEAT. CON.	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	Y			Y
32	Y	Y	20	C5717	CONTROL ROOM ENG. SAFETY FEATR. CON	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	Y			Y
33	Y	Y	20	C5719	CON RM REACTOR & STATION AUX CONT	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	Y			Y
34	Y	Y	20	C5720	CONTROL ROOM REACTOR IN STAT AUX C	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	Y			Y
35	Y	Y	20	C5721	CONTROL ROOM FEEDWATER CONTROL PA	AUX	623	505	2 3	623	NO	ABS	RRS	Y	Y		Y	Y			Y
36	Y	Y	20	C5755C	SFAS CHANNEL 2	AUX	623	502	2 3	623	NO	DOC	RRS	Y	NO	NO	NO	NO	Y	NO	NO
37	Y	Y	20	C5756D	SFAS CHANNEL 4	AUX	623	502	2 5	623	NO	DOC	RRS	Y	NO	NO	NO	NO	Y	NO	NO
38	Y	Y	20	C5761A	SFRCS ACTUATION CHANNEL 1	AUX	623	502	2 5	623	NO	ABS	RRS	Y	Y		Y	NO	Y	NO	NO
39	Y	Y	20	C5762A	SFRCS ACTUATION CHANNEL 1	AUX	623	502	2 5	623	NO	ABS	RRS	Y	NO	NO	NO	NO	Y	NO	NO
40	Y	Y	20	C5762C	SFAS CHANNEL 1	AUX	623	502	2 5	623	NO	DOC	RRS	Y	NO	NO	NO	NO	Y	NO	NO
41	Y	Y	20	C5763A	POST ACCIDENT EQUIP RACK CH4	AUX	623	502	6 7	623	NO	ABS	RRS	Y	Y		Y	Y			Y
42	Y	Y	20	C5763D	SFAS CHANNEL 3	AUX	623	502	2 5	623	NO	DOC	RRS	Y	NO	NO	NO	NO	Y	NO	NO
43	Y	Y	20	C5798	POST ACCIDENT INDICATING PANEL CH2	AUX	623	505	6 7	623	NO	ABS	CRS	Y	Y		Y	Y			Y
44	Y	Y	20	C5799	POST ACCIDENT INDICATING PAN CH1	AUX	623	505	6 7	623	NO	ABS	CRS	Y	Y		Y	Y			Y
45	Y	Y	20	C6714	CREVS CONTROL PANEL	AUX	638	603	7 9	638	NO	ABS	RRS	Y	Y		Y	Y			Y
46	Y	Y	9	C71-1	L.V.S.G. RM VENT FAN 1-1	AUX	603	429	1 4	613	Y	BS	GRS	Y	Y		Y	Y			Y

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47	Y	Y	9	C73-1	AFP ROOM EXHAUST FAN	AUX	565	237	6 7	585	Y	ABS	CRS	Y	NO	NO	NO	Y	Y	Y	Y
48	Y	Y	9	C75-1	CC PMP RM VENT FAN 1-1	AUX	585	328	2 6	603	Y	BS	GRS	Y	Y		Y	Y			Y
49	Y	Y	9	C78-1	BATTERY ROOM VENT FAN 1-1	AUX	603	429	2 4	613	Y	BS	GRS	Y	NO	NO	NO	Y	Y	Y	Y
50	Y	Y	9	C99-1	EXHAUST FAN 1-1	ITK	585	52A	8 9	586	Y	BS	GRS	Y	Y		Y	Y			Y
51	Y	Y	9	C99-2	EXHAUST FAN 1-2	ITK	585	52A	8 9	586	Y	BS	GRS	Y	Y		Y	Y			Y
52		Y	8A	CC-1407A	CCW RETURN ISO FROM LETDOWN COOLER	CTM	585	315	1 4	595	Y	BS	GRS	Y	Y		N/A	Y			Y
53		Y	7	CC-1411A	CCW SUPPLY ISOLATION	CTM	585	315	1 4	597	Y	BS	GRS	Y	Y		N/A	Y			Y
54	Y	Y	7	CC-1467	CCW FRM DH RMVL CLR 1-1...VLV	AUX	545	113	1367	565	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
55	Y	Y	7	CC-1471	CC FRM EDG 1-1 SOL OUTLET VLV	AUX	585	318	1 4	585	Y	BS	GRS	Y	Y		N/A	Y			Y
56		Y	7	CC-1567A	CCW SUPPLY ISOLATION FOR CRD COOLIN	CTM	585	315	1 4	580	Y	BS	GRS	Y	Y		N/A	Y			Y
57	Y	Y	8A	CC-5095	CC LN 1 DISCH ISO VALVE	AUX	585	328	6 7	585	Y	BS	GRS	Y	Y		N/A	Y			Y
58	Y	Y	4	CE1-1	4.16 KV-480V TRANSFORMER	AUX	603	429	6 9	603	Y	BS	GRS	Y	Y		N/A	Y			Y
59	Y	Y	8A	CF-1B	CORE FLOOD TANK 1 ISO VLV	CTM	565	214	1 4	585	Y	BS	GRS	Y	Y		N/A	Y			Y
60		Y	8A	CS-1530	CTMT SPRAY DISCH ISO VALVE TRAIN 1	AUX	585	303	2 4	585	Y	BS	GRS	Y	Y		N/A	Y			Y
61		Y	8	CV-5010A	P71B ISOLATION	CTM	585	316	1 4	589	Y	BS	GRS	Y	NO	Y	Y	Y			Y
62		Y	8	CV-5010C	P73B ISOLATION	CTM	603	410	1 4	610	NO	DOC	CRS	Y	Y		N/A	Y			Y
63		Y	8	CV-5011A	P71B ISOLATION	AUX	585	303	4 7 10	589	Y	DOC	CRS	Y	NO	Y	N/A	Y			Y
64		Y	8	CV-5011B	P68B ISOLATION	CTM	603	410	1 4	603	Y	DOC	CRS	Y	NO	Y	N/A	Y			Y
65		Y	8	CV-5011C	P73B ISOLATION	AUX	603	402	4 7 10	611	Y	DOC	CRS	Y	NO	Y	N/A	Y			Y
66		Y	8	CV-5011D	P74B ISOLATION	CTM	585	317	1 4	597	Y	DOC	CRS	Y	NO	Y	N/A	Y			Y
67		Y	8	CV-5011E	P43B ISOLATION	AUX	585	314	4 7 10	593	Y	DOC	CRS	Y	NO	Y	N/A	Y			Y
68		Y	8	CV-5070	MOTOR OPERATED BUTTERFLY VALVE	CTM	623	ANN	1 4	631	Y	DOC	CRS	Y	Y		N/A	Y			Y
69		Y	8	CV-5071	MOTOR OPERATED BUTTERFLY VALVE	CTM	623	ANN	1 4	631	Y	DOC	CRS	Y	Y		N/A	Y			Y

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70		Y	8	CV-5072	MOTOR OPERATED BUTTERFLY VALVE	CTM	623	ANN	1 4	631	Y	DOC	CRS	Y	Y		N/A	Y			Y
71		Y	8	CV-5073	MOTOR OPERATED BUTTERFLY VALVE	CTM	623	ANN	1 4	631	Y	DOC	CRS	Y	Y		N/A	Y			Y
72		Y	8	CV-5074	MOTOR OPERATED BUTTERFLY VALVE	CTM	623	ANN	1 4	631	Y	DOC	CRS	Y	Y		N/A	Y			Y
73		Y	8	CV-645B	DIFFERENTIAL PRESSURE ISOLATION VALVE	AUX	603	402	4 7 10	611	NO	DOC	CRS	Y	NO	Y	N/A	Y			Y
74	Y	Y	3	D1	4.16 KV SWITCH GEAR	AUX	585	323	2 11	585	Y	ABS	RRS	Y	Y		Y	Y			Y
75	Y	Y	14	D1N	ESSEN DIST PNL "D1N"	AUX	603	429	2 3	603	Y	BS	GRS	Y	NO	NO	Y	NO	Y	NO	NO
76	Y	Y	1	D1NA	ESSENTIAL -125VDC DIST PNL CH1	AUX	603	429	2 3	603	Y	BS	GRS	Y	NO	NO	NO	Y	Y	NO	NO
77	Y	Y	14	D1P	ESSEN DIST PNL "D1P"	AUX	603	429	12347	603	Y	BS	GRS	Y	Y		Y	Y			Y
78	Y	Y	14	D2N	ESSEN DIST PNL "D2N"	AUX	603	428	4 7	603	Y	BS	GRS	Y	Y		Y	Y			Y
79	Y	Y	14	D2P	ESSNTL +125VDC DISTBTN PNL CH2	AUX	603	428	2 3	603	Y	BS	GRS	Y	NO	NO	Y	NO	Y	NO	NO
80	Y	Y	8B	DA-3783	EDG AIR RCVR 1-1-1 TO AIR..VLV	AUX	585	318	12347	595	Y	BS	GRS	Y	Y		N/A	Y			Y
81	Y	Y	8B	DA-3784	EDG AIR RCVR 1-1-2 TO AIR..VLV	AUX	585	318	12347	595	Y	BS	GRS	Y	Y		N/A	Y			Y
82	Y	Y	16	DBC1N	BATTERY CHARGER -125V dc	AUX	603	429	2 3	603	Y	BS	GRS	Y	Y		Y	Y			Y
83	Y	Y	16	DBC1P	BATT CHARGER FOR BATT 1P +125V	AUX	603	429	2 3	603	Y	BS	GRS	Y	Y		Y	Y			Y
84	Y	Y	1	DC MCC-1		AUX	603	429	1 3	603	Y	BS	GRS	Y	Y		Y	Y			Y
85	Y	Y	8A	DH-11	RCS TO DH SYSTEM ISO VALVE	CTM	565	290	1 4	565	Y	BS	GRS	Y	Y		N/A	Y			Y
86	Y	Y	8A	DH-12	RCS TO DH SYSTEM ISO VALVE	CTM	565	290	1 4	565	Y	BS	GRS	Y	Y		N/A	Y			Y
87	Y	Y	8A	DH-1517	DH PUMP 1-1 SUCTION FROM RCS VALVE	AUX	565	236	1367	565	Y	BS	GRS	Y	Y		N/A	Y			Y
88	Y	Y	8A	DH-1518	DH PUMP 1-2 SUCTION FROM RCS	AUX	565	236	1367	565	Y	BS	GRS	Y	Y		N/A	Y			Y
89	Y	Y	8A	DH-1B	DH COOLER 1-1 DISCH TO RCS ISO VALVE	AUX	565	208	6 7	565	Y	BS	GRS	Y	Y		N/A	Y			Y
90	Y	Y	8A	DH-2733	DH PMP 1-1 SUC (BWST OR EMERG SUMP)	AUX	545	105	1367	565	Y	BS	GRS	Y	Y		N/A	Y			Y
91	Y	Y	7	DH-4849	DH COOLDOWN LN RELIEF TO EMERG SUMP	CTM	565	220	1 7	565	Y	BS	GRS	Y	Y		N/A	Y			Y
92		Y	8A	DH-64	DH PMP 1-1 DISCH TO HPI PUMP 1-1 SUL V	AUX	545	105	6 7	565	Y	ABS	CRS	Y	Y		N/A	Y			Y

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93		Y	8A	DH-9B	DH PUMP 1-1 SUCT PRM EMER SUMP VLV	AUX	545	105	6 7	565	Y	ABS	CRS	Y	NO	Y	N/A	Y			Y
94		Y	8	DR-2012A	NORMAL SUMP ISOLATION	CTM	565	292	1 4	514	Y	BS	GRS	Y	Y		N/A	Y			Y
95		Y	8	DW-6831A	DEMINERALIZED WATER ISOLATION	CTM	585	316	1 4	586	Y	BS	GRS	Y	Y		N/A	Y			Y
96	Y	Y	2	E1	480V ESSENTIAL UNIT SUBSTATION	AUX	603	429	6 9	603	Y	BS	GRS	Y	NO	NO	NO	Y	Y	NO	NO
97	Y	Y	1	E11A	480V ESSENTIAL MCC	AUX	565	209	1489	565	Y	BS	GRS	Y	Y		Y	Y			Y
98	Y	Y	1	E11B	480V ESSENTIAL MCC	AUX	585	304	1489	585	Y	BS	GRS	Y	Y		Y	NO	Y	NO	NO
99	Y	Y	1	E11C	480V ESSENTIAL MCC	AUX	585	304	1489	585	Y	BS	GRS	Y	Y		Y	NO	Y	NO	NO
100	Y	Y	1	E11D	480V ESSENTIAL MCC	AUX	565	227	1489	565	Y	BS	GRS	Y	Y		Y	NO	Y	NO	NO
101	Y	Y	1	E11E	480V ESSENTIAL MCC	AUX	603	402	1489	603	Y	BS	GRS	Y	Y		Y	Y			Y
102	Y	Y	1	E12A	480V ESSENTIAL MCC	AUX	603	429	1389	603	Y	BS	GRS	Y	Y		Y	Y			Y
103	Y	Y	1	E12B	480V ESSENTIAL MCC	AUX	585	318	1489	585	Y	BS	GRS	Y	Y		Y	NO	Y	NO	NO
104	Y	Y	1	E12C	480V ESSENTIAL MCC	ITK	576	051	4 9	576	Y	BS	GRS	Y	NO	NO	NO	Y	Y	Y	Y
105	Y	Y	1	E12E	480V ESSENTIAL MCC	AUX	545	101	1489	545	Y	BS	GRS	Y	Y		Y	Y			Y
106	Y	Y	1	E12F	480V ESSENTIAL MCC	AUX	585	318	49	585	Y	BS	GRS	Y	Y		Y	Y			Y
107	Y	Y	1	E14	480V ESSENTIAL MCC	AUX	603	429	489	603	Y	BS	GRS	Y	Y		Y	Y			Y
108	Y	Y	21	E22-1	COMP. COOLING HEAT XCHANGR 1-1	AUX	585	328	1367	585	Y	N/A	N/A	U	N/A		U	Y			U
109	Y	Y	21	E22-3	COMP. COOLING HEAT XCHANGR 1-3	AUX	585	328	1367	585	Y	N/A	N/A	U	N/A		U	Y			U
110	Y	Y	21	E26-1	SEAL RETURN COOLER 1-1	AUX	565	208	2 10	565	Y	N/A	N/A	Y	N/A		Y	Y			Y
111	Y	Y	21	E26-2	SEAL RETURN COOLER 1-2	AUX	565	208	2 10	565	Y	N/A	N/A	Y	N/A		Y	Y			Y
112	Y	Y	21	E27-1	DECAY HEAT REMOVAL COOLER 1-1	AUX	545	113	1367	545	Y	N/A	N/A	U	N/A		U	Y			U
113	Y	Y	21	E34	BWST HEATER	AUX	565	209	1 7	565	Y	N/A	N/A	N/A	N/A		Y	N/A			Y
114	Y	Y	10	E37-1	CAC COIL 1-1 (SW SIDE)	CTM	585	317	1 4	585	Y	DOC	CRS	Y	NO	NO	NO	Y	Y	Y	Y
115	Y	Y	10	E37-3	CTMT AIR COOLER 1-3	CTM	585	317	1 4	585	Y	DOC	CRS	Y	NO	NO	NO	Y	Y	Y	Y

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116	Y	Y	10	E42-4	ECCS ROOM COOLER COIL 1-4	AUX	545	105	4 8	545	Y	ABS	CRS	Y	Y		Y	Y			Y
117	Y	Y	10	E42-5	ECCS ROOM COOLER COIL 1-5	AUX	545	105	8 9	545	Y	BS	GRS	Y	Y		Y	Y			Y
118	Y	Y	2	F1	480V ESSENTIAL UNIT SUBSTATION	AUX	603	428	6 9	603	Y	BS	GRS	Y	NO	NO	NO	Y	Y	NO	NO
119	Y	Y	0	F108-1	EDG 1-1 INTAKE FILTER	AUX	610	N/A	6 9	610	Y	DOC	CRS	Y	N/A		NO	Y	Y	NO	NO
120	Y	Y	1	F11A	480V ESSENTIAL MCC	AUX	603	427	149	603	Y	BS	GRS	Y	NO	NO	Y	NO	Y	NO	NO
121	Y	Y	1	F12A	480V ESSENTIAL MCC	AUX	603	428	489	603	Y	BS	GRS	Y	Y		Y	Y			Y
122	Y	Y	0	F15-1	SERVICE WATER STRAINER 1-1	ITK	576	052	127	576	Y	DOC	CRS	Y	N/A		Y	Y			Y
123	Y	Y	1	F16A	480V ESSENTIAL MCC	AUX	603	428	6 7	603	Y	BS	GRS	Y	NO	NO	NO	Y	Y	Y	Y
124	Y	Y	18	FIS 1422C	CC PMP 1-1 DISCH FLOW INDIC SW	AUX	585	328	6 7	590	Y	BS	GRS	Y	Y		Y	Y			Y
125		Y	18	FT 6425	MU FLOW TRANSMITTER FOR INJ LINE C	AUX	545	105	7 8	548	Y	BS	GRS	Y	Y		Y	Y			Y
126	Y	Y	18	FT DH2B	LP INJ LINE 1 FLOW TRANSMITTER	AUX	545	105	5 6	549	Y	BS	GRS	Y	Y		Y	Y			Y
127		Y	18	FT HP3C	FLOW TRANSMITTER FOR HP-2C	AUX	565	208	7 8	569	Y	BS	GRS	Y	Y		Y	Y			Y
128		Y	18	FT HP3D	FLOW TRANSMITTER FOR HP-2D	AUX	565	208	7 8	569	Y	BS	GRS	Y	Y		Y	Y			Y
129		Y	18	FT MU34	MU FLOW TRANSMITTER FOR INJ LINE A	AUX	565	225	7 8	569	Y	BS	GRS	Y	Y		Y	Y			Y
130	Y	Y	20	HIS NC251	EDG RM VENTILATION FAN 1 LCL	AUX	585	318	1 4	589	Y	BS	GRS	Y	Y		Y	Y			Y
131	Y	Y	20	HIS NC252	EDG RM VENTILATION FAN 2 LCL	AUX	585	318	1 4	589	Y	BS	GRS	Y	NO	Y	Y	Y			Y
132	Y	Y	20	HIS NC314	ECCS RM CLR FAN 1-4 SW	AUX	545	105	5 6	549	Y	BS	GRS	Y	Y		Y	Y			Y
133	Y	Y	20	HIS NC315	ECCS RM CLR FAN 1-5 SW	AUX	545	105	5 6	549	Y	BS	GRS	Y	NO	Y	Y	Y			Y
134	Y	Y	20	HIS NC711	LOW VOLT. SWGR RM VENT FAN 101 LCL	AUX	603	429	6 7	608	Y	BS	GRS	Y	Y		Y	Y			Y
135	Y	Y	20	HIS NC751	CCW PMP RM VNT FAN 1-1 LOC....	AUX	585	328	6 7	589	Y	BS	GRS	Y	NO	Y	Y	Y			Y
136	Y	Y	20	HIS NC781	BATTERY RM VENT FAN 1-1 LCL	AUX	603	429	6 7	607	Y	BS	GRS	Y	NO	Y	Y	Y			Y
137	Y	Y	20	HIS NP1951	EDG FUEL OIL ST TK 1-1 HAND IND SW	YRD	585	N/A	1 6	578	Y	BS	GRS	Y	Y		Y	Y			Y
138	Y	Y	20	HIS NP1951A	EDG FUEL OIL ST TK 1-1 HAND IND SW	AUX	585	321	1 6	595	Y	BS	GRS	Y	Y		Y	Y			Y

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139		Y	20	HIS NP197-1	HS FOR P197-1	AUX	545	105	2 11	548	Y	BS	GRS	Y	Y		Y	Y			Y
140		Y	20	HIS NP197-2	HS FOR P197-2	AUX	545	105	2 11	548	Y	BS	GRS	Y	Y		Y	Y			Y
141		Y	20	HIS NV0645	HS FOR CV-645B	AUX	603	402	2 11	607	Y	BS	GRS	Y	Y		Y	Y			Y
142		Y	8A	HP-2C	HPI LINE 1-1 VALVE	AUX	565	208	1 10	567	Y	BS	GRS	Y	Y		N/A	Y			Y
143		Y	8A	HP-2D	HPI LINE 1-1 VALVE	AUX	565	208	1 10	566	Y	BS	GRS	Y	Y		N/A	Y			Y
144		Y	8A	HP-32	HPI PUMP 1-1 MINI RECIRC ISOL VALVE	AUX	545	105	1 10	555	Y	BS	GRS	Y	Y		N/A	Y			Y
145	Y	Y	20	HS-4688	H. S. FR XHAUST FAN 1-1 NC 9901	ITK	576	052	5 6	581	Y	BS	GRS	Y	Y		Y	Y			Y
146	Y	Y	20	HS-4698	H. S. FR XHAUST FN C99-2 NC 9901	ITK	576	052	5 6	581	Y	BS	GRS	Y	Y		Y	Y			Y
147	Y	Y	20	HS-5902	H. S. FOR AFP ROOM 1 VENT FAN NC 0731	AUX	565	237	6 7	565	Y	BS	GRS	Y	Y		Y	Y			Y
148	Y	Y	0	HV-4906	CTRM EVS STBY COND 1 MOTOR OPER	AUX	656	N/A	2 10	656	NO	DOC	RRS	Y	N/A		Y	Y			Y
149	Y	Y	0	HV-5261	CTRM EMERG VENT FAN 1 INLT MDO	AUX	638	603	2 10	656	NO	U	CRS	NO	N/A		NO	NO	Y	NO	NO
150	Y	Y	0	HV-5301A	CTRM COMPUT CONFER&COMPT SUP..	AUX	638	603	6 7	638	NO	DOC	CRS	Y	N/A		N/A	Y			Y
151	Y	Y	0	HV-5301B	CTRM CTRL CABNET RM Q PNEU OP	AUX	638	603	6 7	638	NO	DOC	CRS	Y	N/A		N/A	Y			Y
152	Y	Y	0	HV-5301C	CTRM CABLE SPRDNG RM Q PNEU OP	AUX	638	603	6 7	638	NO	DOC	CRS	Y	N/A		N/A	Y			Y
153	Y	Y	0	HV-5301D	CTRM I&C SHOP&KTCHN Q PNEU OP	AUX	638	603	6 7	638	NO	DOC	CRS	Y	N/A		N/A	Y			Y
154	Y	Y	0	HV-5301E	CTRM RTRN AIR FANS IN PNEU..OP	AUX	638	603	6 7	638	NO	DOC	CRS	Y	N/A		N/A	Y			Y
155	Y	Y	0	HV-5301F	CTRM TOILET 2 EXH FAN PNEU OP	AUX	638	603	6 7	643	NO	DOC	CRS	Y	N/A		N/A	Y			Y
156	Y	Y	0	HV-5301G	CTRM TOILET EXH FAN PNEU OP	AUX	638	603	6 7	643	NO	DOC	CRS	Y	N/A		N/A	Y			Y
157	Y	Y	0	HV-5301H	CTRM KITCHEN EXH FAN PNEU OP	AUX	638	603	6 7	646	NO	DOC	CRS	Y	N/A		N/A	Y			Y
158	Y	Y	0	HV-5305	L.V.S.G. RM 429 VENT DAMP OPER	AUX	603	429	1 4	620	NO	DOC	CRS	Y	N/A		Y	Y			Y
159	Y	Y	0	HV-5305A	L.V.S.G. RM 429 INTK A DAMP OP	AUX	603	429	1 4	620	NO	DOC	CRS	Y	N/A		Y	Y			Y
160	Y	Y	0	HV-5305B	L.V.S.G. RM INTK B DAMP OPER	AUX	603	429	2 8	613	Y	DOC	CRS	Y	N/A		Y	Y			Y
161	Y	Y	0	HV-5329A	EDG RM 318 AIR DAMP OPERATOR	AUX	585	318	1 4	610	Y	DOC	CRS	Y	N/A		Y	Y			Y

LINE NO	A 46	IP EEE	EQ CLASS	EQ NO	EQUIPMENT DESCRIPTION	BLDG	ELEV	RM	SQUGGER	BASE ELEV	<40	SPECTRUM CAP	SPECTRUM DEMAND	CAP> DEMAND	CAVEATS WORD	CAVEATS INTEN	ANCH OK	INTER-ACT OK	OUTLIER Y/N	OUTLIER OK	EQUIP OK
162	Y	Y	0	HV-5329B	EDG RM 318 AIR DAMP OPERATOR	AUX	585	318	1 4	610	Y	DOC	CRS	Y	N/A		Y	Y			Y
163	Y	Y	0	HV-5329C	EDG RM 318 AIR DAMP OPERATOR	AUX	585	318	1 4	610	Y	DOC	CRS	Y	N/A		Y	Y			Y
164	Y	Y	0	HV-5361A	CABLE SPRDNG RM DMPR INLT OPER	AUX	623	506	6 7	642	NO	DOC	CRS	Y	N/A		Y	Y			Y
165	Y	Y	0	HV-5361B	CABLE SPRDNG RM INLT DMPR OPER	AUX	623	501	6 7	642	NO	DOC	CRS	Y	N/A		Y	Y			Y
166	Y	Y	0	HV-5443A	CCP RM VNT FN 1 RM OUT DAMP OP	AUX	585	328	678	595	Y	DOC	CRS	Y	N/A		Y	Y			Y
167	Y	Y	0	HV-5443B	CCP RM VNT FN 1 RM IN DAMP OP	AUX	585	328	678	595	Y	DOC	CRS	Y	N/A		Y	Y			Y
168	Y	Y	0	HV-5443C	CCP RM VNT FN1-1 RM IN DAMP OP	AUX	585	328	1 6	603	Y	DOC	CRS	Y	N/A		Y	Y			Y
169	Y	Y	0	HV-5597	BAT RM A VENT TO ATM DAMP OPER	AUX	603	429	2 4	615	Y	DOC	CRS	Y	N/A		Y	Y			Y
170	Y	Y	18	IA-630	IA PCV FOR MU66D	AUX	565	208	567	565	Y	BS	GRS	Y	Y		Y	Y			Y
171	Y	Y	18	IA-636	IA PCV FOR MU66A	AUX	565	208	567	565	Y	BS	GRS	Y	Y		Y	Y			Y
172	Y	Y	18	IA-648	IA PCV FOR MU38	AUX	565	208	567	565	Y	BS	GRS	Y	Y		Y	Y			Y
173	Y	Y	18	IA-654	IA PCV FOR MU66B	AUX	565	208	567	565	Y	BS	GRS	Y	Y		Y	Y			Y
174	Y	Y	18	IA-660	IA PCV FOR MU66C	AUX	565	208	567	565	Y	BS	GRS	Y	Y		Y	Y			Y
175	Y	Y	7	ICS-11B	MS LINE 1 ATMOSPHERIC VENT VALVE	AUX	643	601	147	660	NO	DOC	CRS	Y	N/A		N/A	Y			Y
176	Y	Y	5	K3-1	AUXILIARY FEED PMP TURBINE 1-1	AUX	565	237	1 7	565	Y	BS	GRS	Y	NO	Y	Y	Y			Y
177	Y	Y	17	K5-1	EDG 1-1	AUX	585	318	1 9	585	Y	BS	GRS	Y	Y		Y	Y			Y
178	Y	Y	18	LSH 1128	EDG DAY TANK 1-1 LVL SWITCH HI	AUX	595	321	12347	600	Y	BS	GRS	Y	Y		N/A	Y			Y
179	Y	Y	18	LSL 1128	EDG DAY TANK 1-1 LVL SWITCH LO	AUX	595	321	12347	600	Y	BS	GRS	Y	Y		N/A	Y			Y
180	Y	Y	18	LT-1402	CC SRG TNK 1-1 SIDE 1 LV TRANS	AUX	623	501	2 3	623	NO	ABS	CRS	Y	Y		Y	NO	Y	NO	NO
181	Y	Y	18	LT-2787	EDG DAY TANK 1-1 LVL TRANSMITT	AUX	585	318	12347	590	Y	BS	GRS	Y	Y		Y	Y			Y
182		Y	18	LT-CF3B1	CFT 1-1 LEVEL TRANSMITTER	CTM	565	214	1 4	570	Y	BS	GRS	Y	Y		Y	Y			Y
183	Y	Y	18	LT-MU16-1	RC MU TANK LVL TRANSMITTER	AUX	565	AB3	2 4	567	Y	BS	GRS	Y	Y		Y	Y			Y
184	Y	Y	18	LT-RC14-3	RC PRESSURIZER CH 1 LEVEL TRANSMITTE	CTM	585	317	145	589	Y	BS	GRS	Y	Y		Y	Y			Y

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185	Y	Y	18	LT-SP9B3	STEAM GEN 1 STARTUP LEVEL TRANSMITT	CTM	565	285	145	565	Y	BS	GRS	Y	Y		Y	Y			Y
186	Y	Y	7	MS-101	MS LINE 1 ISO VALVE	AUX	643	601	147	647	NO	DOC	CRS	Y	N/A		N/A	Y			Y
187	Y	Y	8A	MS-106	MS LINE 1 TO AFP TURB 1-1 ISO VALVE	AUX	623	500	2 3	623	NO	DOC	CRS	Y	N/A		N/A	Y			Y
188	Y	Y	7	MS-5889A	AFP TURB 1-1 STEAM ADMISSION VALVE	AUX	565	237	3 6	585	Y	BS	GRS	Y	Y		N/A	Y			Y
189	Y	Y	7	MS-5889B	AFP TURB 1-2 STEAM ADMISSION VALVE	AUX	565	238	3 6	585	Y	BS	GRS	Y	Y		N/A	Y			Y
190	Y	Y	7	MU-19	RCP SEAL INJ FLOW CTRL VLV	AUX	585	303	1 7	585	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
191	Y	Y	7	MU-23	BA PMP PNEUMATC DISCH CTRL VLV	AUX	565	240	2 10	571	Y	BS	GRS	Y	Y		N/A	Y			Y
192		Y	8	MU-2A	LETDOWN ISOLATION	CTM	565	214	1 4	578	Y	BS	GRS	Y	Y		N/A	Y			Y
193	Y	Y	8A	MU-2B	RC LETDOWN ISO VALVE	CTM	565	216	1 4	565	Y	BS	GRS	Y	Y		N/A	Y			Y
194	Y	Y	7	MU-32	MU FLOW CTRL VALVE	AUX	565	225	3 6	568	Y	BS	GRS	Y	Y		N/A	Y			Y
195	Y	Y	7	MU-38	RCP SEAL RETURN ISO VALVE	AUX	565	208	567	565	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
196	Y	Y	8A	MU-3971	MU PUMP2 SUCTION 3 WAY MOV	AUX	565	225	3 6	585	Y	BS	GRS	Y	Y		N/A	Y			Y
197	Y	Y	8A	MU-40	BATCH FEED LINE STOP ISO VLV	AUX	565	211	1 8	568	Y	BS	GRS	Y	Y		N/A	Y			Y
198		Y	8A	MU-59A	RCP SEAL RETURN 2-1	CTM	565	214	1 4	565	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
199		Y	8A	MU-59B	RCP SEAL RETURN 2-2	CTM	565	214	1 4	565	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
200		Y	8A	MU-59C	RCP SEAL RETURN 1-1	CTM	565	214	1 4	565	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
201		Y	8A	MU-59D	RCP SEAL RETURN 1-2	CTM	565	214	1 4	565	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
202	Y	Y	8A	MU-6405	RC MU PMP1-1 3-WAY SUCTION VALVE	AUX	565	225	3 6	585	Y	BS	GRS	Y	Y		N/A	Y			Y
203		Y	7	MU-6406	MAKE-UP 2 RECIRC	AUX	565	225	4 6 11	575	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
204		Y	7	MU-6407	MAKE-UP 1 RECIRC	AUX	565	225	4 6 11	575	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
205		Y	7	MU-6409	MAKE-UP 1 DISCH XCONN	AUX	565	225	1 10	571	Y	DOC	CRS	Y	Y		N/A	Y			Y
206	Y	Y	8A	MU-6419	NORMAL MU TO RCS LOOP-1 ISOVLV	AUX	565	208	6 7	565	Y	BS	GRS	Y	Y		N/A	Y			Y
207	Y	Y	8A	MU-6421	MU TO RCS TRAIN2 ISO VALVE	AUX	565	208	6 7	565	Y	BS	GRS	Y	Y		N/A	Y			Y

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208	Y	Y	8A	MU-6422	NORM MU TO RCP SEALS ISO VLV	AUX	565	236	6 7	565	Y	BS	GRS	Y	Y		N/A	Y			Y
209	Y	Y	7	MU-66A	RCP1-2-1 SEAL INJ FLOW ISO VLV	AUX	565	208	6 7	565	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
210	Y	Y	7	MU-66B	P1-2-2 SEAL INJ FLOW CNTRL VLV	AUX	565	208	6 7	565	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
211	Y	Y	7	MU-66C	RCP1-1-1 SEAL INJ FLOW ISO VLV	AUX	565	208	6 7	565	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
212	Y	Y	7	MU-66D	RCP1-1-2 SEAL INJ FLOW ISO VLV	AUX	565	208	6 7	565	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
213		Y	7	NN-236	NITROGEN VALVE ISOLATION	AUX	565	236	2 8	585	Y	BS	GRS	Y	Y		N/A	Y			Y
214	Y	Y	1	NP 1473	EDG 1-1 OIL PUMP CONT BOX CH A	AUX	585	318	7 8	591	Y	BS	GRS	Y	Y		Y	Y			Y
215	Y	Y	20	NV-5305A	L.V.S.G. RM DAMP CTRL STATION	AUX	603	429	2 4	607	Y	BS	GRS	Y	Y		Y	Y			Y
216	Y	Y	20	NV-5305B	L.V.S.G. RM DAMP CTRL STATION	AUX	603	429	2 4	607	Y	BS	GRS	Y	Y		Y	Y			Y
217	Y	Y	20	NV-55970	BATT RM 429B DISCH DMPR LOC SW	AUX	603	429	2 4	607	Y	BS	GRS	Y	Y		Y	Y			Y
218	Y	Y	20	NY-5874B	NEUTRON FLUX MONITORING AMPLIGIER C	AUX	603	402	1 4	608	Y	BS	GRS	Y	Y		Y	Y			Y
219	Y	Y	5	P14-1	AUXILIARY FEEDWATER PUMP 1-1	AUX	565	237	1 7	565	Y	BS	GRS	Y	NO	Y	Y	Y			Y
220	Y	Y	6	P195-1	EDG FUEL OIL TRANSFER PUMP 1-1	YRD	585	N/A	4 6	578	Y	DOC	GRS	Y	N/A		Y	Y			Y
221		Y	5	P197-1	HPI PUMP AC LUBE OIL PUMP	AUX	545	105	6 9	545	Y	BS	GRS	Y	NO	Y	Y	Y			Y
222		Y	5	P197-2	HPI PUMP DC LUBE OIL PUMP	AUX	545	105	6 9	545	Y	BS	GRS	Y	NO	Y	Y	Y			Y
223	Y	Y	6	P3-1	SERVICE WATER PUMP 1-1	ITK	576	052	124	576	Y	DOC	RRS	Y	NO	NO	Y	Y	Y	Y	Y
224	Y	Y	5	P37-1	MAKEUP PUMP 1-1	AUX	565	225	1 4	565	Y	BS	GRS	Y	Y		Y	Y			Y
225	Y	Y	5	P37-2	MAKEUP PUMP 1-2	AUX	565	225	1 4	565	Y	BS	GRS	Y	Y		Y	Y			Y
226	Y	Y	5	P38-1	BORIC ACID PUMP 1-1	AUX	565	240	2 10	565	Y	BS	GRS	Y	Y		N/A	Y			Y
227	Y	Y	5	P38-2	BORIC ACID PUMP 1-2	AUX	565	240	2 10	565	Y	BS	GRS	Y	Y		N/A	Y			Y
228	Y	Y	5	P42-1	DECAY HEAT PUMP 1-1	AUX	545	105	5 6	545	Y	BS	GRS	Y	Y		Y	Y			Y
229	Y	Y	5	P43-1	COMP COOLING PUMP 1-1	AUX	585	328	1367	585	Y	BS	GRS	Y	Y		Y	Y			Y
230	Y	Y	5	P43-3	CC PUMP 1-3	AUX	585	328	1367	585	Y	BS	GRS	Y	Y		Y	Y			Y

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231	Y	Y	5	P56-1	CONTAINMENT SPRAY PUMP 1-1	AUX	545	105	5 6	545	Y	BS	GRS	Y	Y		Y	Y			Y
232	Y	Y	5	P57	BORATED WATER RECIRC PUMP 1-1	AUX	565	209	1 7	565	Y	BS	GRS	Y	Y		Y	Y			Y
233	Y	Y	5	P58-1	HI PRESSURE INJECTION PUMP 1-1	AUX	545	105	5 6	545	Y	BS	GRS	Y	Y		Y	Y			Y
234	Y	Y	18	PDIS 1379A	SW STRNR 1-1 PRESS DIFF IND SW	ITK	576	052	5 6	576	Y	BS	GRS	Y	Y		Y	Y			Y
235		Y	18	PDS 4957	DIFFERENTIAL PRESSURE SENSOR	AUX	545	105	6 9	545	Y	BS	GRS	Y	Y		Y	Y			Y
236	Y	Y	18	PDSH 3981	DG1 JKT CC OUT ISO VLV PDSH	AUX	585	318	1 4	589	Y	BS	GRS	Y	Y		Y	Y			Y
237	Y	Y	18	PI-MU52A	BA PMP 1-1 DISCH LN PRESS INDI	AUX	565	241	4710	569	Y	ABS	CRS	Y	NO	Y	Y	Y			Y
238	Y	Y	18	PS 5301	CTRM H&V SFAS ACT ISO PRES SWT	AUX	638	603	2 11	643	Y	ABS	RRS	Y	Y		Y	Y			Y
239		Y	18	PS MU102B	MK-UP PMP1 OIL PRESS SWTCH	AUX	565	225	7 9	565	Y	BS	GRS	Y	Y		Y	Y			Y
240	Y	Y	0	PSE 226	PRESSURIZER QUENCH TANK RUPTURE DIS	CTM	565	218	1 7	565	Y	EJ	EJ	Y	N/A		N/A	Y			Y
241	Y	Y	0	PSE 5463	PRESSURIZER SAFETY VALVE RUPTURE DIS	CTM	565	218	1 7	609	Y	EJ	EJ	Y	N/A		N/A	Y			Y
242	Y	Y	0	PSE 5464	PRESSURIZER SAFETY VALVE RUPTURE DIS	CTM	565	218	1 7	609	Y	EJ	EJ	Y	N/A		N/A	Y			Y
243	Y	Y	18	PSL 106A	PRESS SWTCH LO FR AFP TURB 1-1 STM IN	AUX	565	237	2 7	569	Y	BS	GRS	Y	Y		Y	Y			Y
244	Y	Y	18	PSL 106B	PRESS SWITCH LOW AT AFP TURB 1-1 SUC	AUX	565	237	2 7	570	Y	BS	GRS	Y	Y		Y	Y			Y
245	Y	Y	18	PSL 106C	PRESS SWITCH LOW FOR AFP TURB 1-1 INL	AUX	565	237	2 7	569	Y	BS	GRS	Y	Y		Y	Y			Y
246	Y	Y	18	PSL 106D	PRESS SWITCH LOW FOR AFP TURB 1-1 INL	AUX	565	237	2 7	570	Y	BS	GRS	Y	Y		Y	Y			Y
247		Y	18	PSL 1376A	SW PMP -1 DISCH SRC TAP PRESS SWITCH	ITK	576	052	6 7	585	Y	BS	GRS	Y	Y		Y	Y			Y
248	Y	Y	18	PSL 3783	EDG STRTNG AIR RCVR 1-1-1 TO..	AUX	585	318	1 4	604	Y	BS	GRS	Y	Y		Y	Y			Y
249	Y	Y	18	PSL 3784	EDG STRTNG AIR RCVR 1-1-2 TO..	AUX	585	318	1 4	604	Y	BS	GRS	Y	Y		Y	Y			Y
250	Y	Y	18	PSL 4930A	AFP 1-1 SUCTION AFTER STRNR PRESS SW	AUX	565	237	3 6	565	Y	BS	GRS	Y	Y		Y	Y			Y
251	Y	Y	18	PSL 4930B	AFP 1-1 SUCTION AFTER STRNR PRESS SW	AUX	565	237	3 6	565	Y	BS	GRS	Y	Y		Y	Y			Y
252	Y	Y	18	PSLL MU66A	PS FOR MU66A	AUX	565	208	567	570	Y	BS	GRS	Y	Y		Y	Y			Y
253	Y	Y	18	PSLL MU66B	PS FOR MU66B	AUX	565	208	567	569	Y	BS	GRS	Y	Y		Y	Y			Y

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254	Y	Y	18	PSLL MU66C	PS FOR MU66C	AUX	565	208	567	569	Y	BS	GRS	Y	Y		Y	Y			Y
255	Y	Y	18	PSLL MU66D	PS FOR MU66D	AUX	565	208	567	570	Y	BS	GRS	Y	Y		Y	Y			Y
256	Y	Y	18	PT-2000	CTMT PRESSURE SFAS CH1 PRESSURE TRA	AUX	603	400	2 3	607	Y	BS	GRS	Y	Y		Y	Y			Y
257	Y	Y	18	PT-2002	CTMT PRESSURE SFAS CH3 PRESSURE TRA	AUX	623	500	2 3	628	NO	GERS	CRS	Y	Y		Y	Y			Y
258	Y	Y	18	PT-5898	CREVS CH 1 REFRIG HEAD PRESS	AUX	638	603	2 11	644	Y	ABS	RRS	Y	Y		Y	Y			Y
259		Y	18	PT-CF4B1	CFT 101 PRESSURE TRANSMITTER	CTM	565	214	1 4	570	Y	BS	GRS	Y	Y		Y	Y			Y
260	Y	Y	18	PT-RC2B4	RCP LOOP 1 HLG WR PRESS TRANS SFAS C	CTM	603	483	145	608	Y	BS	GRS	Y	Y		Y	Y			Y
261	Y	Y	18	PT-SP12B1	STEAM GEN 1-1 OUTLT STEAM PRESS TRA	CTM	585	317	145	589	Y	BS	GRS	Y	Y		Y	Y			Y
262	Y	Y	7	RC 13B	PRESSURIZER CODE SAFETY RELIEF VALVE	CTM	565	218	1 7	609	Y	BS	GRS	Y	Y		N/A	Y			Y
263		Y	7	RC 1719A	RCS DRAIN ISLOATION	CTM	565	220	1 4	578	Y	BS	GRS	Y	Y		N/A	Y			Y
264		Y	7	RC 1773A	RCS DRAIN ISOLATION	CTM	565	220	1 4	566	Y	BS	GRS	Y	Y		N/A	Y			Y
265	Y	Y	8A	RC 200	PRESS VENT LINE STOP VALVE	CTM	585	385	134	589	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
266	Y	Y	7	RC 207	PRZR QUENCH TANK RELIEF VLV TO CTMT	CTM	585	218	145	565	Y	BS	GRS	Y	Y		N/A	Y			Y
267		Y	7	RC 229A	PZR QUENCH TANK ISOLATION	AUX	565	225	6 11	566	Y	BS	GRS	Y	Y		N/A	Y			Y
268	Y	Y	8A	RC 239A	PRESS VAPOR PHASE SAMPLE ISO VALVE	CTM	585	385	134	589	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
269	Y	Y	0	RC 2A	PRZR PWR RELIEF VALVE (SOL PILOT OP)	CTM	623	580	1 4	636	NO	DOC	RRS	Y	N/A		N/A	Y			Y
270	Y	Y	10	S33-1	CREVS WATER COOLED COND 1	AUX	638	603	2 11	638	NO	ABS	RRS	Y	NO	NO	Y	Y	Y	NO	NO
271	Y	Y	10	S61-1	CREVS AIR COOLED CONDENSER 1	AUX	660	N/A	2 11	660	NO	DOC	RRS	Y	Y		Y	Y			Y
272	Y	Y	7	SP-17B1	MS LINE 1 CODE SAFETY VALVE (PSVSP17	AUX	643	601	6 9	660	NO	DOC	CRS	Y	N/A		N/A	Y			Y
273	Y	Y	7	SP-17B2	MS LINE 1 CODE SAFETY VALVE (PSVSP17	AUX	643	601	6 9	660	NO	DOC	CRS	Y	N/A		N/A	Y			Y
274	Y	Y	7	SP-17B3	MS LINE 1 CODE SAFETY VALVE (PSVSP17	AUX	643	601	6 9	660	NO	DOC	CRS	Y	N/A		N/A	Y			Y
275	Y	Y	7	SP-17B4	MS LINE 1 CODE SAFETY VALVE (PSVSP17	AUX	643	601	6 9	660	NO	DOC	CRS	Y	N/A		N/A	Y			Y
276	Y	Y	7	SP-17B5	MS LINE 1 CODE SAFETY VALVE (PSVSP17	AUX	643	601	6 9	660	NO	DOC	CRS	Y	N/A		N/A	Y			Y

LINE NO	A 46	IP EEE	EQ CLASS	EQ NO	EQUIPMENT DESCRIPTION	BLDG	ELEV	RM	SQUGGER	BASE ELEV	<40	SPECTRUM CAP	SPECTRUM DEMAND	CAP> DEMAND	CAVEATS WORD	CAVEATS INTEN	ANCH OK	INTER-ACT OK	OUTLIER Y/N	OUTLIER OK	EQUIP OK
277	Y	Y	7	SP-17B6	MS LINE 1 CODE SAFETY VALVE (PSVSP17	AUX	643	601	6 9	660	NO	DOC	CRS	Y	N/A		N/A	Y			Y
278	Y	Y	7	SP-17B7	MS LINE 1 CODE SAFETY VALVE (PSVSP17	AUX	643	601	6 9	660	NO	DOC	CRS	Y	N/A		N/A	Y			Y
279	Y	Y	7	SP-17B8	MS LINE 1 CODE SAFETY VALVE (PSVSP17	AUX	643	601	6 9	660	NO	DOC	CRS	Y	N/A		N/A	Y			Y
280	Y	Y	7	SP-17B9	MS LINE 1 CODE SAFETY VALVE (PSVSP17	AUX	643	601	6 9	660	NO	DOC	CRS	Y	N/A		N/A	Y			Y
281	Y	Y	7	SS-607	STEAM GEN 1-1 SAMPLE LINE CTMT ISO V	AUX	585	314	1 7	598	Y	BS	GRS	Y	NO	Y	N/A	Y			Y
282	Y	Y	8B	SV-5301	AUX BLDG CTRM DMPR AIR SOL VLV	AUX	638	603	2 4	648	NO	GERS	CRS	Y	Y		Y	Y			Y
283	Y	Y	8B	SV-5301A	CTRM COMP CONF RM&COMP..SOLVLV	AUX	638	603	2 4	649	NO	GERS	CRS	Y	Y		Y	Y			Y
284	Y	Y	7	SW-1356	CAC 1-1 OUTLET TEMP CTRL VALVE	AUX	585	314	6 7	585	Y	BS	GRS	Y	Y		N/A	Y			Y
285	Y	Y	8A	SW-1379	SW STRNR 1-1 DRAIN VALVE	ITK	576	052	12347	576	Y	BS	GRS	Y	Y		N/A	Y			Y
286	Y	Y	8A	SW-1382	SW SUPPLY TO AFP 1-1 ISO VALVE	AUX	565	237	4 6	565	Y	BS	GRS	Y	Y		N/A	Y			Y
287		Y	8A	SW-1399	SW LOOP 1 TO TPCW HX	ITK	585	053	6 7	566	Y	BS	GRS	Y	Y		N/A	Y			Y
288	Y	Y	7	SW-1424	CCW HT XCHANG 1-1 OUT CTRL VLV	AUX	585	328	6 7	585	Y	BS	GRS	Y	Y		N/A	Y			Y
289	Y	Y	8A	SW-2927	CTRM EMERG COND 1-1 TV...VALVE	AUX	638	603	1 7	638	NO	GERS	CRS	Y	Y		N/A	Y			Y
290	Y	Y	8A	SW-2929	SW DISCH TO IN STRUCTURE VALVE	ITK	566	053	12347	566	Y	BS	GRS	Y	Y		N/A	Y			Y
291	Y	Y	8A	SW-2930	SW DISCH TO IN FOREBAY VALVE	ITK	566	053	12347	566	Y	BS	GRS	Y	Y		N/A	Y			Y
292	Y	Y	8A	SW-2931	SW DISCH TO COOLING TWR MU VLV	ITK	566	053	12347	566	Y	BS	GRS	Y	Y		N/A	Y			Y
293	Y	Y	8A	SW-2932	SW DISCH TO COLLECT BASIN VLV	ITK	566	053	12347	566	Y	BS	GRS	Y	Y		N/A	Y			Y
294		Y	8A	SW-5421	SW OUTLET MOV FOR E42-5	AUX	545	105	1 10	546	Y	BS	GRS	Y	Y		N/A	Y			Y
295		Y	8A	SW-5422	SW OUTLET MOV E42-4	AUX	545	105	1 10	546	Y	BS	GRS	Y	Y		N/A	Y			Y
296	Y	Y	7	SW-5896	CTRM EMERG COND 1-1 SW VLV	AUX	638	603	1 7	638	NO	GERS	CRS	Y	NO	Y	N/A	Y			Y
297	Y	Y	21	T10	BORATED WATER STORAGE TANK 1-1	YRD	585	N/A	2 8	585	Y	N/A	N/A	Y	N/A		Y	Y			Y
298	Y	Y	21	T12-I	COMPONENT COOLING SURGE TNK I	AUX	623	501	2 3	623	NO	N/A	N/A	Y	N/A		NO	NO	Y	Y	NO
299	Y	Y	21	T153-1	EDG FUEL OIL STORAGE 1-1	YRD	585	N/A	6 9	585	Y	N/A	N/A	N/A	N/A		Y	Y			Y

LINE NO	A 46	IP EEE	EQ CLASS	EQ NO	EQUIPMENT DESCRIPTION	BLDG	ELEV	RM	SQUGGER	BASE ELEV	<40	SPECTRUM CAP	SPECTRUM DEMAND	CAP> DEMAND	CAVEATS WORD	CAVEATS INTEN	ANCH OK	INTER-ACT OK	OUTLIER Y/N	OUTLIER OK	EQUIP OK
300		Y	21	T18	SFP DEMINERALIZER TANK 1-1	AUX	565	233	2 4	565	Y	N/A	N/A	Y	N/A		Y	N/A			Y
301		Y		T198-1	HEAD TANK FOR HPI 1-1	AUX	545	105	2 4	554	Y	N/A	N/A	Y	N/A		Y	Y			Y
302		Y		T199-1	LUBE OIL RESERVOIR FOR HPI 1-1	AUX	545	105	2 4	545	Y	N/A	N/A	Y	N/A		Y	Y			Y
303		Y	21	T2	PRESSURIZER	CTM	565	218	1 2 4	609	N	N/A	N/A	N/A	N/A		Y	Y			Y
304	Y	Y	21	T4	MAKEUP TANK 1-1	AUX	565	205	1 8	565	Y	N/A	N/A	N/A	N/A		Y	Y	Y	Y	Y
305	Y	Y	21	T46-1	EDG DAY TANK 1-1	AUX	595	321	367	595	Y	N/A	N/A	N/A	N/A		Y	Y			Y
306	Y	Y	21	T7-1	BORIC ACID ADDITION TANK 1-1	AUX	565	240	7 8	565	Y	N/A	N/A	N/A	N/A		NO	Y	Y	Y	Y
307	Y	Y	21	T7-2	BORIC ACID ADDITION TANK 1-2	AUX	565	240	7 8	565	Y	N/A	N/A	N/A	N/A		NO	Y	Y	Y	Y
308	Y	Y	12	T86-1	EDG 1-1 AIR RECEIVER 1-1-1	AUX	585	318	12467	595	Y	BS	GRS	Y	Y		Y	Y			Y
309	Y	Y	12	T86-2	EDG 1-1 AIR RECEIVER 1-1-2	AUX	585	318	12467	595	Y	BS	GRS	Y	Y		Y	Y			Y
310		Y	21	T9-1	CORE FLOOD TANK 1-1	CTM	585	316	2 4	585	Y	N/A	N/A	N/A	N/A		Y	Y			Y
311	Y	Y	18	TE-1356	CTMT COOLER FAN 1 SUCTION TEMP ELEM	CTM	585	317	1 4	585	Y	ABS	CRS	Y	Y		Y	Y			Y
312	Y	Y	18	TE-5329	EDG RM 318 TEMP ELEMENT	AUX	585	318	12347	590	Y	BS	GRS	Y	Y		Y	Y			Y
313	Y	Y	18	TE-5443	CC PMP 1 RM TEMP ELEMENT	AUX	585	328	6 7	595	Y	BS	GRS	Y	NO	Y	Y	Y			Y
314	Y	Y	19	TE-IM07M	INCORE OUTLET M7 TEMP ELEMENT	CTM	578	315	1 7	578	Y	BS	GRS	Y	Y		N/A	Y			Y
315	Y	Y	19	TE-RC3B5	RC LOOP 1 HLG WR TEMP ELEMENT	CTM	565	216	1 4	641	NO	DOC	CRS	Y	Y		N/A	Y			Y
316	Y	Y	19	TE-RC4B2	RCP 1-1 DISCH CLG WR TEMP ELEMENT	CTM	565	216	145	571	Y	BS	GRS	Y	Y		N/A	Y			Y
317	Y	Y	0	TI 5504	PORTABLE RC TEMPERATURE INDICATOR	AUX	585	304	2 11	585	Y	DOC	CRS	Y	N/A		N/A	Y			Y
318	Y	Y	18	TIC 5443	CC PMP 1 RM TEMP INDEX CONTROL	AUX	585	328	6 7	585	Y	BS	GRS	NO	Y	Y	Y	Y			Y
319	Y	Y	18	TS-4688	TEMP SWT FR XHAUST FAN C99-1&2	ITK	576	052	5 6	580	Y	BS	GRS	Y	NO	Y	Y	Y			Y
320	Y	Y	18	TS-5135	TEMP SWITCH FOR AFP ROOM VENT FAN 1	AUX	565	237	2 7	570	Y	BS	GRS	Y	Y		Y	Y			Y
321	Y	Y	18	TS-5261	CTRM EMERG VENT FAN 1 TEMP SWT	AUX	638	603	1 7	638	NO	ABS	RRS	Y	Y		Y	Y			Y
322	Y	Y	18	TS-5318	L.V.S.G. RM DAMP TEMP SWITCH	AUX	603	429	2 4	607	Y	BS	GRS	Y	Y		Y	Y			Y

LINE NO	A 46	IP EEE	EQ CLASS	EQ NO	EQUIPMENT DESCRIPTION	BLDG	ELEV	RM	SQUGGER	BASE ELEV	<40	SPECTRUM CAP	SPECTRUM DEMAND	CAP> DEMAND	CAVEATS WORD	CAVEATS INTEN	ANCH OK	INTER-ACT OK	OUTLIER Y/N	OUTLIER OK	EQUIP OK
323	Y	Y	18	TS-5443	CC PMP RM VNT FN 1 TEMP SWITCH	AUX	585	328	6 7	595	Y	BS	GRS	Y	Y		Y	Y			Y
324	Y	Y	18	TS-5597	TEMP SW FR BATT RM A THERMO	AUX	603	429	367	608	Y	BS	GRS	Y	Y		Y	Y			Y
325	Y	Y	18	TSH 1483	CC HX CCW OUT TEMP SWICH HIGH	AUX	585	328	6 7	595	Y	BS	GRS	Y	Y		Y	Y			Y
326	Y	Y	18	TSH 5421	ECCS RM CLR FAN 1-5 TEMP SW	AUX	545	105	5 6	549	Y	BS	GRS	Y	Y		Y	Y			Y
327	Y	Y	18	TSH 5422	ECCS RM CLR FAN 1-4 TEMP SW	AUX	545	105	5 6	551	Y	BS	GRS	Y	Y		Y	Y			Y
328	Y	Y	18	TSL 5421	ECCS RM CLR FAN 1-5 TEMP SW	AUX	545	105	5 6	549	Y	BS	GRS	Y	Y		Y	Y			Y
329	Y	Y	18	TSL 5422	ECCS RM CLR FAN 1-4 TEMP SW	AUX	545	105	5 6	549	Y	BS	GRS	Y	Y		Y	Y			Y
330	Y	Y	18	TT-1356	CTMT COOLER FAN 1 SUCTION TEMP TRAN	AUX	585	303	1 7	590	Y	BS	GRS	Y	NO	NO	NO	Y	Y	NO	NO
331	Y	Y	18	TT-5443	CC PMP 1 RM TEMP TRANSMITTER	AUX	585	328	6 8	603	Y	BS	GRS	Y	NO	Y	Y	Y			Y
332	Y	Y	14	Y1	ESSEN INSTR DIST PNL "Y1"	AUX	603	429	2 3	603	Y	BS	GRS	Y	Y		Y	Y			Y
333	Y	Y	14	Y1A	120VAC ESSEN INST DIST PANEL	AUX	603	429	2 3	603	Y	BS	GRS	Y	Y		Y	Y			Y
334	Y	Y	14	Y2	ESSEN INSTR DIST PNL "Y2" 120V	AUX	603	428	8 9	603	Y	BS	GRS	Y	Y		Y	Y			Y
335	Y	Y	14	Y2A	120VAC ESSEN INST DIST PANEL	AUX	603	428	1 9	603	Y	BS	GRS	Y	Y		Y	Y			Y
336	Y	Y	14	Y3	ESSEN INSTR DIST PNL "Y3" 120V	AUX	603	429	2 3	603	Y	BS	GRS	Y	Y		Y	Y			Y
337	Y	Y	14	Y4	ESSEN INSTR DIST PNL "Y4" 120V	AUX	603	428	489	603	Y	BS	GRS	Y	Y		Y	Y			Y
338	Y	Y	14	YAU	UPS INSTR DIST PNL "YAU"	AUX	603	429	489	603	Y	BS	GRS	Y	Y		Y	Y			Y
339	Y	Y	1	YE1	480/120 VAC MCC/TRANSFORMER	AUX	585	318	1489	585	Y	BS	GRS	Y	Y		Y	NO	Y	NO	NO
340	Y	Y	1	YE2	240 VAC MCC/TRANSFORMER	AUX	585	304	1489	585	Y	BS	GRS	Y	Y		Y	Y			Y
341	Y	Y	4	YE2A	480-240V TRANSFORMER	AUX	603	405	1367	603	Y	BS	GRS	Y	Y		Y	Y			Y
342	Y	Y	1	YF2	240 VAC MCC/TRANSFORMER	AUX	603	427	489	585	Y	BS	GRS	Y	Y		Y	Y			Y
343	Y	Y	16	YV1	125VDC/120VAC INVERTER CH 1	AUX	603	429	237	603	Y	BS	GRS	Y	Y		Y	Y			Y
344	Y	Y	16	YV2	125VDC/120VAC INVERTER CH 2	AUX	603	428	2 3	603	Y	BS	GRS	Y	Y		Y	NO	Y	NO	NO
345	Y	Y	16	YV3	125VDC 120VAC INVERTER CH 3	AUX	603	429	1 7	603	Y	BS	GRS	Y	Y		Y	NO	Y	NO	NO

LINE NO	A 46	IP EEE	EQ CLASS	EQ NO	EQUIPMENT DESCRIPTION	BLDG	ELEV	RM	SQUGGER	BASE ELEV	<40	SPECTRUM CAP	SPECTRUM DEMAND	CAP> DEMAND	CAVEATS WORD	CAVEATS INTEN	ANCH OK	INTER-ACT OK	OUTLIER Y/N	OUTLIER OK	EQUIP OK
346	Y	Y	16	YV4	125VDC/120VAC INVERTER CH 4	AUX	603	428	4 7	603	Y	BS	GRS	Y	Y		Y	NO	Y	NO	NO
347	Y	Y	16	YVA	UPS "YVA" INVERTER	AUX	603	429	4 6	603	Y	ABS	RRS	NO	NO	NO	NO	Y	Y	NO	NO
348	Y	Y	18	ZC-6452	AFP 1-1 DISCH CTRL VLV POS CONTROLER	AUX	565	237	4 6	569	Y	BS	GRS	Y	Y		Y	Y			Y

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PART 4
INTERNAL FIRE ANALYSIS

Contents

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List of Tables

List of Illustrations

-.0 INTERNAL FIRES

An assessment of the risk from internal fires is included as a part of the overall IPEEE analysis. This section discusses and describes the analyses associated with this portion of the IPEEE.

The discussion is organized as:

4.1 Methodology for Fire Analysis - A brief overview of the methodology used to evaluate the risk associated with internal fires. Description of the Phase I Qualitative Evaluation and Screening process.

4.2 Quantitative Analysis and Screening- Description of Phase II and III of the screening process

4.3 Evaluation of Containment Fires

4.4 Assessment of Outliers for Fire Hazard - Discussion of any fire areas which did not screen through the FIVE process

4.5 Assessment of Other Fire Issues - Discussion of Sandia Fire Issues and other relevant issues.

4.6 Verification and Walkdowns - Description of the verification process used to ensure the accuracy of the fire evaluation process.

4.7 USI A-45, NUREG/CR-5088 and GI 57 - Description of how these specific items are resolved.

4.1 Methodology for Fire Analysis

After reviewing the techniques available for evaluating the effects of fire on the plant, Toledo Edison elected to use the Fire Induced Vulnerability Evaluation (FIVE) (Ref. 1) methodology to perform this work. The FIVE approach was developed by the Electric Power Research Institute (EPRI) and had been reviewed by the NRC prior to issuance. This afforded the opportunity to have a previously accepted approach to help ensure the proper concerns were being addressed. It also provided specific criteria for what would be acceptable levels of risk without excessive analysis. This permitted Toledo Edison to concentrate resources on areas of higher risk.

The FIVE methodology is a screening technique based on conservative assumptions using generic and plant specific data bases for evaluating fire event sequences. The overall objective is to determine the availability of plant systems, components, and cabling to achieve and maintain safe and stable shutdown of the reactor and thereby prevent core damage. The process considers all plant areas and focuses on (but is not limited to) the availability of 10CFR50 Appendix R (Ref. 2) safe shutdown equipment. Appendix R equipment was initially selected because it has been previously reviewed to ensure a minimum set of components that would be free of fire damage so that safe shutdown can be accomplished under the conditions and criteria established in Appendix R.

Once the fire risk evaluation was started, it was quickly learned that the basic FIVE methodology :med to have been developed with newer vintage plants in mind. The design requirements regarding equipment separation have been refined since Davis-Besse was designed and constructed. As a result, few of the Davis-Besse fire compartments screened in the first qualitative phases of the FIVE process. Although many compartments screened by applying the quantitative criterion in FIVE, it also became necessary to modify the FIVE process to evaluate Davis-Besse. The modifications to the FIVE process are described in detail in each section of this report.

Briefly, the changes included more detailed analysis of affected circuits, improved fire initiation frequency quantification, inclusion of fire effects evaluations, and accrediting of fire prevention and suppression activities at the site. These modifications were primarily taken from the EPRI sponsored Fire PRA Implementation Guide (Ref. 3). This Guide draws on the FIVE methodology, adds data from NUREG/CR-4840 and NUREG/CR-2815, and includes insights from the SANDIA and EPRI fire research programs. This allows removal of some of the conservatism included in the original FIVE methodology, but ensures that the safe shutdown capability is maintained.

The FIVE methodology is itself an extension of the 10 CFR 50 Appendix R Fire Hazards Analysis of the plant. The following assumptions were made in the Fire Hazards Analysis Report (FHAR, Ref. 4) analyses:

1. The unit is operating at 100% power.
2. All equipment located in the area of the fire and equipment with cables routed through the fire area fails, unless it can be shown that separation and protection meet the requirements of Section III.G of Appendix R.
3. No equipment failures beyond the fire area need be considered.
4. All embedded conduits are not affected by the fire.
5. Associated circuits that are coordinated are not included since they are designed to electrically clear faults before causing the loss of safe shutdown systems and components.
6. Offsite power is assumed to be lost.

FIVE allows for random, routine failure of modeled equipment that is not damaged by the fire itself. Thus, the depth of defense required to demonstrate safe shutdown capability is greater in the FIVE analysis when compared to the FHAR work. The Davis-Besse IPEEE fire analyses also modified the loss of offsite power assumption of the FHAR. Offsite power was generally assumed to be available, subject to normally modeled random failures. This was not assumed if losing offsite power, due to the presence of control or power cables for the offsite power supplies in a compartment, resulted in higher conditional core damage probabilities.

The FIVE process consists of several phases. Each phase is described in detail in the sections below, but a brief overview will be given here. The first phase of FIVE included identification of fire area and compartment boundaries together with safe shutdown equipment and the route of supporting electrical bles in the plant. This information was qualitatively evaluated to determine if there were any plant locations which could be screened out due to the absence of any safe shutdown equipment or cables or the need for plant shutdown. Compartment fire barriers of the plant were also evaluated to ensure that any eened out compartments could not cause a fire in any compartment that could not be screened out. This evaluation is discussed in the Fire Compartment Interaction Analysis (FCIA) section of the report.

The second phase of FIVE used probabilistic risk analysis for plant areas and compartments that did not pass the initial qualitative screening criteria. This analysis involved the consideration of equipment failures beyond those caused by the fire. Plant areas that had a fire-induced core melt frequency below the FIVE specified cutoff value were screened from further evaluation. Because the equipment included in the FHAR analyses was the minimum required to assure safe shut down, with no additional failures, some areas would not screen due to relatively high failure rates of the remaining train of Appendix R equipment (i.e., the loss of both Emergency Diesel Generators due to a fire in one of the EDG rooms and the random failure of the other EDG). However, other systems beyond the Appendix R equipment were available for inclusion in the probabilistic risk analysis (such as offsite power and the Station Blackout Diesel). As a result of the PRA work, it was necessary to expand the equipment included in the safe shutdown equipment list. (It was not necessary to update the Appendix R safe shutdown equipment list, however. Those analyses remain valid.) The equipment expansion was needed only because of the change in the assumption regarding equipment failures.

The third phase of FIVE was to do detailed fire analysis of the remaining unscreened compartments. This work entailed incorporation of the Fire PRA Implementation Guide information, detailed evaluation of the potential for fire damage due to specific fires within an area, and detailed ·aluation of the function of individual cables within the safe shutdown equipment circuitry. The results of .ese evaluations permitted modification of the fire induced equipment failure lists that were input to the PRA program. Reanalysis including these changes then allowed more compartments to be screened.

Specific analyses for the Control Room and the containment were required, due to the unique features of these areas. The Control Room analysis included consideration for the need to evacuate the area due to the fire and the potential effect of a fire in each individual cabinet. The full analysis is described in a following section. The containment analysis was qualitative, but considered the potential for fire spread and interfacing system LOCAs. The full evaluation is also described below.

It was not possible to establish a fire-induced core melt frequency below the FIVE screening criteria for several areas. These compartments are discussed in the Results section of this portion of the report.

4.1.1 Safe Shutdown Equipment

The first phase of the FIVE methodology requires the identification of equipment needed to complete a safe shutdown of the plant and identification of equipment that could cause a plant shutdown. The shutdown can be either as a result of fire damage or as needed to comply with plant Technical Specifications. The scope of the Technical Specification compliance criterion is limited to Actions Statements requiring shut down within eight hours. Safe shutdown is defined as achieving and maintaining the reactor subcritical, maintaining reactor coolant inventory, and maintaining safe and stable ·utdown conditions. This is consistent with the safety functions that were defined in the Davis-Besse PRA ..odel.

The Davis-Besse Fire Hazards Analysis Report has identified those systems that will be relied upon achieve safe shutdown as a result of a fire in any particular fire area. The basis for selection was generally whichever train of equipment (train 1 or train 2) had the least amount of components and cables within the area. The equipment selected receives essential power so that it would not be affected by a loss of offsite power (LOOP), as the FHAR assumed that a LOOP occurs simultaneously with the fire. To support the FHAR analysis, all the cables for the components in the safe shutdown systems were identified and located on the plant Electrical and Raceway drawings. The FHAR lists all the safe shutdown systems' cables by fire area. This mapping was limited to those systems and components required to achieve safe shutdown during a fire. Other system cables were not mapped into the FHAR database.

FIVE subdivides the FHAR fire areas into fire compartments. The systems used to achieve safe shutdown in each compartment are usually the same as for the fire area as a whole. Minor variations occurred primarily when both trains are designated the safe shutdown train in a fire area. The subdivisions allowed more specific train assessment.

The systems within a nuclear power plant interact extensively. A system that is not required for safe shutdown may initiate a reactor trip or require a shutdown. As stated above, the route of non-safe shutdown system cables is not known as a function of fire area or fire compartment. Rather than attempt to analyze each cable in the plant to determine if it could cause or require a plant shutdown, it was conservatively assumed that every fire compartment contained at least one such cable or piece of equipment. Because of this assumption, a reactor trip was always the initiating transient for the core damage probability analyses. Further, because of the complex nature of the Main Feedwater system, a loss of Main Feedwater was generally assumed to occur along with the reactor trip. Some fire areas were investigated in more detail, to verify the absence of any cables that could cause a Loss of Main Feedwater. For those cases, the nominal unavailability for the MFW system was used in the estimation of core-damage.

A comparison of the plant systems modeled in the PRA fault trees and the FHAR safe shutdown equipment list was performed. The PRA models included all the systems required by the FHAR, as well as other systems not included in the FHAR (for example, offsite power, component cooling water ventilation, and instrument air). Some of the equipment associated with these systems were added to the list of safe shutdown equipment for the IPEEE to improve the realistic modeling of the plant. The cables were mapped and added to databases (discussed in detail below). It was not necessary to add this equipment to the FHAR safe shutdown equipment list, however, since safe shutdown per Appendix R requirements can be demonstrated without it.

1.2 Identification of Fire Areas

The initial step of fire area identification requires establishing the requirements of the fire area. The FIVE manual defines a fire area as:

- An area, as defined in the Appendix R analysis, sufficiently bounded by fire barriers that will withstand the fire hazards within the fire area and, as necessary, to protect important equipment within a fire area from a fire outside the area. A fire area must be made up of fire barriers having at least a 2 hour fire rating or equivalent, with openings in the barriers provided with fire doors, fire dampers, and fire penetration seal assemblies having a fire resistance rating at least equivalent to the barrier in which it is installed.
- Fire area boundaries must be completely sealed with floor-to-ceiling and/or wall-to-wall fire barriers or where such boundaries are not wall-to-wall or floor-to-ceiling with all penetrations sealed to the fire rating required of the boundaries; an evaluation must have been performed by a fire protection engineer and, if required, a system engineer to assess the adequacy of the fire area boundaries to determine whether they can withstand the fire hazards within the area and protect important equipment in the area from a fire outside the area.

The basic fire areas used in the IPEEE fire analysis were taken from the FHAR designations. The FHAR-established fire areas are based on plant drawings and the Barrier Function List (Ref. 27). Each 'efined fire area was evaluated for fire propagation control to ensure that a fire could not spread from one ea to another in three dimensional space. The FHAR contains complete analyses of the fire area boundaries.

The FIVE methodology makes provision for subdividing fire areas into fire compartments. A fire compartment is defined in the FIVE manual as, "a space bounded by non-combustible barriers where heat and products of combustion from a fire within the enclosure will be substantially confined." By inspecting the fire area drawings (References 5 through 11), compartments which met this definition were identified. The definition does not include a specific requirement for the fire rating of the barriers as the fire area barrier definition does, nor does it require rigorous inspection and maintenance of the compartment barriers as the area barrier definition does. The specific rooms included in each fire compartment are listed in Table 4.1.3.1. As can be seen, there are a number of single compartment fire areas. The potential for fire spread between compartments in a fire area is evaluated in Section 4.1.5.

4.1.3 Summary of Safe Shutdown Equipment Locations

The safe shutdown systems considered for this analysis are those systems that have been identified as meeting the requirements of Section III.G of Appendix R to 10CFR50. These systems are identified in Section 3 of the Davis-Besse Unit 1 Fire Hazard Analysis Report (FHAR). The FHAR identifies a minimum set of plant systems and components necessary to achieve the functional goals and assure compliance with the requirements of Appendix R.

Section 4.6 of the FHAR identifies the safe shutdown systems in each of the fire areas. The FHAR ɜlf does not, however, break down the location of safe shutdown systems into individual compartments. The location of safe shutdown equipment and the routing of associated cables and wires required the development of a new database which drew on two existing databases and added further information. The first database is the list of safe shutdown equipment broken down into individual components in each fire area based on work that supported the FHAR. This information is controlled as a part of the Updated Safety Analysis Report (USAR). A second database, known as SETROUTE, identifies the conduits and raceways through which the wires and cables for each plant component run. An extensive manual effort was made to identify the conduits and raceways located in each room and thereby each fire compartment. This was done by manually identifying the cables on plant electrical raceway and grounding drawings. By correlating the safe shutdown equipment components from the FHAR with the conduit and raceway information for each plant component through the manually obtained data, a new database was obtained which would allow the determination of the location by room of each component and wire associated with a given safe shutdown system.

The final database contains correlatable information on a given plant component, the cable/wire number, the conduit/raceway, the room number, the fire compartment, the associated probabilistic risk assessment basic event code from the PRA analysis, and other information. The safe shutdown systems location by compartment derived from this database is given in Table 4.1.3.1. Compartments which contain no safe shutdown equipment are so indicated. A description of each fire compartment is given in Table 4.1.3.2.

Table 4.1.3.1

Location of Appendix R Safe Shutdown Systems

Area	Compartment	Safe Shutdown Systems in Compartment
A	A.01	RCS
	A.02	CCWS, DHRS, ESSPWR, MUPS, RCS
	A.03	CSS, DHRS, HPIS
	A.04	AFWS, CCWS, CSS, DHRS, ESSPWR, HPIS, MUPS, NNI
	A.05	AFWS, CCWS, CREVS, CSS, DHRS, EDG, ESSPWR, HPIS, HVAC, MUPS, SFRCS, SWS
	A.07	AFWS, CCWS, CSS, DHRS, EDG, ESSPWR, HPIS, HVAC, MUPS, NNI, SFAS, SFRCS, SWS
	A.08	AFWS, CACS, CCWS, CSS, DHRS, ESSPWR, MSS, MUPS, NI, NNI, RCS, SFAS, SFRCS, SWS
	A.09	AFWS, CCWS, CSS, DHRS, EDG, ESSPWR, HPIS, MUPS, NNI, SFAS, SFRCS, SWS
AB	AB.01	AFWS, CCWS, CSS, DHRS, ESSPWR, HPIS, MUPS, NNI
	AB.02	AFWS, CCWS, CFS, DHRS, MUPS, NI, NNI, RCS, SFAS, SFRCS
	AB.03	AFWS, DHRS, HPIS, MUPS, NNI, SFAS
	AB.04	CSS, DHRS, MUPS, NNI
	AB.05	AFWS, CCWS, CFS, CSS, DHRS, ESSPWR, HVAC, MSS, MUPS, NI, NNI, RCS, SFAS, SFRCS, SWS
	AB.06	ESSPWR
	AC.01	DHRS, SFAS
B	B.01	AFWS, DHRS, ESSPWR, HPIS, MUPS
BD	BD.01	HVAC, SWS
BE	BE.01	EDG, ESSPWR, HVAC, SWS
BF	BF.01	EDG, ESSPWR, HVAC, SWS
BG	BG.01	EDG, ESSPWR, SWS
BH	BH.01	EDG
BM	BM.01	EDG, HVAC
BN	BN.01	(NONE)
CC	CC.01	AFWS, CREVS, EDG, ESSPWR, HPIS, NI, NNI, SFAS, SFRCS, SWS
	CC.02	AFWS, CCWS, CFS, CSS, DHRS, ESSPWR, HPIS, HVAC, MSS, MUPS, NI, NNI, RCS, SFAS, SFRCS,SWS

Table 4.1.3.1

Location of Appendix R Safe Shutdown Systems (continued)

Area	Compartment	Safe Shutdown Systems in Compartment
D	D.01	AFWS, CCWS, CFS, DHRS, EDG, ESSPWR, HVAC, MUPS, NI, NNI, RCS, SFAS, SFRCS
	D.02	ESSPWR, NNI, RCS
	D.03	AFWS, CACS, CCWS, CFS, DHRS, EDG, ESSPWR, HVAC, MUPS, NI, NNI, RCS, SFAS, SFRCS,SWS
	D.04	DHRS, MUPS, NI, NNI, RCS, SFRCS
DD	DD.01	AFWS, CACS, CCWS, CFS, CREVS, CSS, DHRS, EDG, ESSPWR, HPIS, HVAC, MSS, MUPS, NI,NNI,RCS,SFAS,SFRCS,SWS
DF	DF.01	AFWS, CACS, CCWS, CFS, CREVS, CSS, DHRS, EDG, ESSPWR, HPIS, HVAC, MSS, MUPS, NI,NNI,RCS,SFAS,SFRCS,SWS
DG	DG.01	AFWS, DHRS, ESSPWR, MUPS, NI, NNI, RCS, SFAS
DH	DH.01	AFWS, CREVS, MSS, SFRCS, SWS
	DH.02	AFWS, MSS
	E.01	AFWS, CCWS, CREVS, DHRS, EDG, ESSPWR, HPIS, HVAC, MUPS, SFRCS, SWS
EE	EE.01	AFWS, CCWS, CREVS, ESSPWR, HVAC, MSS, MUPS, RCS, SFAS, SFRCS, SWS
F	F.01	AFWS, CCWS, DHRS, EDG, ESSPWR, HPIS, HVAC, SFRCS, SWS
FF	FF.01	AFWS, CACS, CCWS, CFS, CREVS, CSS, DHRS, EDG, ESSPWR, HPIS, MSS, MUPS, NI, NNI,RCS,SBODG,SFAS,SFRCS,SWS
	FF.02	NNI
	FF.03	(NONE)
G	G.01	AFWS, DHRS, SFAS
	G.02	AFWS, CCWS, CSS, DHRS, ESSPWR, HPIS, HVAC, MUPS, NNI, RCS, SFAS, SFRCS, SWS
	G.03	MUPS
	G.04	AFWS, CCWS, ESSPWR, HVAC, MUPS, SFRCS, SWS
HH	HH.01	AFWS, CREVS, ESSPWR, MSS, SWS
II	II.01	AFWS, EDG, ESSPWR, MSS, MUPS, SFRCS, SWS
	II.02	(NONE)
	II.03	(NONE)
	II.04	(NONE)
	II.05	(NONE)

Table 4.1.3.1

Location of Appendix R Safe Shutdown Systems (continued)

Area	Compartment	Safe Shutdown Systems in Compartment
II	II.06	(NONE)
	II.07	(NONE)
	II.08	(NONE)
	II.09	CREVS, ESSPWR, SWS
J	J.01	AFWS, CCWS, EDG, ESSPWR, HVAC, MUPS, NNI
	J.02	AFWS, EDG, HVAC
K	K.01	AFWS, CCWS, DHRS, EDG, ESSPWR, HVAC, MUPS
	K.02	EDG
MA	MA.01	ESSPWR, SWS
MB	MB.01	AFWS
MC	MC.01	AFWS
	ME.01	EDG
MF	MF.01	EDG
MG	MG.01	(NONE)
OS	OS.01	EDG
P	P.01	AFWS, EDG
	P.02	AFWS, EDG, ESSPWR, HVAC
	P.03	AFWS, CCWS, EDG, ESSPWR, HVAC, MUPS, NI, NNI, RCS, SBODG, SFAS, SFRCS, SWS
Q	Q.01	AFWS, CCWS, CSS, DHRS, EDG, ESSPWR, HPIS, HVAC, MUPS, NNI, SBODG, SFRCS, SWS
R	R.01	AFWS, CCWS, ESSPWR, MUPS, NNI, RCS, SFRCS, SWS
S	S.01	AFWS, CCWS, CREVS, CSS, DHRS, EDG, ESSPWR, HPIS, HVAC, MUPS, NNI, SBODG, SFRCS,SWS
T	T.01	CCWS, SWS
U	U.01	AFWS, CCWS, CFS, CSS, DHRS, ESSPWR, HPIS, HVAC, MSS, MUPS, NI, NNI, RCS, SFAS, SFRCS,SWS

Table 4.1.3.1

Location of Appendix R Safe Shutdown Systems (continued)

Area	Compartment	Safe Shutdown Systems in Compartment
UU	UU.01	CCWS, CREVS, EDG, ESSPWR, HPIS, SFRCS, SWS
	UU.02	EDG, ESSPWR, MUPS
V	V.01	AFWS, CSS, DHRS, ESSPWR, MUPS, NI, NNI, RCS, SFAS, SFRCS
	V.02	AFWS, CCWS, CFS, CSS, DHRS, ESSPWR, HPIS, HVAC, MSS, MUPS, NI, NNI, RCS, SFAS, SFRCS,SWS
X	X.01	AFWS, CACS, CCWS, CREVS, CSS, EDG, ESSPWR, HPIS, HVAC, MUPS, NNI, RCS, SFAS, SWS
Y	Y.01	AFWS, CACS, CCWS, CREVS, CSS, ESSPWR, HPIS, HVAC, MUPS, SWS

Table 4.1.3.2

Fire Compartment Description

All Compartments Contain Safe Shutdown Equipment (SSDE) Unless Noted

Fire Compartment	Room Number	Room Description	Detection Y/N	Suppression Man/Auto
A.01	102	Spent Resin Storage Tank Room	N	Man
	103	Spent Resin Transfer Pump Room	N	Man
	104	Decontamination Area	N	Man
	104A	Monorail Area	N	Man
	106	Radioactive Equipment Storage Room	N	Man
	106A	Sampling Hood Room	N	Man
	107	RC Drain Tank Room	N	Man
	108	RC Drain Tank Pump Room	N	Man
	109	Maintenance Work Area	N	Man
	109A	Passage	N	Man
	111	Concentrate Storage Tank Room	N	Man
A.02	110	Passage	Y	Man
	110A	Passage	Y	Man
	112	Decontamination Area	Y	Man
	116	Misc. Waste Evaporation Room	N	Man
	117	Waste Evaporation Storage Tank/Pump Room	N	Man
	117A	Condensate Tank and Pump Room	Y	Man
	119	Degasifier Room	N	Man
	120	Valve Room	N	Man
	121	Waste Gas Storage Tank Room	N	Man
	122	Valve Access Room	N	Man
A.03	114	Misc. Waste Monitoring Tank Room	N	Man
A.04	115	ECCS Pump Room 1-2	Y	Man
A.05	123	Clean Waste Receiver Tank Room	N	Man
	124	Clean Waste Receiver Tank Room	Y	Auto
	125	Detergent Waste Drain Tank Room	N	Man
	126	Misc. Waste Tank Room	N	Man
A.06	127E	Containment Annulus (East)	Y	Man
A.07	236	No. 2 Mechanical Penetration Room	Y	Auto
A.08	314	No. 4 Mechanical Penetration Room	Y	Auto
A.09	115CC	Cable Chase	Y	Man
	314CC	Cable Chase	Y	Man

Table 4.1.3.2

Fire Compartment Description (continued)

Fire Compartment	Room Number	Room Description	Detection Y/N	Suppression Man/Auto
AB.01	105	ECCS Pump Room 1-1	Y	Man
	113	Decay Heat Cooler Room	Y	Man
	113A	Hatch Area	Y	Man
AB.02	127W	Annulus Space (West)	Y	Man
AB.03	208	No. 1 Mechanical Penetration Room	Y	Auto
	208DC	Duct Chase	N	Man
AB.04	225	Make Up Pump Room	Y	Man
	226A	Vestibule	N	Man
AB.05	303	No. 3 Mechanical Penetration Room	Y	Auto
	303PC	Pipe Chase	N	Man
AB.06	AB-3	Aux. Building Stairwell	N	Man
AC.01	A1	BWST Pipe Trench	N	Man
	A2	PWST Trench	N	Man
B.01	100	Equipment and Pipe Chase	Y	Man
	101	Pipe Tunnel	Y	Man
BD.01	50	Screen Wash Pump Room	N	Man
	54	Stairway	N	Man
BE.01	51	Diesel Fire Pump Room	Y	Man
	55	Diesel Fire Pump Tank Enclosure	N	Man
BF.01	52	Service Water Pump Area	Y	Auto
	52A	Service Water Fan Enclosure	N	Man
BG.01	53	Service Water Valve Room	Y	Man
	53A	Pipe Tunnel-H2O Treatment Building	N	Man
	250	Pipe Tunnel	N	Man
	251	Valve Room	N	Man
BH.01	10	Sample Laboratory	N	Man
	11	Substation	Y	Man
	12	Chemical Storage Room	Y	Man
	12A	Control Room	N	Man
	13	Chlorination Room	Y	Man
	15	Filter Room	Y	Man

Table 4.1.3.2
Fire Compartment Description (continued)

Fire Compartment	Room Number	Room Description	Detection Y/N	Suppression Man/Auto
BM.01	A3	Diesel Oil Pumphouse	Y	Man
BN.01	A4	Emergency Diesel Week Tanks, no SSDE	N	Man
CC.01	411	Corridor	Y	Man
	412	Corridor	Y	Man
	412A	Corridor	Y	Man
	413	Trace Analysis Lab	Y	Man
	414	Health Physics Storage	N	Man
	415	Corridor	Y	Man
	417	Hot Shower Room	Y	Man
	417A	Hot Shower Room	Y	Man
	418	Decontamination Shower	Y	Man
	419	Clean Janitor's Closet	N	Man
	420	Clean Toilet Room	Y	Man
	420A	Shower Area	N	Man
	421	Chemistry Turnover Area	Y	Man
	422	Vestibule	N	Man
	422B	Ladder Space	N	Man
	423	Chemistry Oil Testing Lab	Y	Man
	424	Hot Laboratory	Y	Man
	424A	Chem. Duty Supervisor's Office	N	Man
	424B	Cold Laboratory	Y	Man
	424C	Counting Room	Y	Man
	425	Instrumentation Calibration Room	N	Man
	426	Personnel Lock Area	N	Man
CC.02	411CC1	411 Cable Closet 1	N	Man
	411CC2	411 Cable Closet 2	N	Man
	411CC3	411 Cable Closet 3	N	Man
D.01	214	Core Flooding Tank Area	Y	Man
	215	Let Down Cooler Area	Y	Man
	316	Flooding Tank Area	N	Man
	317A	Emergency Lock Enclosure	N	Man
	407	Hatch Area	N	Man
	700	Passage	N	Man
D.02	216	Steam Generator Area	Y	Man
D.03	217	Core Flood Tank Area	N	Man
	220	Incore Instrument Trench Area	Y	Man
	315	Tank Area	N	Man
	317	Hatch Area	N	Man

Table 4.1.3.2

Fire Compartment Description (continued)

Fire Compartment	Room Number	Room Description	Detection Y/N	Suppression Man/Auto
D.03 (cont.)	410	Passage	Y	Man
	580	Pressurizer Valve Room	Y	Man
	701	Passage	N	Man
D.04	213	Reactor Area	N	Man
	218	Steam Generator Area	Y	Man
	219	Lower Canal Area	N	Man
DD.01	422A	Cable Spreading Room	Y	Auto
DF.01	427	Number 2 Electrical Penetration Room	Y	Auto
DG.01	402	Number 1 Electrical Penetration Room	Y	Auto
DH.01	602	Number 2 Main Steam Line Area	Y	Man
	705	Penthouse	N	Man
DH.02	600	Purge Inlet Equipment Room	Y	Man
	601	Number 1 Main Steam Line Area	Y	Man
	601A	Number 1 Main Steam Line Area	N	Man
	706	Penthouse	N	Man
E.01	237	Aux. Feed Pump 1-1 Room	Y	Man
EE.01	500	Radwaste and Fuel Handling Area	Y	Man
	501	Radwaste Exhaust Fan Room	Y	Auto
	501DC	Duct Chase	N	Man
	515	Purge Exhaust Equipment Room	Y	Man
F.01	238	Aux. Feed Pump 1-2 Room	Y	Man
FF.01	502	Control Cabinet Room	Y	Man
	505	Control Room	Y	Man
	506	Control Room Toilet	Y	Man
	507	Shift Supervisors Office	Y	Man
	509	Control Room Passage	N	Man
	510	Computer Room	Y	Man
	511	Shift Managers Office	Y	Man
	512	SS Admin. Assists Office	Y	Man
	513	Toilet	Y	Man
FF.02	503	Operator Study Room	Y	Man

Table 4.1.3.2
Fire Compartment Description (continued)

Fire Compartment	Room Number	Room Description	Detection Y/N	Suppression Man/Auto
FF.03	504	Control Room Kitchen, no SSDE	Y	Man
G.01	200	Clean Liquid Waste Monitoring Tank Room	N	Man
	201	Clean Liquid Waste Monitoring Tank Room	N	Man
G.02	203	Clean Water Monitor Tank Transfer Pump Room	N	Man
	204	Clean Water Monitor Tank Filter Room	N	Man
	206	Make Up and Purification Filter Room	Y	Man
	207	Mechanical Penetration Room 1 Vestibule	N	Man
	209	Corridor Mechanical Penetration Room !	Y	Auto
	221	Top/Transtube Shield Room	Y	Man
	227	Passage	Y	Auto
G.03	210	SFP Demineralizer Room	N	Man
	211	Valve Room	Y	Man
	212	Valve Room	Y	Man
G.04	228	Demineralizer Room	Y	Man
	230	Demineralizer Filter Room	Y	Man
	231	Clean Waste Booster Pump Room	Y	Man
	232	Valve Room	Y	Man
	233	Demineralizer Room	N	Man
	234	Boric Acid Evaporator Room 1-2	Y	Man
	235	Boric Acid Evaporator Room 1-1	Y	Man
	240	Boric Acid Addition Tank Room	Y	Man
	241	Passage	Y	Man
	242	Valve Room	Y	Man
	243	Waste Gas Compressor Room 1-2	Y	Man
	244	Waste Gas Compressor Room 1-1	Y	Man
HH.01	603	AC Equipment Room	Y	Man
	603A	Records and Storage Area	N	Man
	603B	Vestibule	N	Man
II.01	245	Cooling Water Tank Room	N	Man
	246	Condenser Pit	N	Auto
	247	Heater Drains Valve Room	N	Auto
	248	Condenser Demineralizer Hold Up Tank Room	N	Man
	249	Lube Oil Storage Tank Room	N	Auto
	249DC	Turbine Building Duct Chase	N	Man
	252	Feed Water Pump Room	N	Auto
	253	Condensate Pump Pit	N	Auto
	254	Storage Area	N	Auto
	326	Heater Bay Area	N	Auto

Table 4.1.3.2

Fire Compartment Description (continued)

Fire Compartment	Room Number	Room Description	Detection Y/N	Suppression Man/Auto
II.01 (cont.)	334	Turbine Pedestal Area	N	Auto
	334A	Demineralizer Back Wash Tank Area	N	Auto
	334B	Plant Chemistry Lab	N	Man
	346	Janitor's Closet	N	Auto
	348	Circulating Water Pump House	N	Man
	430	Heater Bay Area	N	Auto
	431	Turbine Area	N	Auto
	431A	Condensate Demineralizer Area	N	Auto
	508	Control Room Vestibule	N	Man
	514	Heater Bay Area	N	Auto
	517	Turbine Operating Floor	N	Auto
	517A	Battery Rack Room	N	Man
	517B	Battery Charger Room	N	Man
	518A	Fire Brigade Locker Room	N	Auto
	518B	Met Lab	N	Auto
	604	Heater Bay Area	N	Auto
	707	Heater Bay Area	N	Auto
II.02	331	Auxiliary Boiler Room, no SSDE	N	Auto
II.03	333	Seal Oil Room, no SSDE	N	Auto
II.04	335	Welding Area, no SSDE	N	Auto
	336	Main Workshop	N	Auto
	336A	Tool Crib	N	Auto
	336B	Supply Storage	N	Auto
	336C	Maintenance Foreman Office	N	Auto
	338	Toilet	N	Auto
	339	Maintenance Office	N	Man
	340	Storekeeper Room	N	Auto
	340A	Electrical Foreman Office	N	Auto
	341	Main Toolroom	N	Auto
II.05	337	Oil Drum Storage Room, no SSDE	N	Auto
II.06	345	Condensate Storage Tank Room, no SSDE	N	Man
II.07	347	Lube Oil Filter Room, no SSDE	N	Auto
II.08	432	Turbine Lube Oil Tank Room, no SSDE	N	Auto
II.09	516	Non Rad Sup ... Eqp. Room	Y	Man

Table 4.1.3.2

Fire Compartment Description (continued)

Fire Compartment	Room Number	Room Description	Detection Y/N	Suppression Man/Auto
J.01	319	Diesel Generator 1-2 Room	Y	Auto
	319A	Diesel Generator 1-2 Room	Y	Man
J.02	320A	Day Tank 1-2 Room	Y	Auto
K.01	318	Diesel Generator 1-1 Room	Y	Auto
	318UL	Diesel Generator 1-1 Room	Y	Man
K.02	321A	Day Tank 1-1 Room	Y	Auto
MA.01	MH3001	Manhole MH3001	N	Man
MB.01	MH3004	Manhole MH3004	N	Man
MC.01	MH3005	Manhole MH3005	N	Man
ME.01	MH3041	Manhole MH3041	N	Man
MF.01	MH3042	Manhole MH3042	N	Man
MG.01	JB30D4	Junction Box JB30D4, no SSDE	N	Man
MH.01	MH3009	Manhole MH3009	N	Man
OS.01	30	Misc. Diesel Room	N	Man
	31	Oil Tank Room	N	Man
	330	Vestibule	N	Man
	703	Passage Elevator Number 2	N	Man
	A5	H2 Trailer Area	N	Man
	A6	Permanent H2 Area	N	Man
	AFE	Aux. Feedwater Exhaust	N	Man
	OS	Outside	N	Man
OS.02	OS	Outside	N	Man
P.01	320	Maintenance	Y	Man
P.02	321	Charge Room	Y	Man
P.03	322	Passage to Diesel Generator Rooms	Y	Man
Q.01	323	High Voltage Switchgear Room B	Y	Man

Table 4.1.3.2

Fire Compartment Description (continued)

Fire Compartment	Room Number	Room Description	Detection Y/N	Suppression Man/Auto
R.01	324	Aux. Shut Down Panel and Transfer SW Room	Y	Man
	324DC	Duct Chase	Y	Man
S.01	325	High Voltage Switchgear Room A	Y	Man
T.01	328	CCW Heat Exchanger and Pump Room	Y	Auto
U.01	310	Passage	Y	Auto
	312	Spent Fuel Pump Room	Y	Man
	313	Hatch Area	Y	Auto
UU.01	327	TB Elevator Machine Room	N	Man
	329	Vestibule	N	Man
	AB1	Aux. Building Stairwell	N	Man
	EL2	Aux. Building Elevator	N	Man
V.01	222	Fuel Transfer Tube Room	N	Man
	223	Cask Pit	Y	Man
	224	Spent Fuel Storage Pool	Y	Man
	300	Fuel Handling Area	Y	Man
	300A	Cask Wash Area	Y	Man
	300B	Drum Storage	Y	Man
	301	Solid Waste Baler Area	Y	Man
	302	Drumming Station	N	Man
	304	Corridor	Y	Auto
	305	Demin. Vessels	Y	Man
	306	New Fuel Storage	Y	Man
	400	Passage	Y	Man
	401	Fuel Handling Exhaust Unit Room	N	Man
	404	Corridor	Y	Man
	405	Storage	Y	Auto
	406	Hot Instrument Shop	Y	Man
X.01	428	Low Voltage Switchgear Room F Bus	Y	Man
	428B	Number 1 Electrical Isolation Room	N	Man
X.02	428A	Battery Room B	Y	Man
Y.01	429	Low Voltage Switchgear Room E Bus	Y	Man
	429A	Number 2 Electrical Isolation Room	N	Man
Y.02	429B	Battery Room A	Y	Man

4.1.4 Screening of Fire Areas

The EPRI FIVE methodology has been used to determine the fire ignition frequencies of the plant equipment supporting safe shutdown at the Davis-Besse Nuclear Power Station. Phase I of the methodology provides a method for screening plant areas whose loss due to fire in that area will have an insignificant impact on the ability to achieve and maintain safe shutdown. The evaluation has been performed for all compartments that contain safe shutdown equipment or cables, or for which a fire in that compartment will result in a plant trip or shutdown. Areas such as offices and maintenance shops have not been considered. An exposure fire is assumed to occur within each fire compartment, and all safe shutdown components within the fire compartment are considered damaged by the fire. In addition, the normal redundant or alternate shutdown path outside the fire compartment is assumed to be unavailable. The fire itself is confined to the fire compartment in question. The Phase I screen takes credit for fire area boundaries as described in the FHAR being effective in preventing the spread of a fire across fire barriers. At Davis-Besse, a fire barrier surveillance program is in place which satisfies the intent of the guidelines in the Sandia Fire Risk Scoping Study Evaluation. This program is described in Section 8 of the FHAR. Also, the fire barriers taken credit for in the Davis-Besse Appendix R Safe Shutdown Analysis are designed and installed in accordance with good fire protection engineering practice and nationally recognized fire protection standards. The safe shutdown systems considered for this analysis are those systems that have been identified as meeting the requirements of Section III.G of Appendix R to 10CFR50. These systems are identified in Section 3 of the Davis-Besse Unit 1 Fire Hazard Analysis Report (FHAR). The FHAR identifies a minimum set of plant systems and components necessary to achieve the functional goals and assure npliance with the requirements of Appendix R. A fire compartment is screened in Phase I if:

1. There are no Appendix R safe shutdown components in the fire compartment, and
2. Following a fire event in the fire compartment, there is no demand for safe-shutdown functions because the plant can maintain normal plant operations.

As a conservative measure in Phase I of this analysis, it is assumed either that there are safe shutdown components involved or the possibility of a plant shutdown exists in all compartments except BN. Compartment BN comprises an area outside of the protected area where the EDG week tanks are located, and contains no exposed safe shutdown components and no need for a plant shutdown should a fire occur in this compartment. Therefore all compartments with the exception of BN were passed on for the more in-depth analysis of the fire compartment interaction analysis and Phase II of the methodology.

4.1.5 Fire Compartment Interactions and Screening (FCIA)

Once the fire areas and compartments of the plant were defined and the location of safe shutdown equipment identified, a fire compartment interaction analysis was completed. The purpose of this analysis was to determine if a fire in one compartment could spread to adjacent compartments, thereby affecting more safe shutdown equipment.

The rules for determining the potential for fire spread are provided by the FIVE methodology. In nmary, these are:

1. There is a boundary that would have no adverse effect on safe shutdown capability.
2. There is a 2-hour or 3-hour rated fire barrier between compartments.
3. There is a boundary consisting of a 1-hour rated fire barrier with a critical combustible loading in the exposing compartment less than 80,000 BTU/.ft^2.
4. There is a boundary where the exposing compartment has a low critical combustible loading of less than 20,000 BTU/ ft^2. and automatic fire detection.
5. There is a boundary where both the exposing and exposed compartments have low critical combustible loadings of less than 20,000 BTU/ ft^2.
6. There is a boundary where automatic fire suppression is installed over combustibles in the exposing compartment.

If two adjacent compartments have a boundary which meets one of the above criteria, it is considered improbable that a fire will spread between the compartments. Note that the fire is considered to originate in the "exposing" compartment with the potential to spread into the "exposed" compartment. It is possible that, based on compartment fire loadings and the location of fire detection and suppression systems, fires may spread in one direction and not in the other.

It was determined from References 5 through 11, that in general, all fire areas are bounded by 3-ır fire rated barriers. An evaluation of the potential for fires to propagate across each fire area boundary ıs included in the FHAR. Each boundary was found acceptable and it was concluded that fires will not spread from area to area. This limits the analysis to interactions between fire compartments within the same fire area. A matrix was made for each multi-compartment fire area which shows each compartment as the initiating (exposing) compartment and evaluates the potential to spread to each of the adjacent compartments. If two compartments are non-adjacent, a "N" was entered in the matrix. If one of the above rules could be used to justify non-propagation, that rule number was entered in the matrix. In such cases a walkdown was performed to confirm the lack of any combustible concentration or continuity at the compartment interface. If propagation could occur, a "PFS" (Potential Fire Spread) was entered in the matrix. The matrix for each multi-compartment fire area is presented in Figure 4.1.5.1.

The analysis revealed that there are only three fire areas where fires can spread between compartments. Specifically, in fire area A, compartment A.09 can spread into A.08 and A.07 because compartment A.09 has a high fire loading (>200,000 BTU/ft^2) and no automatic fire suppression. A.08 cannot spread into A.09, however, because it has automatic fire suppression installed. A.07 cannot spread into A.09 because A.07 has a very low fire loading and also has automatic fire detection and suppression systems installed. Compartment FF.02 can spread into FF.03, but the reverse is not true based on the rules provided by FIVE. Finally, it was found that a fire in any compartment in the containment can spread throughout all compartments in the containment. Fires in the containment are discussed separately in Section 4.4 of this report.

Fire Area A

Exposing → / Exposed ↓	A.01	A.02	A.03	A.04	A.05	A.06	A.07	A.08	A.09
A.01	X	5	N	N	N	N	N	N	N
A.02	5	X	N	5	5	N	N	N	N
A.03	N	N	X	5	5	5	5	N	N
A.04	N	5	5	X	5	5	5	N	2
A.05	N	5	5	5	X	N	5	N	N
A.06	N	N	5	5	N	X	5	6	N
A.07	N	N	5	5	5	5	X	6	PFS
A.08	N	N	N	N	N	4	6	X	PFS
A.09	N	N	N	2	N	N	4	6	X

Figure 4.1.5.1 Fire Compartment Interaction Matrices

Fire Area AB

Exposing → / Exposed ↓	AB. 01	AB. 02	AB. 03	AB. 04	AB. 05	AB. 06
AB. 01	X	5	5	5	N	2
AB. 02	5	X	5	N	5	N
AB. 03	5	5	X	N	4	N
AB. 04	5	N	N	X	N	N
AB. 05	N	5	4	N	X	N
AB. 06	2	N	N	N	N	X

Fire Area CC

Exposing → / Exposed ↓	CC. 01	CC. 02
CC. 01	X	2
CC. 02	2	X

Figure 4.1.5.1 Fire Compartment Interaction Matrices (continued)

Fire Area D

Exposing ⇨ Exposed ⇩	D. 01	D. 02	D. 03	D. 04
D. 01	X	PFS	5	PFS
D. 02	PFS	X	PFS	PFS
D. 03	5	PFS	X	PFS
D. 04	PFS	PFS	PFS	X

Fire Area DH

Exposing ⇨ Exposed ⇩	DH. 01	DH. 02
DH. 01	X	5
DH. 02	5	X

Figure 4.1.5.1 Fire Compartment Interaction Matrices (continued)

Fire Area FF

Exposing → Exposed ↓	FF. 01	FF. 02	FF. 03
FF. 01	X	2	2
FF. 02	2	X	4
FF. 03	2	PFS	X

Fire Area G

Exposing → Exposed ↓	G. 01	G. 02	G. 03	G. 04
G. 01	X	5	N	N
G. 02	5	X	5	5
G. 03	N	5	X	N
G. 04	N	5	N	X

Figure 4.1.5.1 Fire Compartment Interaction Matrices (continued)

Fire Area II

Exposing → / Exposed ↓	II.01	II.02	II.03	II.04	II.05	II.06	II.07	II.08	II.09
II.01	X	2	2	2	2	2	2	2	2
II.02	2	X	N	N	N	N	N	N	N
II.03	2	N	X	N	N	N	N	N	N
II.04	2	N	N	X	N	N	2	2	N
II.05	2	N	N	N	X	N	N	N	N
II.06	2	N	N	N	N	X	N	N	N
II.07	2	N	N	2	N	N	X	2	N
II.08	2	N	N	2	N	N	2	X	N
II.09	2	N	N	N	N	N	N	N	X

Figure 4.1.5.1 Fire Compartment Interaction Matrices (continued)

Fire Area J

Exposing ⇨ / Exposed ⇩	J. 01	J. 02
J. 01	X	2
J. 02	2	X

Fire Area K

Exposing ⇨ / Exposed ⇩	K. 01	K. 02
K. 01	X	2
K. 02	2	X

Figure 4.1.5.1 Fire Compartment Interaction Matrices (continued)

Fire Area OS

Exposing ⇨ / Exposed ⇩	OS. 01	OS. 02
OS. 01	X	6
OS. 02	6	X

Fire Area P

Exposing ⇨ / Exposed ⇩	P. 01	P. 02	P. 03
P. 01	X	2	N
P. 02	2	X	6
P. 03	N	6	X

Figure 4.1.5.1 Fire Compartment Interaction Matrices (continued)

Fire Area UU

Exposing ⇨ / Exposed ⇩	UU. 01	UU. 02
UU. 01	X	5
UU. 02	5	X

Fire Area V

Exposing ⇨ / Exposed ⇩	V. 01	V. 02
V. 01	X	2
V. 02	2	X

Figure 4.1.5.1 Fire Compartment Interaction Matrices (continued)

Fire Area X

Exposing → / Exposed ↓	X.01	X.02
X.01	X	3
X.02	3	X

Fire Area Y

Exposing → / Exposed ↓	Y. 01	Y. 02
Y. 01	X	3
Y. 02	3	X

Figure 4.1.5.1 Fire Compartment Interaction Matrices (continued)

The results of the Fire Compartment Interaction Analysis are used by the FIVE program in the detailed fire analyses of each compartment. Where inter-compartmental fire propagation is possible, the initiating frequencies of the compartments are added and the equipment failed due to the fire included the components in both compartments. This was done in accordance with the FIVE methodology.

4.1.5.1 Fire Compartment Interaction Walkdown Results

In order to verify the results of the Fire Compartment Interaction analysis (FCIA), a walkdown of the compartments was completed. The goal of the walkdown was to ensure that accredited physical barriers exist between compartments or that there are no significant accumulations of combustibles near open compartment boundaries. One of the FCIA rules is that a low fire loading in a compartment will preclude fire propagation. If the bulk of the fire load were concentrated at a compartment boundary, however, fire spread may occur despite the FCIA conclusion.

The results of the walkdown are included in Table 4.1.5.1. Under the FIVE method, it was not always necessary to evaluate the fire spread potential in both directions. This is noted in the table. Only one location (in compartment A.08) was found where fire propagation between compartments might occur despite FCIA results. Further evaluation of this compartment boundary is provided below.

Compartment A.07 has a pipe chase which allows pipes to pass vertically from the floor of A.07 through its ceiling and up into compartment A.08. There are several pipes passing through or across the ›e chase in A.08 that are insulated with a black rubber foam, known as Rubatex. This material will ignite it exposed to a high enough temperatures (FIVE lists a critical temperature of 734°F). This has potential to allow a fire to propagate from A.07 to A.08.

In inspecting the area around the pipe chase, it was concluded that the only mechanism for reaching the temperatures required to ignite the Rubatex would be if the covered piping were directly in the plume of a fire. This conclusion was supported by evaluations of the configuration with the FIVE analysis tools. Radiant energy, and convective heat transfer, as caused by hot gas jets, would not cause the insulation to reach the required temperatures. The inspection also revealed that there are no significant combustibles in or near the pipe chase so that formation of a plume under the pipes of concern is not feasible. Consequently, there is no means available to propagate a fire from area A.07 to A.08.

Additionally, it was found that there are sprinklers surrounding the pipe chase in compartment A.08. These sprinklers would experience nearly the same temperatures as the pipes of concern. Since their actuation temperature is so much lower than the critical temperature of the Rubatex insulation, it is expected that the sprinklers would actuate before the insulation would become a problem. This would further assure that the insulation does not reach its critical temperature.

Table 4.1.5.1 FCIA Walkdown Results

Comp.	Comp.	Direction*	Eval**	Comp.	Comp.	Direction*	Eval**
A.01	A.02	BOTH	1	AB.01	AB.02	BOTH	3
A.02	A.04	BOTH	2	AB.01	AB.03	BOTH	3
A.02	A.05	BOTH	1	AB.01	AB.04	BOTH	1
A.03	A.04	BOTH	1	AB.01	AB.06	BOTH	1
A.03	A.05	BOTH	1	AB.02	AB.03	BOTH	3
A.03	A.06	BOTH	3	AB.02	AB.04	BOTH	1
A.03	A.07	BOTH	1	AB.02	AB.05	BOTH	3
A.04	A.05	BOTH	1	AB.03	AB.04	BOTH	1
A.04	A.06	BOTH	3	AB.03	AB.05	BOTH	3
A.04	A.07	BOTH	3				
A.05	A.07	BOTH	1	FF.01	FF.02	BOTH	1
A.06	A.07	BOTH	3	FF.01	FF.03	BOTH	3
A.06	A.08	BOTH	3	FF.02	FF.03	FF.02 TO FF.03	1
A.07	A.08	BOTH	4				
A.07	A.09	A.07 TO A.09	1	G.01	G.02	BOTH	1
A.08	A.09	A.08 TO A.09	1	G.02	G.03	BOTH	1
				G.02	G.04	BOTH	1
D.01	D.O3	BOTH	5	Y.01	Y.02	BOTH	1
II.01	II.02	BOTH	1	P.01	P.02	BOTH	1
II.01	II.03	BOTH	1	P.01	P.03	BOTH	1
II.01	II.04	BOTH	1	P.02	P.03	BOTH	1
II.01	II.05	BOTH	1				
II.01	II.06	BOTH	1	CC.01	CC.02	BOTH	1
II.01	II.07	BOTH	1	K.01	K.02	BOTH	1

Table 4.1.5.1 FCIA Walkdown Results (continued)

Comp.	Comp.	Direction*	Eval**	Comp.	Comp.	Direction*	Eval**
II.01	II.08	BOTH	1	J.01	J.02	BOTH	1
II.01	II.09	BOTH	1	UU.01	UU.02	BOTH	1
II.04	II.07	BOTH	1	DH.01	DH.02	BOTH	1
II.04	II.08	BOTH	1	V.01	V.02	BOTH	1
II.07	II.08	BOTH	1	OS.01	OS.02	BOTH	1
				X.01	X.02	BOTH	1

* Direction of potential fire spread
** Evaluation Notes

1. The compartments are separated by walls and/or closed doors.
2. The compartments are separated by walls and a door with a fire detection system actuated door.
3. There is no significant concentration of combustibles around the opening between compartments.
4. There are significant combustibles near the opening. Further evaluation is required.
5. This area was not walked down. Containment close-out requirements would prohibit any concentration of combustibles.

Thus, based on this evaluation, it was concluded that a fire would not propagate from compartment A.07 to A.08. Because this was the only compartment boundary of concern from the FCIA walkdown, all ıes from the walkdown were resolved. The results of the FCIA were confirmed and are acceptable when compared to the actual plant.

4.2 Quantitative Analysis and Screening

Each of the compartments that survived the qualitative screening that constitutes Phase I of the FIVE process was subjected to quantitative evaluation in Phase II. The first step in this quantitative evaluation was to compare a bounding estimate of the fire initiated core-damage frequency in each compartment to the FIVE screening criterion of 1×10^{-6}/yr. Compartments whose bounding core-damage frequencies fell below this criterion were judged not to pose a significant potential to constitute a vulnerability to severe accidents. Compartments for which these bounding estimates were above the criterion were retained for progressively more detailed analysis. In these more detailed analyses, some of the conservatism applied in the bounding assessments were removed. The resulting core-damage frequencies were again compared to the FIVE screening criterion; compartments that could not be screened were evaluated in still more detail to determine whether they might present a potential vulnerability.

The first step in these quantitative evaluations was to estimate the frequency of fire initiation in each compartment. This frequency is designated as F_1 in the FIVE methodology. The product of this frequency and the conditional probability that a fire in the compartment could lead to core damage constitutes the estimate of core-damage frequency in each compartment.

4.2.1 Calculation of Fire Initiation Frequencies (F1)

The purpose of Phase II is to identify potential fire vulnerabilities to equipment, components, and cables necessary to assure the capability for safe and stable plant shutdown conditions. Phase II is a multi-step progressive probabilistic evaluation that considers the sequence of events which must occur to create the loss of safe shutdown functions. Step 1 of Phase II of the FIVE methodology requires a counting of ignition sources in each of the compartments identified in Phase I. Fixed or transient ignition sources are individual pieces of plant equipment or a hot work activity (grinding, welding) with the potential to ignite nearby combustibles resulting in damage to safe shutdown related equipment or cables. The count of fixed ignition sources was obtained from the Davis-Besse Configuration Equipment Summary (DBCES). This system is used to support the equipment information requirements for all Davis-Besse Engineering, Maintenance, Maintenance Planning, and Operations. As such, it contains the most complete listing of all plant equipment available on-site. From this data base, a listing of all equipment relevant to the determination of fire ignition frequency was made along with corresponding room and fire area. Consistent with FIVE guidance, the following assumptions were made in the selection of the ignition sources from that list:

1. Pumps with one or less horse power rating were not included. These are minor fire sources which typically are only energized intermittently and contain small amounts of combustible materials. For compartments determined to be potentially significant, walkdowns were used to ensure no such equipment is sufficiently close to critical equipment or cables to cause damage.

2. Cranes, trolley, and hosts are not included. This equipment is only energized intermittently and usually is attended while operating. For compartments determined to be potentially significant, walkdowns were used to ensure no such equipment is sufficiently close to critical equipment or cables to cause damage.

3. The cable construction utilized at Davis-Besse satisfies the requirements of IEEE 383. Although self-ignited cable fires are discounted, the FIVE methodology recognizes a small contribution from self-ignited fire in qualified cable associated with junction boxes. However, sustained combustion from such fires is deemed to be not credible and fires of this type are not considered further. This is consistent with plant experience and with assertions made in previous PRA studies.

4. Non-Qualified cables and splices in the cables were not included. Non-qualified cable is not used in open raceways at Davis-Besse. The enclosure of the cable significantly reduces the likelihood of a fire in a non-qualified cable or splice as well as any fire propagating along the length of the cable.

5. Lighting distribution panels were included as electrical cabinets.

6. Individual low voltage (<120 VAC) breakers were not included. Breakers are generally not stand alone devices as they are usually collectively contained within cabinets. Breakers in 480 V motor control centers and up were counted as individual electrical cabinets.

7. Static voltage regulators and inverters were included as electrical cabinets.

8. Fuse panels were not included as these are essentially passive components which are deemed as non-credible fixed sources of sustained fires.

9. Batteries for emergency lights were not included. These are small, isolated batteries and have not shown indications in previous fire reports (e.g. NSAC-178L) to cause problems as fire ignition sources.
10. Contribution from air dryers were assumed to be included with the frequency for their associated air compressors.

A walkdown of the plant was performed to verify the count of ignition sources. The equipment count, as verified from the walkdowns, was used as the final value. A summary of the fixed ignition sources are contained in Table 4.2.1.1 through Table 4.2.1.6.

The approach taken to evaluate the ignition source frequency is that implemented by revision 1.0 of the FIVE computer code. For each fire compartment, the most appropriate location is selected from the choices provided by the FIVE software. A weighting factor is selected by the software depending upon the selected location, and is utilized to translate generic fire frequencies for a location to specific, single unit fire frequencies.

Generic fire frequencies for typical buildings in the nuclear industry have been compiled by EPRI. A total of 800 events over a period from 1965 to 1988 were identified from 114 BWR and PWR units across the United States representing a total sample size of about 1300 reactor years of operation. This data has been incorporated in the FIVE software. A fire frequency is automatically selected given the type and number of plant equipment and its location. The product is taken of the weighting factor and the fire frequency for each type of equipment and a sum taken over all types of equipment in each fire area.

The contribution to fire frequency due to combustible transients of various types are taken into account by use of the FIVE methodology. The recommended transient fire frequency of 1.3E-03 is adjusted by the ignition source weighting factor. This weighting considers the contribution due to cigarette smoking, extension cords, heaters, candles, overheating, and hot pipes. The use of cigarettes and candles are not permitted within the Davis-Besse Nuclear Power Station; therefore, their contribution is neglected. A contribution due to fires caused by welding and cutting has been included. Cable fires caused by welding were not considered credible given the sole use in all plant areas of cable with fire properties consistent with "qualified" cable, and the existence of strict administrative controls with respect to welding. Plant procedures require that all welding and cutting have a 30 minute fire watch after the cessation of such activities. It is therefore reasonable to expect that the likelihood of such fires is much lower than the 5.1E-03 suggested by the FIVE methodology. The remainder of the factors were conservatively included.

The transient contributions are added to the fire frequencies from fixed sources obtained above to obtain the Compartment Fire Frequency (F1). These values are given in Table 4.2.1.7.

With respect to hydrogen, there are two significant sources of hydrogen at the Davis-Besse Nuclear Power Station. The first is used to fill the main electrical generator. The main generator, during operation, contains hydrogen to facilitate cooling and to reduce windage losses. If necessary, gas can be added to maintain generator casing pressure via supply lines fed from a hydrogen storage facility located in the yard. Both the generator and associated hydrogen supply piping are located in fire compartment II.01

(turbine building). As such, the ignition frequency associated with miscellaneous hydrogen fires was ded to fire compartment II.01.

The second significant source of hydrogen provides gas to the makeup tank for oxygen scavenging in the RCS. This source is supplied from bottles located in the cryogenic storage area located west of the Auxiliary Building. Hydrogen addition to the makeup tank is done in a batch fashion by operators, and the hydrogen supply is normally isolated from system piping in the Auxiliary Building. Any break or leak in the piping would therefore be limited to the hydrogen contained in the small pipe itself. As such, inclusion as a fixed source of the potential for a significant fire from hydrogen addition piping in the Auxiliary Building is considered to be negligible.

There are also small transportable tanks of hydrogen and hydrogen mixtures used inside the Davis-Besse Nuclear Power Station. The tanks containing a hydrogen mixture generally contain two percent hydrogen in an otherwise inert gas. This is below the flammability limit of hydrogen in a normal air environment and is therefore not considered an ignition source. The portable tanks of pure hydrogen are not considered to be permanent installations, and are handled in the plant administratively as a transient combustible. Therefore, for the purposes of this analysis, they are considered to be included in the frequency assigned to transient combustibles.

Also considered, but not explicitly included in the analysis was hydrogen in gaseous waste systems and produced from station batteries. When considering these potential sources, NUREG/CR-5759 (Ref. 12) found that station batteries and gaseous waste systems pose a negligible level of risk when compared to other sources. This conclusion was also found for portable tanks of hydrogen, which supports the above assumption of including these in the transient combustible frequency.

The preceding discussion of station hydrogen sources is consistent with the plant assessment of Generic Letter 93-06 (Ref. 13).

Table 4.2.1.1

Ignition Sources in Auxiliary Building

Area	Electrical Cabinets	Pumps	Transf.	Ventilation Systems	Fire Cab.	Air Comp.	Elevator
A.01	7	4	1	-	-	-	-
A.02	23	6	1	-	-	1	-
A.03	-	2	-	-	-	-	-
A.04	30	5	-	2	-	-	-
A.05	-	6	-	-	-	-	-
A.06	-	-	-	-	-	-	-
A.07	9	-	-	-	-	1	-
A.08	9	4	-	1	4	-	-
A.09	-	-	-	-	-	-	-
AB.01	2	8	-	3	-	-	-
AB.02	-	-	-	-	-	-	-
AB.03	-	-	-	1	-	1	-
AB.04	2	6	1	-	-	1	-
AB.05	1	2	-	1	-	-	-
AB.06	-	-	-	-	-	-	-
AC.01	-	-	-	-	-	-	-
B.01	13	-	-	-	-	-	-
BM.01	-	-	-	-	-	-	-
CC.01	4	-	1	-	1	-	-
CC.02	-	-	-	-	-	-	-
DF.01	92	-	7	-	-	-	-
DG.01	39	-	2	-	-	-	-
DH.01	1	-	1	-	-	-	-
DH.02	4	-	-	1	1	-	-
E.01	1	1	-	1	-	-	-
EE.01	82	1	3	8	-	-	-
F.01	1	1	-	1	-	-	-
G.01	-	-	-	-	-	-	-
G.02	181	1	5	-	2	-	-

Table 4.2.1.1
Ignition Sources in Auxiliary Building (continued)

Area	Electrical Cabinets	Pumps	Transf.	Ventilation Systems	Fire Cab.	Air Comp.	Elevator
G.03	-	-	-	-	-	-	-
G.04	-	16	-	-	-	-	-
HH.01	49	2	1	10	1	-	-
II.01	466	65	32	6	12	2	-
II.02	4	4	-	1	1	-	-
II.03	1	4	-	-	-	-	-
II.04	8	-	2	-	-	1	-
II.05	-	-	-	-	-	-	-
II.06	1	-	-	-	-	-	-
II.07	-	1	-	-	-	-	-
II.08	-	5	-	-	-	-	-
II.09	1	-	-	-	-	-	-
MA.01	-	-	-	-	-	-	-
MB.01	-	-	-	-	-	-	-
MC.01	-	-	-	-	-	-	-
ME.01	-	-	-	-	-	-	-
MF.01	-	-	-	-	-	-	-
MG.01	-	-	-	-	-	-	-
OS.01	-	1	-	-	-	-	-
OS.02	-	1	-	-	-	-	-

Table 4.2.1.1

Ignition Sources in Auxiliary Building (continued)

Area	Electrica Cabinets	Pumps	Transf.	Ventilation Systems	Fire Cab.	Air Comp..	Elevators
P.01	-	-	-	-	-	-	-
P.02	1	-	-	-	3	-	-
P.03	1	-	-	-	-	-	-
U.01	37	2	1	-	2	-	-
UU.01	1	-	-	-	-	-	1
UU.02	-	-	-	-	-	-	-
T.01	-	3	-	2	-	-	-
V.01	24	3	4	-	-	-	-
V.02	79	6	4	-	-	-	-
Total:	693	80	32	32	14	4	1

Table 4.2.1.2

Ignition Sources Associated with the Emergency Diesel Generators

Area	Electrical Cabinets	Transformers	Ventilation Systems	Air Compressors
J.01	27	1	2	1
J.02	-	-	1	-
K.01	28	1	2	2
K.02	-	-	1	-
Total:	55	2	6	3

Table 4.2.1.3

Ignition Sources in the Intake Structure

Area	Electrical Cabinets	Pumps	Transformers	Ventilation Systems	Fire Cabinets	Battery
BD.01	11	3	1	2	2	-
BE.01	11	3	1	-	1	1
BF.01	51	5	-	-	-	-
BG.01	1	4	-	4	-	-
BH.01	89	15	6	7	1	-
Total:	163	30	8	13	4	1

Table 4.2.1.4

Ignition Sources in the Switchgear & Battery Rooms

Area	Electrical Cabinets	Transformers	Ventilation Systems	Motor Generator	Battery
Q.01	40	-	-	-	-
R.01	8	-	-	-	-
S.01	42	1	-	-	-
X.01	142	10	2	1	2
Y.01	129	7	2	-	2
Total:	361	18	4	1	4

Table 4.2.1.5

Ignition Sources in the Turbine Building

Area	Electrical Cabinets	Pumps	Transformers	Ventilation Systems	Fire Cabinets	Air Compressors	Boilers
II.01	466	65	32	6	12	2	-
II.02	4	4	-	1	1	-	1
II.03	1	4				-	-
II.04	8	2	2	-	-	1	-
II.05	-	-	-	-	-	-	-
II.06	1						
II.07	-	1					
II.08	-	5					
II.09	1	-	-	-			
Total:	481	81	34	7	13	3	1

Table 4.2.1.6

Total Fixed Ignition Sources

Area	Electrical Cabinets	Pumps	Trans.	Ventilation Systems	Fire Cab	Air Comp.	Elevators	Batteries	Boilers	Motor Gen.
Auxiliary Building	693	80	32	32	14	4	1	-	-	-
Intake Structure	163	30	8	13	4	-	-	1	-	-
Turbine Building	481	81	34	7	13	3	-	-	1	-
Emergency Diesel Generator	55	-	2	6	-	3	-	-	-	-
Switchgear/ Battery Room	361	-	18	4	-	-	-	4	-	1
Total:	1,753	191	94	62	31	10	1	5	1	1

Table 4.2.1.7

Fire Frequency F1

Area	Ignition Source	Location Weighting Factor	Number of Sources	Number of Sources in Location	F1
A.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	7	693	1.92E-04
	Pumps	1.0	4	80	9.50E-04
	Transformers(Indoor)	1.0	1	94	8.40E-05
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.75E-04
					1.76E-03
A.02		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	23	693	6.31E-04
	Pumps	1.0	6	80	1.43E-03
	Transformers(Indoor)	1.0	1	94	8.40E-05
	Transients	1.0	10	83	1.56E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Air Compressors	1.0	1	10	4.70E-04
					3.14E-03
A.03		Auxiliary Bldg.(PWR)			
	Pumps	1.0	2	80	4.75E-04
	Transients	1.0	10	83	1.56E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					1.01E-03
A.04		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	30	693	8.23E-04
	Pumps	1.0	5	80	1.19E-03
	Ventilation Subsystems	1.0	2	62	3.07E-04
	Transients	1.0	10	83	1.56E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					2.85E-03
A.05		Auxiliary Bldg.(PWR)			
	Pumps	1.0	6	80	1.43E-03
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.75E-04
					1.96E-03
A.06		Auxiliary Bldg.(PWR)			
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Transients	1.0	10	83	1.57E-04
					5.30E-04

Table 4.2.1.7

Fire Frequency F1 (continued)

Area	Ignition Source	Location Weighting Factor	Number of Sources	Number of Sources in Location	F1
A.07		Auxiliary Bldg.(PWR)			
	Transients	1.0	10	83	1.69E-04
	Welding/Cutting Fires	1.0	1	83	4.03E-04
	Electrical Cabinets	1.0	9	693	2.47E-04
	Air Compressors	1.0	1	10	4.70E-04
					1.25E-03
A.08		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	9	693	2.47E-04
	Pumps	1.0	4	80	9.50E-04
	Ventilation Subsystems	1.0	1	62	1.53E-04
	Fire Protection Panels	1.0	4	31	3.10E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	4.03E-04
					2.19E-03
AB.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	2	693	5.48E-05
	Pumps	1.0	8	80	1.90E-03
	Ventilation Subsystems	1.0	3	62	4.60E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Transients	1.0	10	83	1.57E-04
					2.95E-03
AB.02		Auxiliary Bldg.(PWR)			
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Transients	1.0	10	83	1.57E-04
					5.30E-04
AB.03		Auxiliary Bldg.(PWR)			
	Ventilation Subsystems	1.0	1	62	1.53E-04
	Air Compressors	1.0	1	10	4.70E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	4.03E-04
					1.15E-03

Table 4.2.1.7

Fire Frequency F1 (continued)

Area	Ignition Source	Location Weighting Factor	Number of Sources	Number of Sources in Location	F1
AB.04		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	2	693	5.48E-05
	Pumps	1.0	6	80	1.43E-03
	Transformers(Indoor)	1.0	1	94	8.40E-05
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Air Compressors	1.0	1	10	4.70E-04
					2.56E-03
AB.05		Auxiliary Bldg.(PWR)			
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Electrical Cabinets	1.0	1	693	2.74E-05
	Pumps	1.0	2	80	4.75E-04
	Ventilation Subsystems	1.0	1	62	1.53E-04
					1.19E-03
AB.06		Auxiliary Bldg.(PWR)			
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					5.30E-04
AC.01		Transformer Yard			
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Other Hydrogen Fires	1.0	1	83	4.16E-05
					4.74E-04
B.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	13	693	3.56E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					8.86E-04

Table 4.2.1.7

Fire Frequency F1 (continued)

Area	Ignition Source	Location Weighting Factor	Number of Sources	Number of Sources in Location	F1
BD.01		Intake Structure			
	Electrical Cabinets	1.0	1	1	2.40E-03
	Fire Protection Panels	1.0	2	31	1.55E-04
	Transformers(Indoor)	1.0	1	94	8.40E-05
	Others	1.0	1	1	3.20E-03
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Ventilation Subsystems	1.0	2	62	3.07E-04
					6.68E-03
BE.01		Intake Structure			
	Electrical Cabinets	1.0	1	1	2.40E-03
	Fire Protection Panels	1.0	1	31	7.74E-05
	Transformers(Indoor)	1.0	1	94	8.40E-05
	Others	1.0	1	1	3.20E-03
	Fire Pumps	1.0	1	1	4.00E-03
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					1.03E-02
BF.01		Intake Structure			
	Electrical Cabinets	1.0	1	1	2.40E-03
	Others	1.0	1	1	3.20E-03
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					6.13E-03
BG.01		Intake Structure			
	Others	1.0	1	1	3.20E-03
	Ventilation Subsystems	1.0	4	62	6.13E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Electrical Cabinets	1.0	1	1	2.40E-03
					6.74E-03
BH.01		Intake Structure			
	Electrical Cabinets	1.0	1	1	2.40E-03
	Fire Pumps	1.0	1	1	4.00E-03
	Transformers(Indoor)	1.0	6	94	5.04E-04
	Fire Protection Panels	1.0	1	31	7.74E-05

Table 4.2.1.7

Fire Frequency F1 (continued)

Area	Ignition Source	Location Weighting Factor	Number of Sources	Number of Sources in Location	F1
BH.01	Others	1.0	1	1	3.20E-03
(cont.)	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Ventilation Subsystems	1.0	7	62	1.07E-03
					1.18E-02
BM.01		Transformer Yard			
	Transients	1.0	10	83	1.57E-04
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					5.92E-04
CC.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	4	693	1.10E-04
	Transformers(Indoor)	1.0	1	94	8.40E-05
	Fire Protection Panels	1.0	1	31	7.74E-05
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					8.01E-04
CC.01		Auxiliary Bldg.(PWR)			
	Transients	1.0	4	83	6.75E-05
	Welding Cable Fires	1.0	1	83	6.15E-05
					1.24E-04
DD.01		Cable Spreading Rm.			
	Electrical Cabinets	1.0	1	1	3.20E-03
	Transformers(Indoor)	1.0	1	111	7.12E-05
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					3.65E-03
DF.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	92	693	2.52E-03
	Transformers(Indoor)	1.0	7	94	5.88E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					3.64E-03

Table 4.2.1.7

Fire Frequency F1 (continued)

Area	Ignition Source	Location Weighting Factor	Number of Sources	Number of Sources in Location	F1
DG.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	39	693	1.07E-03
	Transformers(Indoor)	1.0	2	94	1.68E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					1.77E-03
DH.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	1	693	2.74E-05
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Transformers(Indoor)	1.0	1	94	8.40E-05
					6.42E-04
DH.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	4	693	1.10E-04
	Ventilation Subsystems	1.0	1	62	1.53E-04
	Fire Protection Panels	1.0	1	31	7.74E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Transients	1.0	10	83	1.57E-04
					8.70E-04
E.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	1	693	2.74E-05
	Pumps	1.0	1	80	2.38E-04
	Ventilation Subsystems	1.0	1	59	1.61E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Ventilation Subsystems	1.0	1	62	1.53E-04
	Transients	1.0	10	83	1.57E-04
					1.11E-03
EE.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	82	693	2.25E-03
	Pumps	1.0	1	80	2.38E-04
	Transformers(Indoor)	1.0	3	94	2.52E-04
	Ventilation Subsystems	1.0	8	62	1.23E-03
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					4.49E-03

Table 4.2.1.7

Fire Frequency F1 (continued)

Area	Ignition Source	Location Weighting Factor	Number of Sources	Number of Sources in Location	F1
F.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	1	693	2.74E-05
	Pumps	1.0	2	80	4.75E-04
	Ventilation Subsystems	1.0	1	62	1.53E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Transients	1.0	10	83	1.57E-04
					9.48E-04
G.01		Auxiliary Bldg.(PWR)			
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					5.30E-04
G.02		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	181	693	4.96E-03
	Pumps	1.0	1	80	2.38E-04
	Transformers(Indoor)	1.0	5	94	4.20E-04
	Fire Protection Panels	1.0	2	31	1.55E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					6.31E-03
G.04		Auxiliary Bldg.(PWR)			
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					5.30E-04
G.04		Auxiliary Bldg.(PWR)			
	Pumps	1.0	16	80	3.80E-03
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					4.33E-03
HH.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	49	693	1.34E-03
	Pumps	1.0	2	80	4.75E-04
	Transformers(Indoor)	1.0	1	94	8.40E-05
	Ventilation Subsystems	1.0	10	62	1.53E-03
	Fire Protection Panels	1.0	1	31	7.74E-05
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					4.04E-03

Table 4.2.1.7

Fire Frequency F1 (continued)

Area	Ignition Source	Location Weighting Factor	Number of Sources	Number of Sources in Location	F1
II.01		Turbine Bldg.			
	Electrical Cabinets	1.0	466	481	1.26E-02
	Other Pumps	1.0	65	81	5.06E-03
	Fire Protection Panels	1.0	12	31	9.29E-04
	Transformers(Indoor)	1.0	32	94	2.69E-03
	Air Compressors	1.0	2	10	9.40E-04
	Ventilation Subsystems	1.0	6	62	9.19E-04
	Transients	1.0	10	83	1.57E-04
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	T/G Excitor	1.0	584	620	3.77E-03
	T/G Hydrogen	1.0	584	620	5.18E-03
	T/G Oil	1.0	584	620	1.23E-02
					4.49E-02
II.02		Turbine Bldg.			
	Electrical Cabinets	1.0	4	481	1.08E-04
	Other Pumps	1.0	4	81	3.11E-04
	Fire Protection Panels	1.0	1	31	7.74E-05
	Ventilation Subsystems	1.0	1	62	1.53E-04
	Transients	1.0	10	83	1.57E-04
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Boiler	1.0	1	1	3.74E-04
					1.60E-03
II.03		Turbine Bldg.			
	Electrical Cabinets	1.0	1	481	2.70E-05
	Other Pumps	1.0	4	81	3.11E-04
	Transients	1.0	10	83	1.57E-04
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					9.30E-04
II.05		Turbine Bldg.			
	Transients	1.0	10	83	1.57E-04
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					5.92E-04
II.06		Turbine Bldg.			
	Transients	1.0	10	83	1.57E-04
	Welding Cable Fires	1.0	1	83	6.15E-05

Table 4.2.1.7

Fire Frequency F1 (continued)

Area	Ignition Source	Location Weighting Factor	Number of Sources	Number of Sources in Location	F1
II.06		Turbine Bldg.			
(cont.)	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Electrical Cabinets	1.0	1	481	2.70E-05
					6.19E-04
II.07		Turbine Bldg.			
	Other Pumps	1.0	1	81	7.78E-05
	Transients	1.0	10	83	1.57E-04
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					6.69E-04
II.08		Turbine Bldg.			
	Other Pumps	1.0	5	81	3.89E-04
	Transients	1.0	10	83	1.57E-04
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					9.80E-04
II.09		Turbine Bldg.			
	Transients	1.0	10	83	1.57E-04
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Electrical Cabinets	1.0	1	481	2.70E-05
					6.19E-04
J.01		Diesel Generator Rm.			
	Diesel Generators	0.5	1	1	2.60E-02
	Electrical Cabinets	0.5	1	1	2.40E-03
	Transformers(Indoor)	1.0	1	94	8.40E-05
	Air Compressors	1.0	1	10	4.70E-04
	Ventilation Subsystems	1.0	2	62	3.07E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					2.98E-02
J.02		Diesel Generator Rm			
	Transients	1	10	83	1.57E-04
	Welding/Cutting Fires	1	1	83	3.74E-04
	Ventilation Subsystems	1	1	62	1.53E-04
					6.83E-04

Table 4.2.1.7

Fire Frequency F1 (continued)

Area	Ignition Source	Location Weighting Factor	Number of Sources	Number of Sources in Location	F1
K.01		Diesel Generator Rm			
	Diesel Generators	0.5	1	1	2.60E-02
	Electrical Cabinets	0.5	1	1	2.40E-03
	Transformers(Indoor)	1.0	1	94	8.40E-05
	Air Compressors	1.0	2	10	9.40E-04
	Ventilation Subsystems	1.0	2	62	3.07E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					3.03E-02
K.02		Diesel Generator Rm			
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Ventilation Subsystems	1.0	1	94	1.01E-04
					6.31E-04
MA.01		Transformer Yard			
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					4.35E-04
MB.01		Transformer Yard			
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					4.35E-04
MC.01		Transformer Yard			
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					4.35E-04
ME.01		Transformer Yard			
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					4.35E-04
MF.01		Transformer Yard			
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					4.35E-04

Table 4.2.1.7

Fire Frequency F1 (continued)

Area	Ignition Source	Location Weighting Factor	Number of Sources	Number of Sources in Location	F1
MG.01		Transformer Yard			
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					4.35E-04
OS.01		Transformer Yard			
	Yard Trans.(LOSP)	1.0	1	1	1.60E-03
	Transients	1.0	10	83	1.57E-04
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Hydrogen Tanks	1.0	2	89	7.19E-05
					2.26E-03
0S.02		Transformer Yard			
	Yard Trans	1.0	1	1	4.00E-03
	Transients	1.0	10	83	1.57E-04
	Welding Cable Fires	1.0	1	83	6.15E-05
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					4.59E-03
P.01			Auxiliary Bldg.(PWR)		
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					5.30E-04
P.02		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	1	693	2.74E-05
	Fire Protection Panels	1.0	3	31	2.32E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					7.90E-04
P.03		Auxiliary Bldg.(PWR)			
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Electrical Cabinets	1.0	1	693	2.74E-05
					5.58E-04
Q.01		Switchgear Room			
	Electrical Cabinets	0.2	1	1	3.00E-03
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					3.53E-03

Table 4.2.1.7

Fire Frequency F1 (continued)

Area	Ignition Source	Location Weighting Factor	Number of Sources	Number of Sources in Location	F1
R.01		Switchgear Room			
	Electrical Cabinets	0.2	1	1	3.00E-03
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					3.53E-03
S.01		Switchgear Room			
	Transformers(Indoor)	1.0	1	94	8.40E-05
	Electrical Cabinets	0.2	1	1	3.00E-03
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					3.61E-03
T.01		Auxiliary Bldg.(PWR)			
	Pumps	1.0	3	80	7.13E-04
	Ventilation Subsystems	1.0	2	62	3.07E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Transients	1.0	10	83	1.57E-04
					1.55E-03
U.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	37	693	1.01E-03
	Pumps	1.0	2	80	4.75E-04
	Transformers(Indoor)	1.0	1	94	8.40E-05
	Fire Protection Panels	1.0	2	31	1.55E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					2.20E-03
UU.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	1	693	2.74E-05
	Elevator Motors	1.0	1	2	3.15E-03
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					3.71E-03
UU.02		Auxiliary Bldg.(PWR)			
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					5.30E-04

Table 4.2.1.7

Fire Frequency F1 (continued)

Area	Ignition Source	Location Weighting Factor	Number of Sources	Number of Sources in Location	F1
V.01		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	24	693	6.58E-04
	Pumps	1.0	3	80	7.13E-04
	Transformers(Indoor)	1.0	4	94	3.36E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					2.24E-03
V.02		Auxiliary Bldg.(PWR)			
	Electrical Cabinets	1.0	79	693	2.17E-03
	Pumps	1.0	6	80	1.43E-03
	Transformers(Indoor)	1.0	4	94	3.36E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					4.46E-03
X.01		Battery Room			
	RPS MG Sets	1.0	1	1	5.50E-03
	Ventilation Subsystems	1.0	2	62	3.07E-04
	Transformers(Indoor)	1.0	7	94	5.88E-04
	Electrical Cabinets	0.5	1	1	3.00E-03
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					9.93E-03
X.02	Batteries	0.5	1	1	1.60E-03
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Transients	1.0	10	83	1.57E-04
					2.13E-03
Y.01		Battery Room			
	Electrical Cabinets	0.5	1	1	3.00E-03
	Ventilation Subsystems	1.0	2	62	3.07E-04
	Transformers(Indoor)	1.0	7	94	5.88E-04
	Transients	1.0	10	83	1.57E-04
	Welding/Cutting Fires	1.0	1	83	3.74E-04
					4.43E-03
Y.02	Batteries	0.5	1	1	1.60E-03
	Welding/Cutting Fires	1.0	1	83	3.74E-04
	Transients	1.0	10	83	1.57E-04
					2.13E-03

4.2.2 Initial Quantitative Screening

In the first, bounding assessment of core-damage frequency for each of the compartments, all of the equipment and equipment with cables in a particular compartment was assumed to fail as a consequence of a fire in the compartment. The integrated event-tree/fault-tree model developed for the IPE was then applied to obtain an estimate of the conditional probability of core damage, given these failures. This conditional probability is the value for "P_2" from the FIVE methodology.

4.2.2.1 Identification of Fire Effects

For each fire compartment, information was compiled from various data sources to identify the equipment and cables present in the compartment. These sources included the FHAR, the Davis-Besse Configuration Equipment Summary (DBCES, Ref. 14) database, and the computerized cable routing database SETROUTE (Ref. 15). A list of the equipment and cables in each compartment was assembled and then cross-checked among the various sources. Each of the entries in the list was then examined to determine the effect of failure of the associated piece of equipment or cable relative to the potential for core damage as modeled in the IPE. The corresponding basic event from the integrated core-damage model was then identified.

In matching up basic events to damaged equipment, several assumptions were made. These included:

1. where applicable, hot shorts initiate equipment movement to an undesired state (in cases where both states were modeled due to being undesirable for different sequences, both failure states were included in the basic event flag file for the compartment)
2. a loss of function was assumed if only motive power cable could be potentially affected
3. where both power and control cables could be potentially affected, the component was assumed to move to its undesired state prior to loss of function
4. relay cabinet faults were assumed to propagate to the closest applicable cabinet included in the IPE models
5. where components were not modeled in the fault tree, no basic event was assigned since the IPE modellers deemed the component incapable of preventing success
6. where the effects of a component failure are included in a module of the fault tree, that module's basic event and probability were assigned

Because the various wires associated with control of a component frequently follow very different paths through the plant it was possible, by detailed evaluation of the circuit schematic, to identify which cables would not affect the circuits function due to either hot shorts or open circuiting. This information was used in the cable to basic event correlation database.

Circuit analysis was also used in modeling hot shorts. If a component could fail to its undesired .te due to one wire shorting to ground, a bounding value for hot short probability of 0.20 was assigned (indicating that 20% of cable failures could lead to this condition, based on NUREG/CR-2258 (Ref. 16)). If a circuit required two wires to short together or to short to ground together for failure to the undesired state to occur, a probability of 0.04 (or 0.20^2) was assigned. Higher order combinations were deemed highly unlikely and were assigned failure probabilities of 0.00.

In addition to verifying the completeness of the FHAR with respect to the locations of equipment and cables of potential importance to the availability of core cooling, the use of multiple sources allowed for the consideration of systems not explicitly treated by the FHAR. These included the offsite power circuits, the power supply from the station blackout diesel generator (SBODG), and (where verified) the main feedwater (MFW) system.

In developing the FHAR, no credit was given to the use of MFW to provide for decay heat removal. Therefore, the FHAR does not explicitly identify cables and equipment associated with the MFW system. For the IPEEE assessment, the MFW system was also generally assumed to be unavailable. This assumption was made because power and control cables for the system pass through many compartments of the plant, and it was judged not to be an efficient use of resources to attempt to track all of these cables. In a small number of cases, however, a careful review was made to determine whether a fire could affect the MFW system. In these cases, if it was determined that the MFW system was not affected by the fire, the system was credited as a potential means for core cooling.

In developing the FHAR, it was also assumed that offsite power was lost in every case. For the IPEEE assessment, a review was made of the offsite power circuits to determine the compartments in which related power and control cables were located. For most compartments, offsite power was found to be nominally available; consideration of the potential for it to fail due to random events following the plant trip was retained, as was the case for the IPE. Areas which were found to have the potential for damage to power and control cable, resulting in a loss of offsite power were:

Room	Compartment
301	V.01
400	V.01
401	V.01
422A	DD.01
505	FF.01
MH3005	MC.01

For cases in which the fire was presumed to affect offsite power, the conditional probability of core damage was calculated twice. In the first calculation, offsite power was assumed to be lost, just as other equipment failures were taken to be certain. In the second, offsite power was allowed to fail at its nominal unavailability. This was done because the loss of offsite power has both positive and negative effects on the potential for core damage. The loss of offsite power results in reliance on the diesel generators to support ›g-term heat removal via the steam generators, and for successful makeup/high pressure injection (HPI) cooling (referred to at some plants as feed-and-bleed cooling) if all feedwater is lost. If offsite power is lost, however, the potential for a loss-of-coolant accident (LOCA) due to failure of the seals for the reactor plant pumps (RCPs) induced by a loss of seal cooling is significantly diminished. This is because tripping of the RCPs (which would be assured by the loss of offsite power) would preclude the development of substantial seal leakage following the loss of seal cooling. Thus, it would not necessarily have been conservative to assume that offsite power was lost with certainty. After the two sets of calculations had been completed, the higher of the two values for P_2 was selected and used in determining whether the compartment could be screened.

The SBODG was installed after the FHAR was completed, and is therefore not reflected in any FHAR analyses. The SBODG is located in a separate, detached building. Because there are relatively few power and control cables associated directly with the SBODG that pass through the other plant buildings, tracking these cables was a manageable task. It was possible to determine those compartments in which the availability of power from the SBODG could be affected by a fire, and to credit the nominal use of the generator for other compartments.

4.2.2.2 Modeling of Fire-Induced Failures

To estimate the frequency of core damage for the IPE, an integrated set of event trees (which delineate accident sequences) and fault trees (in which system and component failures are developed in detail) was constructed. The core-damage frequency was estimated by applying the fault-tree linking approach to obtain cut sets corresponding to core-damage sequences. For the fire analysis, it was necessary to account for the potential failures of components identified in the previous step by modifying the existing integrated core-damage model.

It would, in theory, have been possible to estimate the value for P_2 for each compartment using the existing IPE cut sets. In practice, however, it proved to be necessary to generate new cut sets to support the fire assessment. First, the equipment assumed to fail as a consequence of the fire would have an effective unavailability of 1.0. Depending on the compartment, the number of basic events corresponding to failed components ranged from one to more than 20. Because it was necessary to apply probabilistic truncation in generating the original IPE cut sets, they would not necessarily have been adequately complete when considering the combinations of failed equipment implied for a particular compartment.

In addition to the need to consider the possibility that some cut sets relevant for a fire in a particular compartment would be missing from those generated for the IPE due to truncation, it was necessary to account for some types of failures that were not included in the IPE models. These included:

- In a few cases, the fire-induced failures affected equipment that was not modeled in the IPE. This was generally the case for components that were very reliable compared to other components whose failures would have a similar impact on plant response.
- In some cases, the fire could cause equipment to actuate spuriously due to hot shorts, and these spurious actuations may have been neglected in the IPE.
- For many compartments, during fire conditions the operators are instructed by procedure to secure equipment to avoid the potential that spurious operation could cause more serious problems than would the unavailability of the equipment itself. These operator actions also needed to be taken into account.

Thus, the approach taken in calculating the value for P_2 was to generate new core-damage cut sets each of the fire compartments. This was done by using the transient event tree from the IPE. This event tree, which is shown in Figure 4.2.2.1, provided the development for all core-damage sequences that were initiated by events other than LOCAs. It was assumed that a plant trip would result from a fire in each of the compartments examined, either due to automatic action or by operator action in responding to the fire. The core-damage sequences from the transient event tree for which cut sets were obtained included the following (the initiating event designation, T, and the subscript "T" have been omitted for simplification):

- Sequence BP, a total loss of feedwater to the steam generators and a failure of pressurizer relief valves to operate to protect the reactor coolant system (RCS) from overpressurization.
- Sequence BQX, a total loss of feedwater and failure to maintain RCS integrity (e.g., due to a stuck-open pressurizer relief valve). Following successful makeup/HPI cooling, long term cooling via recirculation from the containment emergency sump fails to be established.
- Sequence BU, a total loss of feedwater and failure of makeup/HPI cooling.
- Sequence BLX, a total loss of feedwater but successful makeup/HPI cooling, with failure to recover feedwater in the long term and failure to establish long-term recirculation from the containment emergency sump.
- Sequence BWX, a total loss of feedwater but successful makeup/HPI cooling. Feedwater is eventually restored, but the pressurizer relief valves used in support of makeup/HPI cooling fail to reclose, and recirculation from the containment emergency sump fails to be established.
- Sequence QU, which entails a small LOCA due to failure of cooling for the RCP seals and failure of HPI.
- Sequence QX, which also involves a RCP seal LOCA, with successful HPI but failure of long-term cooling via recirculation from the containment sump.

It may be noted that the event tree in Figure 4.2.2.1 includes one additional core-damage sequence, BQU, which was not quantified separately. This sequence was subsumed into sequence BU; it was evaluated separately for the IPE only to allow the outcomes to be differentiated in quantifying the containment event tree for different plant-damage states.

In addition, the event tree indicates that there is further development for sequences involving failure to scram (the end state designated as TK). These sequences were not evaluated explicitly for the fire analysis, primarily because fire-induced failure modes that could cause the reactor protection system (RPS) to fail to function were judged to be very unlikely. The RPS is designed such that deenergization of trip circuits causes the control rods to be released; although fire effects could conceivably cause hot shorts that would result in temporarily preventing a trip, the circuits could not reasonably be expected to remain energized long enough to result in damage to the core. The unavailability of the trip system due to causes other than fire was assessed for the IPE to be sufficiently small that it could be neglected for the fire ılysis.

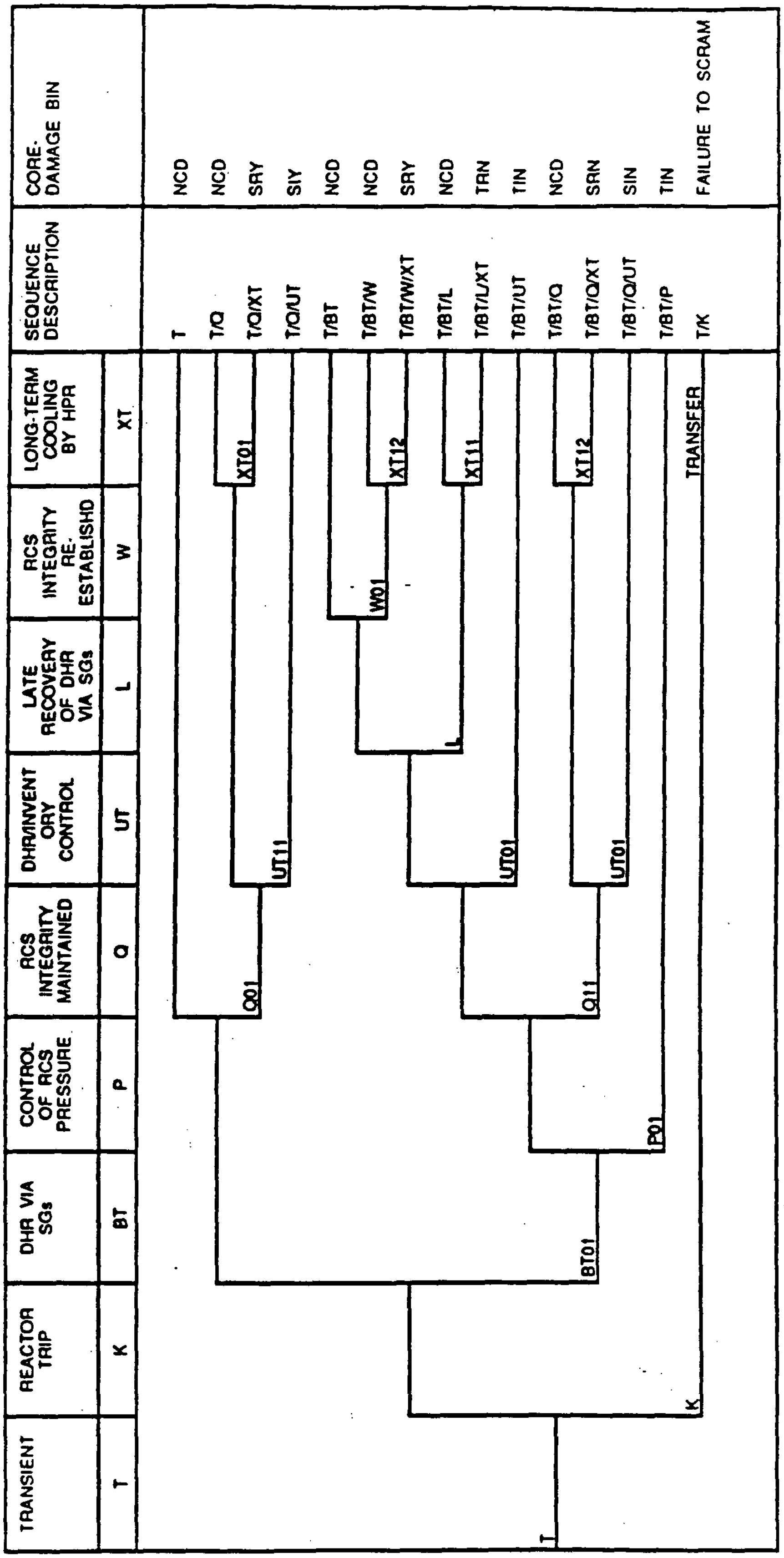


Figure 4.2.2.1 Transient Event Tree

As noted above, some changes to the integrated core-damage model were necessary to ̣ommodate the impact of fires. For some compartments, it was determined that hot shorts could cause undesired actuations of components. For example, a hot short could cause a normally-open motor-operated valve to close spuriously. In most of these cases the fire would also, if it persisted long enough, result in the loss of motive power for the valve. A blanket assumption was made in modeling such failures that the first effect of the fire would be for the spurious actuation to occur, with power then lost. Obviously, if power were lost first, the effect of the hot short could be neglected. Thus, assuming that the valve would be positioned to its undesired state and then depowered by the fire (so that it would remain in that state) was a simplifying, though potentially conservative, step. To account for the effects of hot shorts when they were identified, new basic events were introduced into the integrated core-damage model.

New events were also added to the model to reflect operator actions called for in the abnormal procedure for plant fires (Ref. 17). The most important of these called for tripping one of the turbine-driven AFW pumps to ensure that control faults could not lead to overfeeding its associated steam generator, which could in turn lead to the loss of both turbine-drive pumps due to water carryover into the steam supply lines to the pump turbines. The procedure places the burden on the operating crew to evaluate the plant conditions and to determine which of the actions called for in the procedure (such as tripping an AFW pump) should actually be implemented, depending on the damage caused by the fire. Operators were interviewed to gain further insight into the actual use of this procedure, and it was determined that the operators would expect to maintain the function of the AFW pump called out in the procedure if the other pump was unavailable. Therefore, a basic event reflecting operator action to trip a pump was added the fault tree development for each pump. In cases when the opposite pump started but later failed to ̣un, it was assumed that the operators would have already tripped the pump called for in the procedure. If the opposite pump failed to start, however, the potential that the operators would diagnose the situation and decide to allow the pump called out in the procedure to remain in operation was evaluated probabilistically. If there were other fire-induced faults that could affect the availability of the pump directly or indirectly (such as by causing a control problem), these were accounted for as appropriate failures of the pump. Other human interactions inferred by the fire procedure were handled similarly on a case-by-case basis.

In addition to events which could result in a plant transient, the potential for a fire to induce a LOCA was also assessed. All high/low pressure interfaces from the reactor coolant system to other plant systems were reviewed to establish if there were any credible scenarios for which a circuit fault or hot short could be postulated which would result in the opening of the interface. This review indicated that the only credible path was the potential for an induced opening of the pressurizer pilot operated relief valve (PORV). All other paths were found to be not credible due to various programmatic controls which are designed to prevent such occurrences during power operations. As an example, the drop line from the reactor coolant system to the decay heat system has administrative controls which ensure that the two in-series motor operated isolation valves are closed and depowered when the plant is at power. Areas outside of containment for which the induced PORV opening is applicable are DD.01 (Cable Spread Room), DF.01 (Electrical Penetration Room 2), FF.01 (Control Room), and X.01 (Low Voltage Switch Gear Room 2).

The potential for a fire to induce a stuck-open PORV was accounted for in the context of sequences J and QX, which reflect transient-induced LOCAs with failure of short-term and long-term cooling, respectively. A new basic event reflecting the possibility for a hot short that could cause the PORV to open was introduced into the integrated core-damage model. If this hot short occurred, it was assumed that the PORV would not reclose. This is a potentially conservative assumption, since the control circuit for the PORV is designed such that the valve will close when it is de-energized. Since the hot short would not be expected to persist indefinitely, the PORV should eventually close (unless it mechanically sticks open).

It should be noted that the PORV was generally assumed to be unavailable (e.g., to support makeup/HPI cooling) for the areas in which these hot shorts were a possibility. The cut sets in which the PORV was assumed to be unavailable at the same time that it was assumed to be stuck open were reviewed individually, and where appropriate the complement of the event reflecting the hot short was applied to provide more realistic treatment.

4.2.2.3 Estimation of Conditional Core-Damage Probabilities (P2)

Once the components potentially affected by a fire in a particular compartment and their corresponding basic events were identified, the integrated model was requantified to obtain the conditional probability of core damage (the value for P_2). This was done by generating new core-damage sequences for each fire compartment. Each of the basic events corresponding to components that were present in the fire compartment was set to "true" (i.e., certain to be failed) before the cut sets were generated. For components potentially subject to failure by hot shorts, a probability of 1.0 was used for the corresponding basic event in the process of generating the cut sets.* This was done because the actual probability would depend on the specific number of wire combinations in the circuit and compartment of interest that could fail in such a way as to create the undesired outcome. As noted earlier, a bounding conditional probability for a hot short of 0.2 was used for each applicable pair of wires.

After the cut sets were generated, they were reviewed in the same manner as was done for the IPE. That is, they were examined to determine that they made logical sense, especially in light of the implicit fire-induced failures, and that human interactions were accounted for appropriately. Many of the cut sets contained one or more basic events reflecting human interactions. These were evaluated using the same quantification methods as in the IPE, taking into account the context of the individual cut sets (including, where applicable, the impact of the fire itself). Opportunities for further operator intervention to prevent core damage (i.e., recovery) were also considered. Some new (relative to the IPE) interactions and recovery events were identified and evaluated. Once the cut sets were determined to be appropriate characterizations of core-damage sequences given the fire-induced faults for an compartment, and the assessment of human interactions was completed, the probabilities for the cut sets for each of the core-

* It should be noted that there is a difference between setting the basic events to "true" and applying a probability of 1.0 in the quantification process. The events whose states are set to "true" are implicitly failed in the cut sets, but do not appear explicitly. This avoids the generation of a large number of supersets, which would occur if the events were retained with probabilities set to 1.0. The hot shorts appear as failure events in the cut sets, and their probabilities can be adjusted at the time of the cut-set review to apply the appropriate value. If the hot shorts are not certain to occur, this approach allows other failures with similar effects to be accounted for properly in the cut sets.

damage sequences identified in the previous section were summed to provide the value of P_2 for the npartment.

4.2.2.4 Results from Initial Screening Quantification

The fire initiation frequencies (F1) and conditional probabilities of core damage (P2) were estimated for each of the fire compartments. If the product of F1 * P2 was less than 1.0E-06, the compartment was considered not to pose the potential for a severe-accident vulnerability, and was removed from further evaluation. This is the cut off value recommended by the FIVE methodology. Of the 70 compartments for which quantitative assessments were applicable, it was possible to screen out 46 based on these bounding calculations. Results for all of the compartments that were screened as a consequence of this initial bounding assessment are summarized in Table 4.2.2.1.

It should be noted that in pursuing this general strategy, no effort was made to revisit previously screened areas when potential conservatisms were subsequently identified. For example, the cable/basic event combinations were initially assigned in a conservative fashion; if detailed circuit analysis performed for examining an unscreened fire area indicated that a specific cable would not, in fact, result in loss of component function, this insight was not factored into previously screened fire areas which had cables for that same circuit. This is consistent with the overall screening process, whereby it is sufficient to simply demonstrate that area-specific risk is <u>less</u> than a particular threshold. Any potential non-conservatisms identified were, of course, factored back into any applicable fire area.

Tab.. 4.2.2.1

Summary of Areas Screened Based on $F_1 \times P_2$

Area	Area Description	Initiation Frequency (F_1; per year)	Significant Credited Mitigating Systems Affected by Fire[1,2]	Conditional CDP (P_2)	Bounding CDF ($F_2 = F_1 \times P_2$; per year)	Note
A.01	Auxiliary Building Rooms 102, 103, 104, 104A, 106, 106A, 107, 108, 109, 109A, 111	1.8×10^{-3}	• Auxiliary feedwater (operator action to trip) • 480 v MCCs	1.0×10^{-4}	1.9×10^{-7}	
A.02	Auxiliary Building Rooms 110, 110A, 112, 116, 117, 117A, 119, 120, 121, 122	3.2×10^{-3}	• Auxiliary feedwater (operator action to trip) • Makeup • Low pressure injection • ECCS room cooling • CCW supply to RCPs	3.1×10^{-4}	9.8×10^{-7}	
A.03	Auxiliary Building Room 114	1.0×10^{-3}	• Auxiliary feedwater (operator action to trip) • High pressure injection • Low pressure injection	1.2×10^{-4}	1.2×10^{-7}	
A.04	Auxiliary Building Room 115	2.9×10^{-3}	• Auxiliary feedwater (operator action to trip) • High pressure injection • Low pressure injection • 480 v MCC F11E • ECCS room coolers	8.9×10^{-5}	2.6×10^{-7}	
A.06	Auxiliary Building Room 127E	5.7×10^{-4}	• Auxiliary feedwater (direct and operator action to trip) • Safety features actuation • PORV • Low pressure injection • Makeup • ECCS room cooling	2.0×10^{-4}	1.1×10^{-7}	

Table 4.2.1 (continued)
Summary of Areas Screened Based on $F_1 \times P_2$

Area	Area Description	Initiation Frequency (F_1; per year)	Significant Credited Mitigating Systems Affected by Fire[1,2]	Conditional CDP (P_2)	Bounding CDF ($F_2 = F_1 \times P_2$; per year)	Note
AB.01	Auxiliary Building Rooms 105, 113, 113A	3.0×10^{-3}	• High pressure injection • Low pressure injection • Makeup • Auxiliary feedwater (operators trip train 1) • 480 v MCCs E12E, F11E, F11F • CCW supply to RCPs	2.9×10^{-4}	8.7×10^{-7}	
AB.02	Auxiliary Building Room 127W	5.7×10^{-4}	• Auxiliary feedwater (direct and operator action to trip) • Startup feedwater • Safety features actuation • Low pressure injection • Makeup • PORV block valve • CCW supply to non-essential headers • RCP seal return	3.0×10^{-4}	1.7×10^{-7}	
AB.04	Auxiliary Building Rooms 225, 226A	2.6×10^{-3}	• Auxiliary operator action to trip) • Low pressure injection • Makeup • RCP seal injection • PORV block valve	2.7×10^{-4}	6.9×10^{-7}	

Table 4.2.... (continued)
Summary of Areas Screened Based on $F_1 \times P_2$

Area	Area Description	Initiation Frequency (F_1; per year)	Significant Credited Mitigating Systems Affected by Fire[1,2]	Conditional CDP (P_2)	Bounding CDF ($F_2 = F_1 \times P_2$; per year)	Note
AB.05	Auxiliary Building Rooms 303, 303PC	1.2×10^{-3}	• Auxiliary feedwater (direct and operator action to trip) • Low pressure injection • Makeup • RCP seal injection and return • PORV block valve • Low voltage switchgear room cooling • 480 v MCC E11E • CCW supply to non-essential headers • Safety features actuation	7.6×10^{-4}	9.4×10^{-7}	
AB.06	Auxiliary Building Room AB3	5.7×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater	7.5×10^{-5}	4.3×10^{-8}	
AC.01	Borated water storage tank pipe trench	5.1×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater • Low pressure injection • Safety features actuation	1.8×10^{-4}	9.3×10^{-8}	

Table 4.2.-.. (continued)
Summary of Areas Screened Based on $F_1 \times P_2$

Area	Area Description	Initiation Frequency (F_1; per year)	Significant Credited Mitigating Systems Affected by Fire[1,2]	Conditional CDP (P_2)	Bounding CDF ($F_2 = F_1 \times P_2$; per year)	Note
B.01	Equipment and pipe chase and pipe tunnel	9.3×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater • High pressure injection • Low pressure injection • Makeup • 480 v MCCs E12E, F11E, F11F • ECCS room coolers	2.1×10^{-4}	1.9×10^{-7}	
BD.01	Screenwash pump Room	6.7×10^{-3}	• Auxiliary feedwater (operator action to trip) • Backup service water	1.1×10^{-4}	7.5×10^{-7}	
BE.01	Diesel fire pump area	1.0×10^{-2}	• Auxiliary feedwater (operator action to trip) • Service water • 480 v MCCs E12C, E12D, F12C	9.5×10^{-5}	9.8×10^{-7}	
BG.01	Service water valve Room no. 2	6.8×10^{-3}	• Auxiliary feedwater (operator action to trip) • Service water • 480 v MCC F12C	1.4×10^{-4}	9.8×10^{-7}	
BH.01	Water treatment building laboratories and Control Room	1.2×10^{-2}	• Auxiliary feedwater (operator action to trip) • Service water	5.6×10^{-5}	6.6×10^{-7}	
BM.01	Diesel oil storage tank and pumphouse	6.4×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater • Service water	4.3×10^{-5}	2.8×10^{-8}	

Table 4.2.... (continued)

Summary of Areas Screened Based on $F_1 \times P_2$

Area	Area Description	Initiation Frequency (F_1; per year)	Significant Credited Mitigating Systems Affected by Fire[1,2]	Conditional CDP (P_2)	Bounding CDF ($F_2 = F_1 \times P_2$; per year)	Note
CC.01	Auxiliary Building elevation 603	8.4×10^{-4}	• Auxiliary feedwater (operator action to trip) • High pressure injection • Emergency diesel generator • Safety features actuation • Service water	1.8×10^{-4}	1.5×10^{-7}	
CC.02	Auxiliary Building ..	1.3×10^{-4}	• Auxiliary feedwater (direct and operator action to trip) • Startup feedwater • High pressure injection • Low pressure injection • Makeup • Low voltage switchgear room cooling • RCP seal injection and seal cooling • Service water	6.6×10^{-3}	8.8×10^{-7}	
DG.01	No. 1 electrical penetration room	1.8×10^{-3}	• Auxiliary feedwater (direct and operator action to trip) • Low pressure injection • Makeup • PORV block valve • 480 v MCCs E11E, E16B • Emergency diesel generator • RCP seal injection and seal cooling • Safety features actuation	5.1×10^{-4}	9.2×10^{-7}	

Table 4.2.... (continued)
Summary of Areas Screened Based on $F_1 \times P_2$

Area	Area Description	Initiation Frequency (F_1; per year)	Significant Credited Mitigating Systems Affected by Fire[1,2]	Conditional CDP (P_2)	Bounding CDF ($F_2 = F_1 \times P_2$; per year)	Note
DH.01	Main steam penthouse	6.8×10^{-4}	• Auxiliary feedwater (direct and operator action to trip) • Main steam	2.4×10^{-4}	1.7×10^{-7}	
DH.02	Main steam penthouse	9.1×10^{-4}	• Auxiliary feedwater (operator action to trip) • Main steam	2.9×10^{-4}	2.6×10^{-7}	
E.01	Auxiliary feed pump no. 1 room	1.2×10^{-3}	• Auxiliary feedwater (direct and operator action to trip) • High pressure injection • Low pressure injection • 4 kV bus C1 • Low voltage switchgear room cooling • Component cooling water • Service water	7.8×10^{-4}	9.0×10^{-7}	
G.01	Makeup and purification rooms	5.7×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater • Safety features actuation	1.7×10^{-4}	9.9×10^{-8}	
G.03	Makeup and purification rooms	5.7×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater • Makeup	4.9×10^{-5}	2.8×10^{-8}	

Table 4.2.... (continued)
Summary of Areas Screened Based on $F_1 \times P_2$

Area	Area Description	Initiation Frequency (F_1; per year)	Significant Credited Mitigating Systems Affected by Fire[1,2]	Conditional CDP (P_2)	Bounding CDF ($F_2 = F_1 \times P_2$; per year)	Note
HH.01	Ac equipment room	4.1×10^{-3}	• Auxiliary feedwater (operator action to trip) • Main steam	4.5×10^{-5}	1.8×10^{-7}	
II.02	Auxiliary Boiler room	2.9×10^{-3}	• Auxiliary feedwater (operator action to trip)	1.7×10^{-5}	4.9×10^{-7}	
II.03	Seal oil room	9.8×10^{-4}	• Auxiliary feedwater (operator action to trip)	3.8×10^{-5}	3.7×10^{-8}	
II.05	Oil drum storage area	6.4×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater	2.8×10^{-4}	1.8×10^{-7}	
II.06	Office building front door vestibule	6.6×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater	2.6×10^{-4}	1.7×10^{-7}	
II.07	Lube oil filter room	7.2×10^{-4}	• Auxiliary feedwater (operator action to trip)	1.1×10^{-4}	7.6×10^{-8}	
II.08	Turbine building…	1.0×10^{-3}	• Auxiliary feedwater (operator action to trip)	9.3×10^{-5}	9.6×10^{-8}	
II.09	Non-rad supply air & exhaust equipment room	6.6×10^{-4}	• Auxiliary feedwater (operator action to trip) • Instrument buses Y1 and Y3	3.8×10^{-4}	2.5×10^{-7}	

Table 4.2.... (continued)
Summary of Areas Screened Based on $F_1 \times P_2$

Area	Area Description	Initiation Frequency (F_1; per year)	Significant Credited Mitigating Systems Affected by Fire[1,2]	Conditional CDP (P_2)	Bounding CDF ($F_2 = F_1 \times P_2$; per year)	Note
J.01	Diesel generator no. 2 room	3.0×10^{-2}	• Auxiliary feedwater (direct and operator action to trip) • Makeup • Emergency diesel generator • Low voltage switchgear room cooling • Misc. power buses	2.4×10^{-5}	7.2×10^{-7}	3
J.02	Diesel generator no. 2 day tank room	7.2×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater • Dc bus P2N	5.2×10^{-5}	3.8×10^{-8}	
K.01	Diesel generator no. 1 room	3.0×10^{-2}	• Auxiliary feedwater (direct and operator action to trip) • Makeup • Emergency diesel generator • Low voltage switchgear room cooling • Misc. power buses	2.5×10^{-5}	7.6×10^{-7}	3
K.02	Diesel generator no. 1 day tank room	7.3×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater	4.8×10^{-5}	3.5×10^{-8}	
MB.01	Manhole MH3004	4.7×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater	1.2×10^{-4}	5.7×10^{-8}	

Table 4.2.z.. (continued)
Summary of Areas Screened Based on $F_1 \times P_2$

Area	Area Description	Initiation Frequency (F_1; per year)	Significant Credited Mitigating Systems Affected by Fire[1,2]	Conditional CDP (P_2)	Bounding CDF ($F_2 = F_1 \times P_2$; per year)	Note
MC.01	Manhole MH3005	4.7×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater	1.0×10^{-4}	4.9×10^{-8}	4
ME.01	Manhole MH3041	4.7×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater	6.1×10^{-5}	2.9×10^{-8}	
MF.01	Manhole MH3042	4.7×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater	4.5×10^{-5}	2.1×10^{-8}	
MG.01	Junction box JB30D4	4.7×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater	6.0×10^{-5}	2.8×10^{-8}	
P.01	Maintenance room	5.7×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater	6.0×10^{-5}	3.4×10^{-8}	
P.02	Passage to diesel generator rooms	8.3×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater	9.1×10^{-4}	7.6×10^{-7}	

Table 4.2.1. (continued)
Summary of Areas Screened Based on $F_1 \times P_2$

Area	Area Description	Initiation Frequency (F_1; per year)	Significant Credited Mitigating Systems Affected by Fire[1,2]	Conditional CDP (P_2)	Bounding CDF ($F_2 = F_1 \times P_2$; per year)	Note
UU.01	Machine room and elevator vestibule	3.7×10^{-3}	• Auxiliary feedwater (direct and operator action to trip) • High pressure injection • Component cooling water • Service water • Emergency diesel generator	2.5×10^{-4}	9.2×10^{-7}	
UU.02	Elevator and stairwell	5.7×10^{-4}	• Auxiliary feedwater (operator action to trip) • Startup feedwater • Makeup • 4 kV bus C2	7.8×10^{-5}	4.5×10^{-8}	

Notes to Table 4.2.2.1

1. Main feedwater is assumed to be unavailable, except as noted for specific areas.
2. Systems identified in Table may be only partially disabled by fire.
3. Main feedwater credited for this compartment (after verifying that system should be unaffected directly or indirectly due to fire in the area)..
4. Potential loss of offsite power for the compartment.

Acronyms used in Table 4.2.2.1

CCW	Component cooling water	ECCS	Emergency core cooling system
PORV	Pilot-operated relief valve	RCP	Reactor coolant pump

⅃.3 Screening Based on Adjusted Initiation Frequencies

Additional screening calculations were made for the compartments remaining after the initial evaluation. These calculations took into account adjustments to the fire frequencies to remove some of the conservatism in the frequencies for specific initiation sources.

It has been determined that among the events included in the fire events data base used to generate the fire initiation frequencies are many fires that are not severe enough to cause significant damage. Severity factors for various fixed sources of ignition, based on a careful review of the data base, have been suggested and are summarized in Table 4.2.3.1 (Ref. 3). For each compartment, the contribution to the fire initiation frequency for each of these sources was multiplied by the appropriate severity factor.

Table 4.2.3.1
SEVERITY FACTORS FOR FIXED FIRE SOURCES

Ignition Source	Severity Factor
Control Room electrical cabinets	0.2
Switchgear Room electrical cabinets	0.12
Indoor transformers	0.1
Diesel generators (skid fires)	0.2
Motor generator sets	0.14
Pumps (not including RCPs and MFW)	0.2
Ventilation subsystems	0.08

In addition to the impact on the fixed fire sources, EPRI review of the Fire Event Data Base (FEDB) determined that manual suppression of welding fires by the fire watch was effective in most cases, even before there was time for response by the fire brigade. An examination of the FEDB indicated that 85% of the welding fires were extinguished by the fire watch before there was significant damage. Thus, the contributions to the fire initiation frequencies due to welding-related fires was reduced by a factor of 0.15 as well.

There were additional features that could be credited to reduce the effective frequency of transient fires further. In addition to the credit for early suppression of welding-related fires by the fire watch, the following factors were taken into account: manual non-suppression for transient fires other than those that are welding-related; an exposure factor that reflects the amount of transient combustibles that might be present and the manner in which they are stored; and an inspection factor, reflecting the frequency of inspections of transients in the fire compartments. An additional factor that would take into account the floor area of potential targets relative to the critical distance of possible fires was considered but was not ›licitly credited.

The first of these additional factors addresses non-suppression of transient fires that are not lding-related. This took into consideration the possibility of suppression by the first responders (potentially before the full fire brigade had assembled). A non-suppression probability of 0.65 was applied if the compartment was equipped with an automatic detection system <u>and</u> if the first responders would be expected to arrive within 5 minutes (i.e., before the fire reached its peak heat release) (Ref. 3). Information from Davis-Besse operations personnel responsible for fire brigade response was used to verify that initial response within 5 minutes was likely for particular compartments (Ref. 18).

The second of the additional factors reflects the potential for transients to be exposed to a fire ignition source. A reduction factor of 0.1 is recommended if, for a given compartment, all of the following criteria are met (Ref. 1):

- Flammable and combustible liquids are stored in approved containers.
- Ordinary combustibles or radiation work protection clothing are stored in enclosed metal cabinets or metal containers with covers actuated by fusible links, or with self-extinguishing lids approved by Factory Mutual.
- All exposed transient combustibles used by plant personnel while working in the compartment are removed upon completion of the work unless otherwise approved.

It was determined that all of these criteria were met for the fire compartments of concern at Davis-Besse (Ref. 19). In addition to these three criteria, other possible combustible materials were considered. These included solid materials used for radiation protection, such as anti-contamination clothing, step-off ls, etc. It was determined that the cloth bags used for the collection of used anti-contamination clothing are fire resistant (Ref. 20). Furthermore, although plastic bags are sometimes present at the boundaries for radiation protection compartments, these plastic bags are present in very small quantities compared to the floor area of the compartment, and are not located near potential ignition sources. Therefore, it was judged that they could be neglected. In addition, step-off pads represent very low combustible loadings, and cover only a very small portion of the floor area in any given compartment. Therefore, it was judged that the exposure factor of 0.1 could reasonably be applied for the transient fires at Davis-Besse.

The third of the reduction factors identified above accounts for the frequency of inspections that would verify compliance with requirements for control of transient combustibles. The transient fire frequencies effectively reflect conditions over a one-year period, assuming no inspections. Because fire protection inspections are conducted on a monthly basis at Davis-Besse, a factor of 1/12 can be applied to the initiation frequencies for transient fires.

As discussed above, the reduction factors are based on consideration of transient fire sources. To be consistent with the FIVE methodology, however, as shown below these were applied to the <u>sum</u> of the transient and fixed ignition source frequencies.

For the compartments that were not screened as a result of the initial quantitative step, the initiation frequencies were adjusted using the severity factors for fixed sources, the non-suppression probabilities for transient fires (0.15 for welding-related fires and 0.65 for other transient fires in compartments provided h automatic detection), and the exposure and inspection factors for transient fires. The adjusted frequencies were then multiplied by the conditional core-damage probabilities to obtain new screening imates of core-damage frequency:

$$CDF=(\Sigma IF_{fixed}*SF_{fixed})*(CCDP_{all}) + (IF_{trans}+ \Sigma IF_{fixed}) * MS_{trans} * CCDP_{all}$$

Where: IF_{fixed} = Sum of the fixed ignition source frequencies

SF_{fixed} = Fire severity factor for the fixed sources

$CCDP_{all}$ = CCDP with all equipment in the compartment failed

IF_{trans} = Frequency of the transient ignition source(s)

MS_{trans} = Probability of failure to manually suppress transient fires

Substituting and expanding,

$$CDF = [(\Sigma IF_{fixed}*SF_{fixed}) * (CCDP_{all})] + \{[(IF_{trans,weld}*MS_{trans,weld}) + (IF_{trans,nonweld})*MS_{trans,nonweld})] + \Sigma IF_{fixed}\} * EF * INF * CCDP_{all}$$

Where: $IF_{trans,weld}$ = Welding transient ignition source frequency

$MS_{trans,weld}$ = Manual non-suppression probability for welding transient ignition fires

$IF_{trans,non\text{-}weld}$ = Non-welding transient ignition source frequency

$MS_{trans,non\text{-}weld}$ = Manual non-suppression probability for non-welding transient ignition fires

EF = Exposure factor

INF = Inspection frequency

Substituting fixed numerical values into this equation yields,

$$CDF = [(\Sigma IF_{fixed}*SF_{fixed})*(CCDP_{all})] + \{[(4.0E\text{-}4*0.15) + (1.7E\text{-}4*0.65)] + \Sigma IF_{fixed}\} * 0.10 * 0.083 * CCDP_{all}$$

or,

$$CDF = [(\Sigma IF_{fixed}* SF_{fixed}) * (CCDP_{all})] + (1.7E\text{-}4 + \Sigma IF_{fixed}) * (8.3E\text{-}3) * CCDP_{all}$$

The use of adjusted initiating frequencies and severity factors allowed an additional 15 compartments to screen. These are summarized in Tables 4.2.3.2.a and 4.2.3.2.b. For the entries in Table 4.2.3.2.a, the adjusted initiation frequencies reflect only the severity factors for fixed sources of ignition and the immediate non-suppression probability for welding-related fires. The compartments screened without the need to credit the other reduction factors for transient fires. The entries in Table 4.2.3.2.b include the use of the reduction factors for transient fires.

Table ..2.3.2.a
Summary of Compartments Screened Based on Adjusted Initiation Frequencies

Compartment	Compartment Description	Total Fire Frequency (F_1; per year)	Effective Frequency (F_1^*; per year)	Important Fire Effects	Conditional CDP (P_2)	Bounding CDF ($F_2^* = F_1^* \times P_2$; per year)
A.07	No. 2 Mechanical penetration room	1.3×10^{-3}	3.5×10^{-4}	Equipment failures: • Main feedwater (assumed) • 480v MCC F11 Operator actions: • Trip AFW pump 2 • Isolate AFW 2 flowpaths	2.0×10^{-3}	7.1×10^{-7}
AB.03	No. 1 Mechanical penetration room & pipeway area	1.2×10^{-3}	3.4×10^{-4}	Equipment failures: • Main feedwater (assumed) Operator actions: • Trip AFW pump 1 • Isolate AFW 2 to SG 1 flowpath	9.3×10^{-4}	3.1×10^{-7}
BF.01	Service water pump room	6.2×10^{-3}	1.2×10^{-3}	Equipment failures: • Main feedwater (assumed) Operator actions: • Trip AFW pump 2 • Isolate AFW 2 flowpaths	6.2×10^{-4}	7.2×10^{-7}
MA.01	Manhole	4.7×10^{-4}	7.0×10^{-5}	Equipment failures: • Main feedwater (assumed) Operator actions: • Trip AFW pump 2	4.4×10^{-4}	3.1×10^{-7}
V.02	Spent fuel storage area	4.5×10^{-3}	8.1×10^{-4}	Equipment failures: • Main feedwater (assumed) Operator actions: • Trip AFW pump 1	5.3×10^{-4}	4.3×10^{-7}

Table ..2.3.2.b
Summary of Compartments Screened Based on Adjusted Initiation Frequencies

Compart-ment	Compartment Description	Fixed Ignition Source Freq. (1/yr) Total (F_I) / Effective (F_I*)	Adjusted Transient Fire Freq. (1/yr) (F_T*; per year)	Important Fire Effects	Conditional CDP (P_2)	Bounding CDF [(F_I* x P_2)+(F_I* x P_2)]; (1/year)	Note
DF.01	No. 2 Electrical penetration room	3.1×10^{-3}/ 3.6×10^{-4}	4.4×10^{-6}	Equipment failures: • Main feedwater (assumed) Operator actions: • Trip AFW pump 2 • Isolate AFW 2 flowpath	2.7×10^{-3}	9.9×10^{-7}	1
EE.01	Radwaste and purge air handling equipment area	4.0×10^{-3}/ 3.6×10^{-4}	5.1×10^{-6}	Equipment failures: • Main feedwater (assumed) Operator actions: • Trip AFW pump 2 • Isolate AFW 2 flowpaths	1.7×10^{-3}	7.6×10^{-7}	
F.01	Auxiliary feedwater pump No. 2 room	4.2×10^{-4}/ 6.3×10^{-5}	1.9×10^{-6}	Equipment failures: • Main feedwater (assumed) • AFW pump 2 • Startup feed pump Operator actions: • Trip AFW pump 2 • Isolate AFW 2 flowpaths	1.3×10^{-2}	8.6×10^{-7}	
G.04	Liquid radwaste and boration equipment area	3.8×10^{-3}/ 7.6×10^{-4}	7.8×10^{-6}	Equipment failures: • Main feedwater (assumed) Operator actions: • Trip AFW pump 1 • Isolate AFW 2 to SG 1flowpath	6.5×10^{-4}	5.0×10^{-7}	

Table 4.2.3._.o (continued)
Summary of Compartments Screened Based on Adjusted Initiation Frequencies

Compart-ment	Compartment Description	Fixed Ignition Source Freq. (1/yr) Total (F_I) / Effective (F_I^*)	Adjusted Transient Fire Freq. (1/yr) (F_T^*; per year)	Important Fire Effects	Conditional CDP (P_2)	Bounding CDF [(F_I^* x P_2)+(F_I^* x P_2)]; (1/year)	Note
P.03	Passage to diesel generator rooms	2.7×10^{-5}/ 3.3×10^{-6}	1.5×10^{-6}	Equipment failures: • Main feedwater (assumed) Operator actions: • Trip AFW pump 2	5.5×10^{-2}	2.6×10^{-7}	
T.01	Component cooling water pump room	1.0×10^{-3}/ 1.7×10^{-4}	2.8×10^{-6}	Equipment failures: • Main feedwater (assumed) • Component cooling water pumps Operator actions: • Trip AFW pump 2	4.3×10^{-3}	7.3×10^{-7}	
V.01	Spent fuel storage area	1.5×10^{-3}/ 2.1×10^{-4}	3.2×10^{-6}	Equipment failures: • Main feedwater (assumed) Operator actions: • Trip AFW pump 1	4.4×10^{-3}	9.2×10^{-7}	2
X.02	Station Battery Room B	1.6×10^{-3}/ 1.6×10^{-3}	1.5×10^{-5}	Equipment failures: • Main feedwater (assumed) Operator actions: • Trip AFW pump 2	3.2×10^{-4}	5.2×10^{-7}	

Table 4.2.3.2.b (continued)
Summary of Compartments Screened Based on Adjusted Initiation Frequencies

Compart-ment	Compartment Description	Fixed Ignition Source Freq. (1/yr) Total (F_1) / Effective (F_1^*)	Adjusted Transient Fire Freq. (1/yr) (F_T^*; per year)	Important Fire Effects	Conditional CDP (P_2)	Bounding CDF [(F_1^* x P_2)+(F_1^* x P_2)]; (1/year)	Note
Y.01	No. 2 Low Voltage Switchgear Room	4.2×10^{-3}/ 4.8×10^{-4}	5.4×10^{-6}	Equipment failures: • Main feedwater • Essential bus C1 • Low-voltage ac and dc buses (train 1) Operator actions: • Trip AFW pump 2	2.0×10^{-3}	9.6×10^{-7}	
Y.02	Station Battery Room A	1.6×10^{-3}/ 1.6×10^{-3}	1.5×10^{-5}	Equipment failures: • Main feedwater (assumed) • Station battery (train 1) Operator actions: • Trip AFW pump 1	4.4×10^{-4}	7.1×10^{-7}	

Notes to Table 4.2.3.2.b

1. Potential for fire induced PORV opening for this compartment.
2. For this compartment, the value of P_2 was calculated both assuming offsite power was lost and not assuming so. The higher of the two values was used in assessing the calculated bounding core-damage frequency.

4.2.4 Compartment-Specific Screening Assessments

Some compartments were subjected to screening assessments using the approaches outlined in the preceding two sections, but with special considerations for the arrangement of equipment or locations of potential sources of ignition. The limited fire modeling described below was also taken into consideration.

4.2.4.1 Use of FIVE Fire Modeling

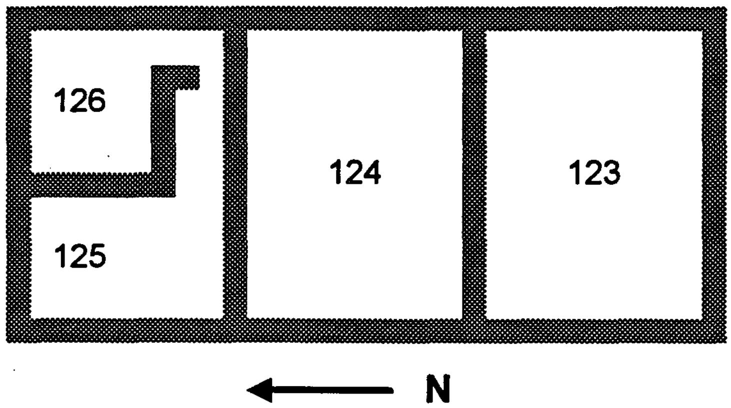
The fire modeling portion of the FIVE methodology can be used to determine if the heat release from fixed and transient sources is large enough to cause damage to targets in the room. There are basically three scenarios that can be evaluated with the FIVE fire modeling software. The first is for the target located within the plume of the fire. The next is for the target to be located outside the plume. This includes the effects of the ceiling jet sublayer and the bulk rise in the hot gas layer temperature of the compartment. The last case that can be evaluated is for radiant exposure of the target to the fire. From the scenarios applicable for the particular compartment, FIVE can be used to calculate a time to critical damage of the target based on the characteristics of the source, the target, and the selected damage threshold criteria. Phase II (step 3) of the FIVE methodology can be extended such that probabilities of fire damage, based upon the scenario characteristics and the critical combustible loading of each compartment, can be determined. This final extension has not been used in the Davis-Besse fire analysis.

A limited use of FIVE's fire modeling capability, however, has been utilized in this analysis. .eening criteria were developed to be used in the field during walkdowns of the fire area. In particular, closed, non-ventilated low energy (i.e., < 480 V) cabinets were not considered to be a credible ignition source. Ventilated electrical cabinets and panels with IEEE 383 cable, dry transformers and motors were characterized with a heat release rate of 65 BTU/sec (Ref. 3). Generally, the approach used the geometry of a typical room, and based on the fire intensity of 65 BTU/sec an equivalent area of oil (about 0.5 square feet). was modeled. Cases were run for cables exposed to radiant energy to the side and convection and radiation to a target in the plume. Thermal damage criteria of 700 degree-F and 1.0 BTU/ft^2/sec combined radiant and convective critical flux were utilized, which are appropriate for the qualified cable used in the Davis-Besse Nuclear Power Station. By this method, a distance was determined which assures the exposed components are undamaged within the nominal response time of the fire brigade. This distance was been determined to be under two feet for radiant heat transfer alone and approximately four feet for convection and radiation to a target in the plume. The horizontal damage distance from the fire to a cable near the ceiling due to the ceiling jet is four feet, and the time required for bulk room heatup is large for all normal sized rooms. These conclusions are supported by curves from the "Methods of Quantitative Fire Hazard Analysis", EPRI TR-100443. It should be noted that these distances are for unprotected, exposed cables. At Davis-Besse, essentially all cables are contained in either conduits or solid bottom cable trays which provides significant additional thermal protection. Therefore, when considering the potential impact of a fire source on surrounding equipment during walkdowns for this screening analysis, conduits and cable trays outside of the above distances were eliminated as being credibly damaged.

It should also be noted that a considerable amount of work was done to characterize the fire ›perties of oil filled transformers located in the Low Voltage Switchgear Rooms. Details of this characterization are found in Section 4.2.5.6.

When postulating a particular piece of equipment to be the fire source, the source is of course assumed to be damaged. If there are no other pieces of equipment within the calculated maximum damage distance, and other considerations such as hot gas layer formation, and bulk room temperature were shown to also not affect other equipment, the fire source was eliminated from further consideration. This approach is consistent with the EPRI Fire PRA Analysis guide.

4.2.4.2 Compartment A.05 (Clean Waste, Detergent Waste, & Miscellaneous Waste Tanks)



Compartment A.05 includes Rooms 123 & 124 (Clean Waste Receiver Tank Rooms), 125 (Detergent Waste Drain Tank Room), and 126 (Miscellaneous Waste Drain Tank Room) in the Auxiliary Building. The fixed ignition source data indicates that all of the fire frequency in this compartment comes from pumps. There are several sump pumps present, as well as four larger pumps which service various liquid radwaste storage tanks. The larger pumps are all located in Room 125. The sump pumps (2 oz. oil) do not represent a threat to any significant components or cables due to their distance from other equipment. The four larger pumps (none of which are safe shutdown components) and their associated volume of lubricating oil are:

Detergent Waste Drain Tank Pump (P52)	1 pint oil
Miscellaneous Waste Drain Tank Pump (P51)	2.5 pints oil
Clean Waste Receiver Tank Transfer Pumps (P49-1, 2)	2 quarts oil

The only significant components in Room 125 are cables which provide power to the train 2 ECCS pumps. These are located in conduits along the northern portion of the western wall. Physical measurements show these conduits to be located at least 12 ft. from the Detergent Waste Drain Tank Pump, the nearest of the four large pumps. To obtain an indication as to the potential of postulated pump fires to affect the conduits, a bounding FIVE calculation (Ref. 1) was performed for a fire involving 0.5 'lons of oil. A critical radius of less than 11 ft. was estimated based on the maximum potential intensity of this quantity of oil. Additionally, the quantity of oil available was found to be insufficient to generate a : gas layer in the large volume of A.05. [It should be noted that the above pumps are relatively close together. As such, the characteristic fire was taken as 0.5 gallons of oil at the position of the nearest pump, P52. There is a floor drain between pump P52 and the ECCS conduits, located approximately 5 feet from the conduits. Only the 1 pint of oil from pump P52, however, might be postulated to actually migrate toward this drain. This is of insufficient quantity to present a credible threat to the ECCS cables contained in these relatively large conduits.]

With the actual distance between the conduits and the nearest pump (~ 12 ft.) greater than the critical radius (~ 11 ft.), and the quantity of combustible material insufficient to establish a hot gas layer in the compartment, these fixed ignition sources do not represent a credible threat to any significant components. Therefore the risk from fixed sources in this compartment is negligible.

With respect to transient initiated fires, as stated above, the only significant components in Room 125 are cables located in conduits along the northern portion of the western wall. These conduits are well away from the access doorway which leads to Room 124 and on to Room 123, both of which contain significant cables. As such, postulated fires from transient ignition sources that could affect the significant conduits in Room 125 would not be able to impact the other rooms of this compartment. With only train 2 ECCS pumps affected, the value for $CCDP_{all}$ for a fire from a transient source in Room 125 would be small. In addition, because of the small floor area associated with the ECCS conduits, there is limited exposure to such sources. Therefore, the risk from transient fires in Room 125 is insignificant.

There are no cables or components of significance in Room 126. In Rooms 123 and 124, cables of importance include both trains of Auxiliary Feedwater and ECCS pumps. Per the Fire Hazards Analysis Report, however, these trains meet the intent of Appendix R III.G.2.c separation criteria by virtue of adequate separation distance, with detection and suppression present in the rooms. As such, consistent with the FIVE methodology, the loss of only one path at a time from a single fire was assumed.

Therefore, $CCDP_{all}$ was calculated separately for Train 1 and Train 2 components unavailable to determine the most limiting case. Results indicate a $CCDP_{all}$ value of 1.35E-2 for train 1 unavailable and 1.65E-2 for train 2 unavailable. The higher of these two values was used in the screening CDF calculation.

In Rooms 123 and 124, the only fixed ignition sources are small pumps in the sumps of these large tank rooms. Separation between these small pumps and other cables or components is sufficient such that any fires from these minor sources will only damage the sump pumps themselves, and no hot gas layer would be formed. Thus, in these rooms, only transient ignition sources are capable of damaging significant plant equipment.

With these simplifications, the screening CDF value was calculated:

$CDF = F^{*}_{1,transient} * P_{2,all\ failed}$

Where: $F^{*}_{1,transient}$ = adjusted initiation frequency for transient sources

$= 3.8 \times 10^{-6}/yr$

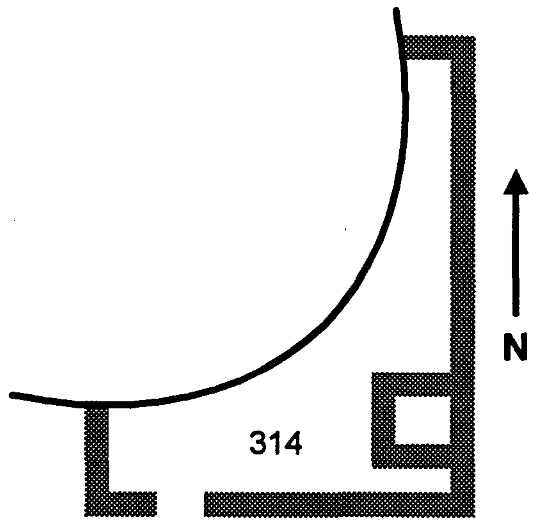
$P_{2,all\ failed}$ = CCDP given failure of all equipment in compartment
= 1.7×10^{-2}/yr

CDF = (3.8×10^{-6}/yr) * (1.7×10^{-2}/yr)

CDF = 6.3×10^{-8}/yr

Given the value well below 1E-6, fire compartment A.05 readily screens from further consideration.

4.2.4.3 Compartment A.08 (No. 4 Mechanical Penetration Room)



Compartment A.08 is comprised of Room 314 in the Auxiliary Building. In this compartment the potential fixed sources of ignition are widely separated from cables in the room that are potentially of concern. In addition, there are no significant intervening exposed combustibles present which might represent a source of secondary ignition. Moreover, the fixed sources are not large enough to produce a hot gas layer that could affect these cables. Transient materials, which could be located virtually anywhere in the room, do not necessarily have the same degree of separation, and thus were assumed to have the capability of affecting the important cables.

Therefore, two sets of calculations for the value of P_2 were made. In the first set, all of the cables and other relevant equipment in the compartment were assumed to fail as a consequence of a fire. The conditional core-damage probability calculated under this assumption was multiplied only by the ignition frequency associated with transient-related fires. The second set of calculations was made assuming that selected cables that were determined to be unaffected by fires from fixed sources were not failed. This second conditional core-damage probability was then multiplied only by the ignition frequency for fixed sources. The two resulting products were then added to provide an overall core-damage frequency for the compartment. In both cases, no explicit credit was taken for the automatic fire suppression in this room.

The screening frequency calculated for compartment A.08 is:

$CDF = F^{*}_{1,fixed} * P_{2,partial\ failed} + F^{*}_{1,transient} * P_{2,all\ failed}$

Where: $F^{*}_{1,fixed}$ = adjusted initiation frequency for fixed ignition sources

$= 2.7x10^{-4}/yr$

$P_{2,partial\ failed}$ = CCDP given failure of equipment affected by fixed sources

$= 3.0x10^{-3}$

$F^{*}_{1,transient}$ = adjusted initiation frequency for transient sources

$= 3.7x10^{-6}/yr$

$P_{2,all\ failed}$ = CCDP given failure of all equipment in compartment

$= 8.5x10^{-2}$

$CDF = (2.7x10^{-4}/yr) * (3.0x10^{-3}) + (3.7x10^{-6}/yr) * (8.5x10^{-2})$

$CDF = 1.1x10^{-6}/yr$

As can be seen, the frequency is slightly greater than the screening criterion. As noted previously, however, no explicit credit was taken for the automatic fire suppression in this room. If this would have been done, use of the reduction factor for operation of the sprinkler system could have been utilized, reducing the estimated conditional core damage frequency. Based on these factors, it is concluded that ther refinement of the fire induced CDF of this area would readily have a value of less then 1.0E-6, and this compartment is considered screened based on quantitative and qualitative analysis.

4.2.4.4 Compartment DD.01 (Cable Spreading Room)

Fire compartment DD.01 is the plant Cable Spreading Room which is located beneath the Control Room. This area contains essential cabling in both solid bottom cable trays and in conduit, and is surrounded by reinforced concrete and 12 inch concrete block construction having a fire resistance rating of three hours. The area is provided with an automatic fire detection system, and is also provided with a dual-feed automatic wet-pipe sprinkler system which provides area protection. Manual fire suppression equipment is also provided for this area.

The FIVE methodology assigns a location specific fixed fire ignition frequency of 3.2E-03 to a cable spreading room, based on the presence of electrical cabinets. Review of plant documentation and a field walkdown has verified that there are only two fixed ignition sources in this compartment. These consist of a single, small lighting panel and a dry lighting transformer in close proximity to the panel. The lighting panel is closed and unvented, and is low energy (< 480 V).

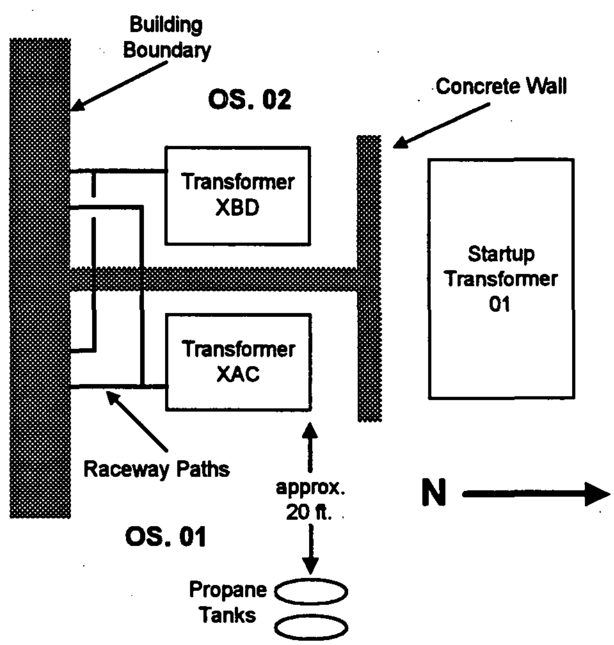
Since cable with fire characteristics equivalent to qualified cable is used for the small lighting panel in this room, the heat release rate for the panel is expected to be no more than the 65 BTU/sec for the small transformer. As described previously, the critical radial distance for radiant energy damage is about two feet, and the threshold vertical elevation distance for damage is about four feet. No significant safe shutdown cables are within these distances.

Since the lighting panel is close to the ceiling, the possibility of formation of a hot gas jet was also examined. A damage distance of four feet is expected under these conditions. Due to the proximity of the lighting transformer, both the panel and transformer were considered for this possibility. A check of plant drawings of the area around the lighting panel/transformer and a field walkdown indicated that no significant safe shutdown cables are within this distance. It is therefore concluded that there are no significant fixed ignition sources in the cable spreading room that can reasonably be capable of creating or supporting a damaging fire.

Transients brought into this area are subject to Auxiliary Building transient combustible and ignition source administrative requirements. Cables in this area are either in metal-bottom trays with Kaowool covering or in conduit. Since there are no other exposed combustibles in the area, welding/cutting fires are also unlikely to generate a significant fire. As in all plant areas, a thirty minute fire watch is required by procedure after all welding and cutting activities. The automatic fire detection and suppression systems present also reduce the likelihood of a transient fire source either creating or supporting a damaging fire.

Therefore, based on the absence of credible ignition sources and combustibles which could sustain a fire of sufficient magnitude to damage significant components in the cable spreading room, this compartment is to be considered screened on a qualitative basis.

4.2.4.5 Compartments OS.01 and OS.02 (Bus Tie Transformers)



Compartments OS.01 and OS.02 are outside areas that contain startup transformer 01 and two step-wn transformers (XAC and XBD; see figure above). Each of the transformers is separated from the others by concrete walls. Although not explicitly credited, each transformer has an oil collection basin underneath and a deluge suppression system to limit the potential for fire spread. As a result of the walkdown of this area, it was concluded that fires could not propagate directly from any of the three transformers to either of the other two, and the area was divided into two separate fire areas. In addition, the transformers are sufficiently separated from the plant such that a fire originating in either OS compartment would not interact with the turbine building or the Auxiliary Building.

It should be noted that none of the severity factors previously discussed were utilized in estimating the risk from either of these compartments.

In OS.01 two 250 gallon propane storage tanks are located approximately 20 ft. east (line of sight) of the transformers. Because a fire involving one of the tanks could affect both startup transformer 01 and transformer XAC, it was decided to calculate a value for P_2 assuming both of these (as well as the cables associated with them) were affected. Application of the F1 * P2 yields the following:

$$CDF = F_1 * P_2$$

$$CDF = (2.31E\text{-}3 / yr) * (6.52E\text{-}5)$$

$$= 1.51E\text{-}7 / yr$$

This results in a core damage frequency less than the screening criterion, and this compartment can be screened. Note: Startup transformer 01 has not been evaluated as a separate fire area, as it is sufficiently isolated such that it does not affect other components. As noted above, however, it was assumed to be affected during the estimation of P_2 for OS.01.

In OS.02, a fire frequency of 2.24E-3 per year was calculated assuming only transformer XBD (which is protected by the concrete walls from the propane tanks) would be affected. Application of the F1*P2 yields the following:

$$CDF = F_1 * P_2$$

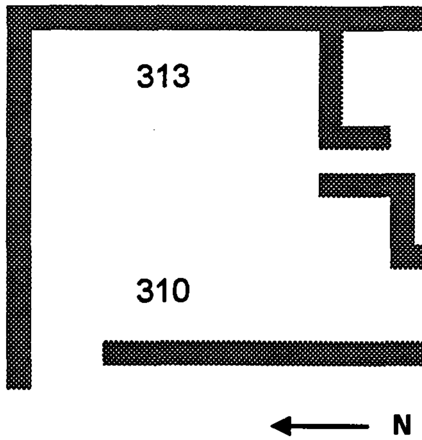
$$CDF = (2.24E\text{-}3/yr) * (1.84E\text{-}4)$$

$$= 4.12E\text{-}7 / yr$$

This results in a core damage frequency less than the screening criterion, and this compartment can be screened.

For conservatism, the result of summing these two areas was also checked. Adding the above calculated bounding core-damage frequencies for compartments OS.01 and OS.02 results in a total CDF of (1.51E-7 + 4.12E-7) = 5.63E-7, which remains below the screening criterion.

4.2.4.6 Compartment U.01 (Spent Fuel Pool Pump Room and Mix Tanks and Hatch Area)



Compartment U.01 encompasses three rooms in the Auxiliary Building, including the room housing the spent fuel pool cooling pumps (Room 312). With severity factors applied, a fire induced core damage frequency of 1.5E-6 per reactor year is calculated, which is slightly above the screening criterion of 1.0E-6.

$$CDF = F^{*}_{1,fixed} * P_{2,all\ failed} + F^{*}_{1,transient} * P_{2,all\ failed}$$

Where: $F^{*}_{1,fixed}$ = adjusted initiation frequency for fixed ignition sources

$= 2.4\text{x}10^{-4}/yr$

$P_{2,all\ failed}$ = CCDP given failure of all equipment in compartment

$= 6.3\text{x}10^{-3}/yr$

$F^{*}_{1,transient}$ = adjusted initiation frequency for transient sources

$= 3.4\text{x}10^{-4}/yr$

$$CDF = (2.4\text{x}10^{-4}/yr) * (6.3\text{x}10^{-3}/yr) + (3.4\text{x}10^{-4}/yr) * (6.3\text{x}10^{-3}/yr)$$

$$CDF = 1.5\text{x}10^{-6}/yr$$

As shown above, the fire area consists of a passageway (Room 310), the hatch area (Room 313), and in a separate room, the Spent Fuel Pool Cooling equipment room (Room 312). The latter is separated from the others by a full wall with a door and a ventilation opening through it. The door is normally closed except to allow passage between the rooms. The ventilation opening is approximately 10 feet above the floor and is 3 feet by 3 feet in size.

The critical cables in this fire area are divided by train. The Train 2 cables of concern primarily ṇsit from floor to ceiling in the south end of Room 310 and do not enter Room 312. The Train 1 cables of interest enter the northwest corner of Room 310, turn 180° in the horizontal plane, and then turn vertically and exit the ceiling of the room to enter the cable chases leading to the Cable Spread Room. They also do not enter Room 312.

There are several mitigating factors associated with this room that give confidence that the room would easily screen with detailed fire modeling. Because of this, further analysis was not performed. Among the reasons for this conclusion are:

1. The area is divided into rooms with a substantial barrier between the SFP cooling pumps and the critical damage targets. Accounting for this separation would significantly reduce the P_2 value for a fire started by these ignition sources.
2. There are wet pipe sprinklers throughout Rooms 310 and 313, particularly between the ventilation opening of the SFP Cooling Room and the critical damage targets. This would further decrease the probability that the targets would be damaged by a fire initiated by the SFP Cooling pumps.
3. The main part of this fire area is a passageway from a stairwell and elevator to other portions of the plant. Consequently it is heavily traveled. This increases the probability of fire detection beyond the typical value for ionization detectors that are also present in the area.

Based on these factors, it is concluded that further refinement of the fire-induced CDF for this area ..ould readily have a value of less then 1.0E-6. Therefore this area is considered screened based on quantitative and qualitative analysis.

4.2.5 Compartments Requiring More Detailed Analysis

After the bounding analysis of the compartments described in the preceding sections was completed, it was necessary to perform somewhat more detailed analyses for the areas which had not yet screened to assess whether fires in them might pose the possibility of a vulnerability. These more detailed analyses in some cases took into account automatic suppression or more careful consideration of spatial arrangements and the potential for fire propagation.

4.2.5.1 Treatment of Automatic Fire Suppression

For compartments in which automatic suppression was credited for fixed sources of fire initiation, it was necessary to consider both the probability that the suppression system would not function and the possibility that core damage could result even if it did function as designed. Manual suppression was only credited for fires initiated by transient sources. The event tree shown in Figure 4.2.5.1 illustrates the manner in which automatic suppression was taken into account for fixed sources.

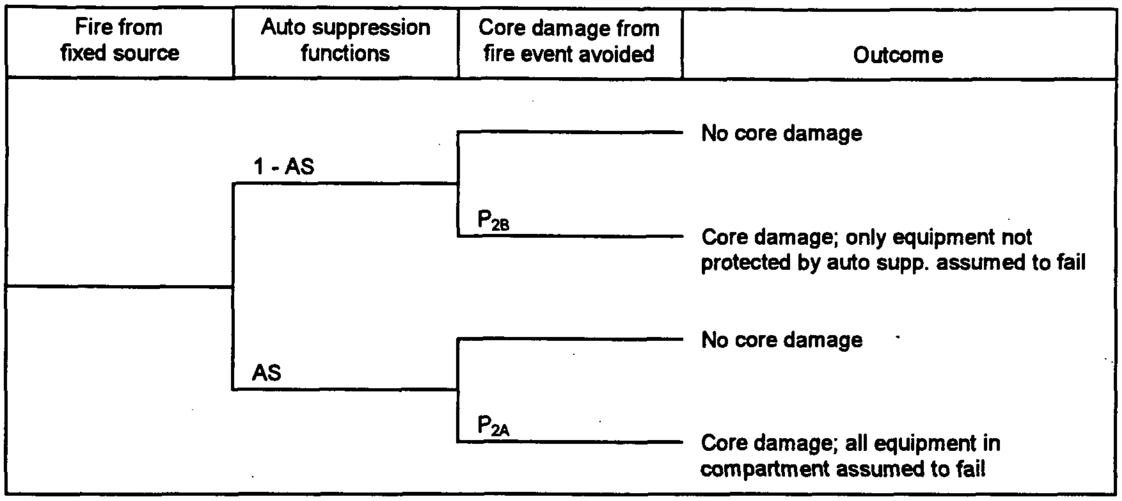
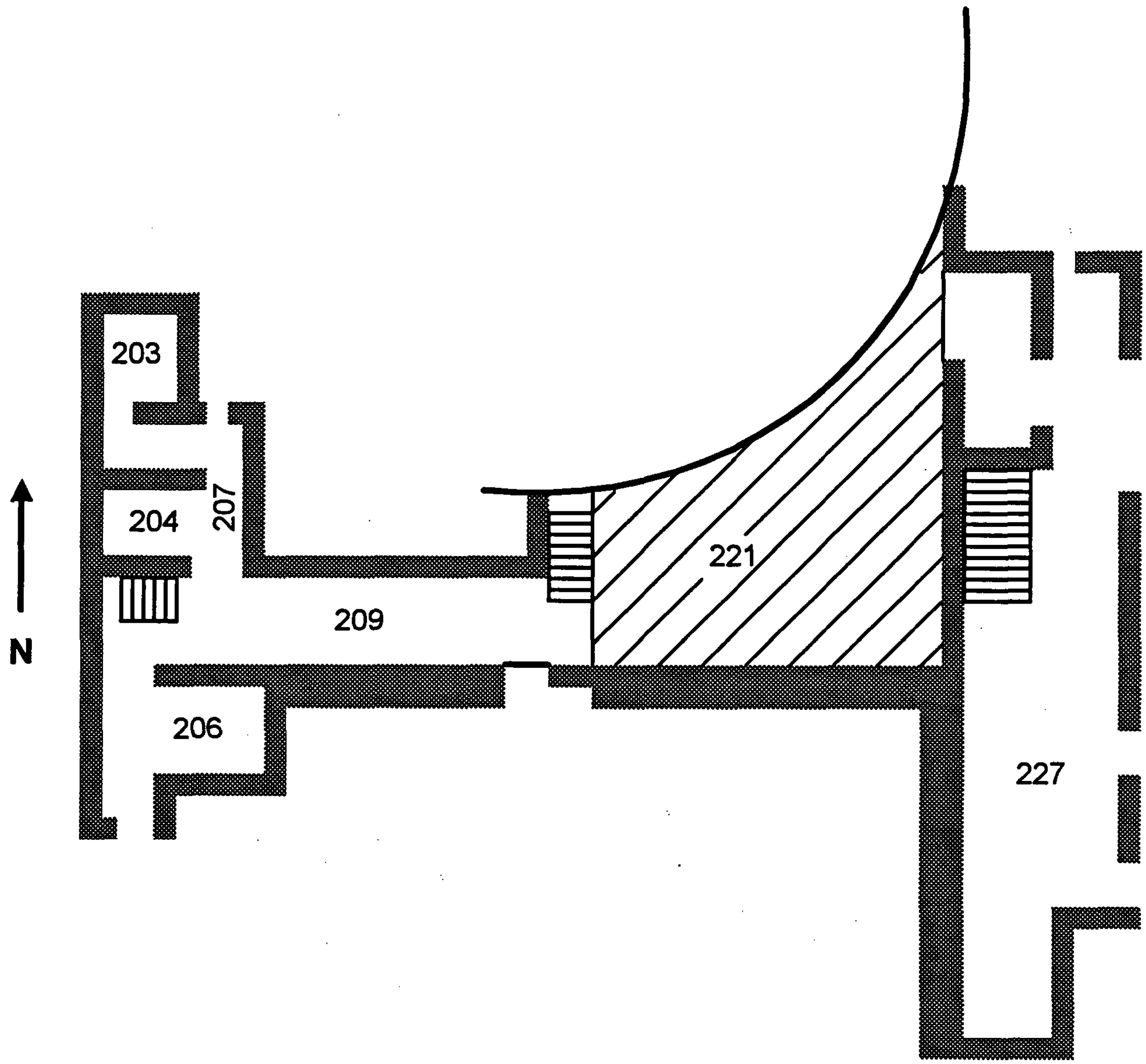


Figure 4.2.5.1 Event Tree for Considering Impact of Automatic Suppression

The value of P_{2A} reflects the failure of all equipment and cables in the compartment, just as though there were no suppression system present. The value calculated for P_{2B} represents the conditional core-damage probability assuming that only cables or other equipment that is not protected by the fire suppression system are affected. Protection was assumed to be afforded if the suppression system could limit the effects of a fire from a fixed source. During walkdowns, the coverage of the suppression system was verified. The value for P_{2A} is used for transient sources, as this is more characteristic of the potential nage from this type of ignition source than P_{2B}. Therefore, the calculated bounding core-damage rrequency when the availability of automatic suppression is considered can be calculated from the expression:

$$CDF = F^{*}_{1,fixed} * AS * P_{2A} + F^{*}_{1,fixed} * (1-AS) * P_{2B} + F^{*}_{1,transient} * P_{2A}$$

'.5.2 Compartment G.02 (Auxiliary Building 565' Elevation, West Passageways)



As shown, compartment G.02 includes several small rooms and two passageways in the Auxiliary Building. Rooms 203, 204, 206, and 207 connect to one end of a corridor designated as Room 209. At the other end of this corridor is an elevated passageway (Room 221). This passageway connects to Room 227. Rooms 209 and 227 are each equipped with an automatic wet-pipe suppression system. If the suppression system were to function properly, fires could not propagate from the room(s) at either end of the elevated passageway (Room 221) to the room(s) at the other end. Thus, it was decided to partition this compartment into three cases for separate analysis. They are:

Case A: The fire initiates in Rooms 203, 204, 206, 207 or 209. Although an automatic suppression system is in Room 209, the placement of sprinkler heads and spray shields may decrease the effectiveness of suppression for some fixed ignition sources. As such, no explicit credit was taken for automatic suppression for fires initiated in these rooms (203, 204, 206, 207, and 209). Cables in Room 221 (the elevated passageway connected to the corridor that constitutes Room 209) were also assumed to be affected by a fire initiated in these rooms. Even if the automatic suppression system in Room 209 fails to provide adequate suppression, it is assumed that the fire would not propagate through Room 221 to affect the cables in Room 227, given the lack of combustibles in Room 221 and the presence of a 3 foot deep steel beam/concrete structure in the path of hot gases.

Case B: The fire initiates in Room 221. If the automatic suppression system functions in Rooms 209 and 227, only the cables in Room 221 were assumed to fail. If the suppression system does not function, all of the cables in all of the rooms were assumed to be affected.

Case C: The fire initiates in Room 227. If the automatic suppression system functions, only the unprotected cables in Room 221 were assumed to be affected. If the fire suppression system does not function, all of the cables in Rooms 221 and 227 were assumed to be affected.

Based on available data, an unavailability for the wet-pipe suppression system (designated as "AS" in Figure 4.2.5.1) of 0.02 was assumed (Ref. 3). For each of these three cases, it was necessary to calculate two values for P_2 (designated as "P_{2A}" and "P_{2B}" in Figure 4.2.5.1). The calculated bounding core-damage frequency for each of the three cases was determined from the expression provided above, and then ...nmed. As indicated by the summary of the relevant values provided below, the compartment readily ...reens due primarily to the large degree of train separation of cables on either side of the elevated passageway (Room 221).

	Case A	Case B	Case C
Adjusted initiation frequency for fixed sources ($F'_{1,fixed}$) (/yr)	2.1×10^{-4}	4.0×10^{-5}	4.5×10^{-4}
Adjusted initiation frequency for transient sources ($F'_{1,transient}$) (/yr)	2.2×10^{-6}	8.0×10^{-7}	4.2×10^{-6}
Unavailability of wet-pipe suppression (AS)	1.0	0.02	0.02
Conditional probability of core damage given failure of all equipment in compartment (P_{2A})	2.2×10^{-4}	1.2×10^{-3}	4.6×10^{-4}
Conditional probability of core damage given failure of equipment not protected by suppression (P_{2B})	N/A	2.1×10^{-4}	2.1×10^{-4}
Core-damage frequency (/yr)	4.7×10^{-8}	1.0×10^{-8}	1.0×10^{-7}
Composite core-damage frequency for compartment G.02 (/yr) (Σ of CDF)		1.6×10^{-7}	

ʼ.5.3 Compartment II.01 (Turbine Building)

Compartment II.01 comprises much of the Turbine Building, ranging from the 565′ elevation to the 643′ elevation. The lower levels contain equipment such as the main feedwater pumps, the motor driven feedwater pump and the condensate pumps. Upper levels contain equipment such as the deaerator storage tanks and feedwater heaters. The compartment has essentially full-zone fire suppression coverage provided by wet pipe sprinkler systems, with the exception of the turbine bearings which have a manual suppression system. There are also a substantial amount of inspections and observations of this area by operators, security personnel, and general turbine building traffic.

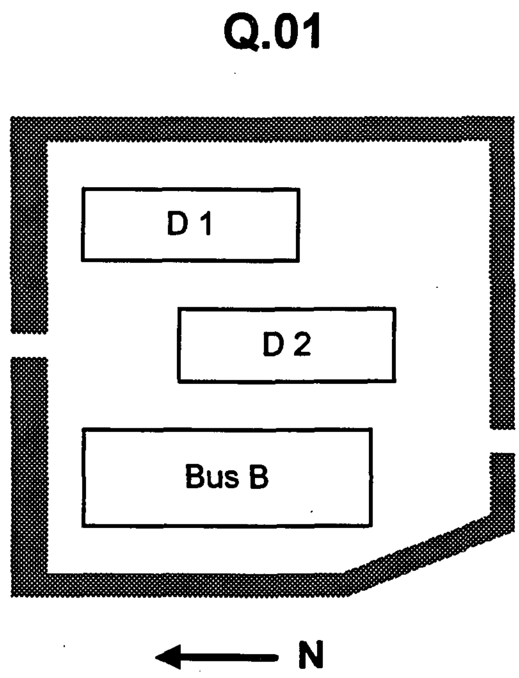
The proximity of the area suppression to fixed sources was verified by plant walkdowns. In addition to the turbine bearings, the only areas not directly covered are the main condenser and ancillary rooms. These areas were walked down and verified as not containing any cables or equipment important to safe plant shutdown. As with other compartments, a plant trip and loss of main feedwater were assumed to occur coincident with a fire in the Turbine Building.

It should be noted that given the layout of Davis-Besse, compartment II.01 is not one of the previously identified plant areas where a loss of offsite power may occur as a direct consequence of a fire. Major yard transformers which supply offsite power are outside and would not directly interact with fires postulated in II.01. Application of the auto suppression event tree yields the following:

Adjusted initiation frequency for fixed sources ($F'_{1,fixed}$)	1.5×10^{-2}/yr
Adjusted initiation frequency for transient sources ($F'_{1,transient}$)	1.2×10^{-4}/yr
Unavailability of wet-pipe suppression (AS)	0.02
Conditional probability of core damage given failure of all equipment in compartment (P_{2A})	3.7×10^{-4}
Conditional probability of core damage given failure of equipment no protected by suppression (P_{2B})	5.7×10^{-6}
Core-damage frequency for compartment II.01	2.3×10^{-7}/yr

As can be seen, the calculated value is below the screening criterion of 1E-6, and this compartment readily screens.

`.5.4 Compartment Q.01 (High Voltage Switchgear Room B)



As shown, High Voltage Switchgear Room B has a fairly simple geometry, and contains three major .ctrical buses, 13.8 kV B Bus, 4.16 kV bus D1, and 4.16 kV bus D2. Initially, the simplistic treatment of F1*P2, utilizing the previously discussed severity factors, indicated a CDF value of greater then 1E-5. To better estimate the CDF, the room was subdivided by analyzing the impact of a fire initiated within each major bus separately. Given the degree of separation between the buses, it is not expected that any fires could actually propagate from one bus to another. Accordingly, a careful walkdown was conducted of the area in close proximity to each bus to identify conduits and components which could be affected by bus fires. Guidance utilized to determine critical interaction distances were the same as previously discussed in Section 4.2.4.1. As noted previously in this report, and is generally the case at Davis-Besse, all cables in Q.01 are contained either in conduit or flat bottomed cable trays containing Kaowool.

Physically, each bus is subdivided into a number of cubicles which are largely self-contained and separated from one another by a double wall with an intervening air gap. As the main bus conductor does run through the respective cubicles, however, the entire bus frequency was combined and taken as the fire frequency to be multiplied by the P_2 value associated with the bus. With respect to the "bus P_2" value, all conduits and components which could be affected by a fire from any of the cubicles were assumed to be failed. This is conservative as some conduits would realistically only be affected by one or two cubicles (e.g., overhead conduits running perpendicular to the bus); this treatment does, however, remove uncertainties associated with modeling of fire propagation between cubicles of each respective bus.

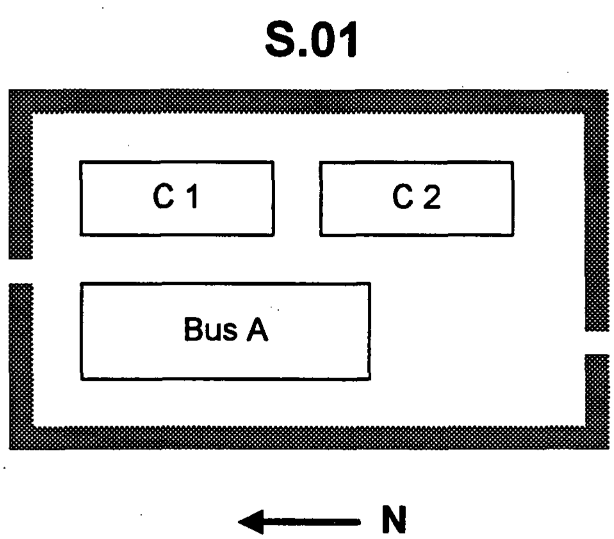
Results of this evaluation are:

	B Bus	Bus D1	Bus D2
Adjusted initiation frequency for fixed sources ($F'_{1,fixed}$) (/yr)	1.28 x 10^{-4}	1.0 x 10^{-4}	1.0 x 10^{-4}
Adjusted initiation frequency for transient sources ($F'_{1,transient}$) (/yr)	2.5 x 10^{-6}	2.3 x 10^{-6}	2.3 x 10^{-6}
Conditional probability of core damage given failure of all equipment affected by Bus	1.9 x 10^{-2}	2.0 x 10^{-2}	4.5 x 10^{-3}
Core-damage frequency (/yr)	2.5 x 10^{-6}	2.2 x 10^{-6}	4.8 x 10^{-7}
Composite core-damage frequency for compartment Q.01 (/yr) (Σ of CDF)		5.1 x 10^{-6}	

As can be seen, the composite CDF for this compartment is somewhat above the screening criteria. The relatively high conditional core-damage probabilities for these buses are the result primarily of the potential effect of a fire in this area on the availability of AFW. A fire in one of the buses could affect one of the two turbine-driven feedwater pumps, and would also cause power to be unavailable to the motor-driven feed pump. The eventual depletion of dc power due to the loss of charging capability from this train of ac power would also cause the PORV to be lost. Thus, a fire in the room could present a significant ıllenge to the ability to maintain feedwater to the steam generators, and could also affect the options for performing makeup/HPI cooling.

As noted previously, however, it is expected that a cubicle by cubicle fire modeling effort would lower the estimated CDF. Therefore, the above estimated value is considered to be bounding.

4.2.5.5 Compartment S.01 (High Voltage Switchgear Room A)

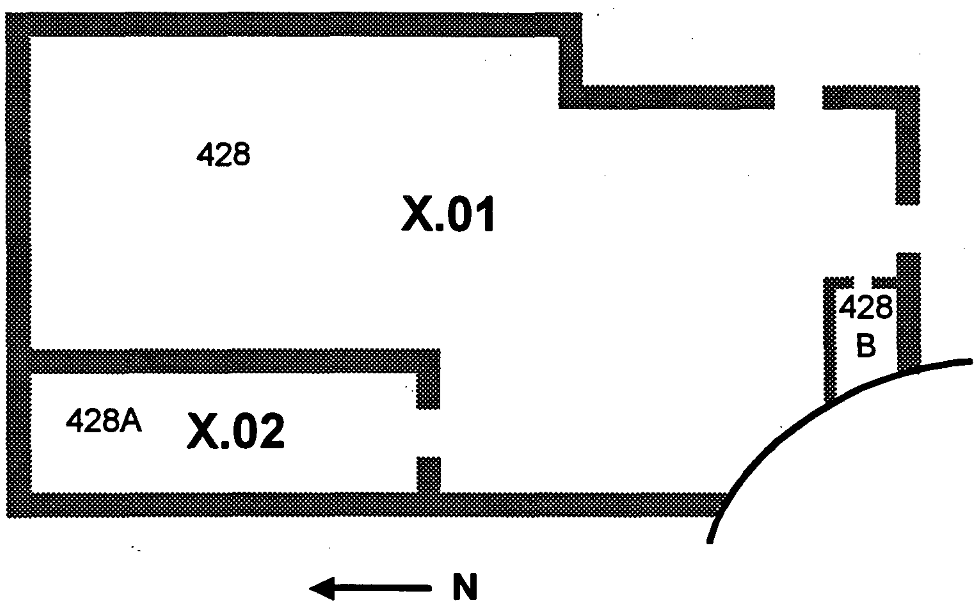


High Voltage Switchgear Room A also has a fairly simple geometry and is similar to compartment J1 discussed in the preceding section. The probabilistic treatment was also done in the same manner as for Q.01, with results:

	A Bus	Bus C1	Bus C2
Adjusted initiation frequency for fixed sources ($F'_{1,fixed}$) (/yr)	1.1×10^{-4}	1.0×10^{-4}	9.6×10^{-5}
Adjusted initiation frequency for transient sources ($F'_{1,transient}$) (/yr)	2.4×10^{-4}	2.3×10^{-6}	2.2×10^{-6}
Conditional probability of core damage given failure of all equipment affected by Bus	1.1×10^{-2}	8.9×10^{-4}	8.8×10^{-4}
Core damage frequency (/yr)	1.2×10^{-6}	9.4×10^{-8}	8.7×10^{-8}
Composite core-damage frequency for compartment S.01 (/yr) (Σ of CDF)		1.4×10^{-6}	

As can be seen, the core-damage frequency for this compartment was calculated to be slightly higher than the screening criterion. It is lower than that for analogous Room Q.01 primarily because a fire in compartment S.01 would not affect the availability of power to the motor-driven feed pump, nor would it directly affect operation of the PORV for makeup/HPI cooling. The PORV could be affected indirectly, however, because the fire could cause a loss of power to the PORV block valve. If the block valve were initially closed, it would not be possible to open it to allow use of the PORV.

4.2.5.6 Compartment X.01 (No. 1 Low Voltage Switchgear Room)



As shown, the No. 1 Low Voltage Switchgear Room has a fairly simple geometry. It contains, ~~h~~owever, a variety of important components and cables (in conduits). Based on the initial simplistic calculations of F1*P2 utilizing severity factors, this compartment represented the highest risk plant compartment. [It should be noted that X.02, the associated Station Battery Room is a separate fire compartment and was previously screened from further consideration as shown in Table 4.2.3.2]. To better estimate the screening CDF, the room was examined on a cabinet by cabinet basis, with the impact of a fire initiated within each calculated separately. A careful walkdown was conducted of the area in close proximity to each component to identify conduits and equipment which could be potentially affected. Guidance utilized to determine critical interaction distances were the same as previously discussed in Section 4.2.4.1. Consistent with guidance in Ref. 3, low energy cabinets (i.e., < 480 V) with qualified cable were assumed to have fires which were largely self-extinguishing and would not affect surrounding equipment. As a measure of conservatism, several components located in close proximity to a concentrated number of overhead conduits were assumed to fail all equipment in the compartment.

In addition to the above normal fire modeling, this compartment also contains four oil filled transformers which were investigated in special detail. The subject components are high voltage power transformers containing a silicone based di-electric fluid which cools the transformer internals via external, convectively cooled, radiators. Depending upon the flammability of the transformer fluid, transformer fires may pose a significant threat to targets located in the switchgear rooms. The fluid used in the subject transformers is Dow Corning 561, which is classified as "less flammable" (i.e. fire point > 572 deg F).

Following a significant effort (Ref. 28) to characterize the fire risk associated with these components, including a review of literature detailing research results and the experience base to-date, a position was formulated. Based on both the statistical and deterministic evidence presented, fires involving silicone transformer fluid are highly unlikely, whether induced by internal transformer faults or as a result of an external heat flux. In the event of an ignition, the fire would be confined and would self extinguish upon removal of the fire inducing energy source (e.g. electrical arcing). The only possible mechanism for damage beyond the confines of the transformer itself would be as a result of an explosion, possibly resulting in missile projection. As such, the following approach was adopted:

1. The nominal fire frequency derived for indoor transformers based on the FEDB (i.e., 8.4E-05 per transformer year for DB) was utilized. This is nearly a factor of 100 higher than the evidence gathered for silicone transformer fires suggests. This approach, however, will compensate for any minor fires which might have been missed by the references cited. For deterministic fire modeling of minor transformer fires, a heat release rate of 65 Btu/s (~65kW) was utilized, as suggested by Ref.3.

2. To account for the, albeit highly remote, possibility of a more energetic event involving silicone transformer fluid (e.g., explosion), a frequency of occurrence of 1.1E-06 per transformer yr was utilized. Under these conditions the entire contents of the switchgear room were assumed to be failed.

The results of the analyses for compartment X.01 are tabulated below.

Ignition Source	Adjusted Frequency (Transient Plus Fixed)	Conditional Probability of Core Damage	Core-Damage Frequency
Room 428A	1.61×10^{-3}/yr	3.19×10^{-4}	5.16×10^{-7}/yr
Room 428B	2.13×10^{-5}/yr	1.00×10^{-2}	2.14×10^{-7}/yr
Room 428 Room-wide sources (including catastrophic transformer fires)	7.02×10^{-5}/yr	7.26×10^{-2}	5.10×10^{-6}/yr
Transformers AF2 or BF2	5.09×10^{-5}/yr	6.92×10^{-4}	3.52×10^{-8}/yr
Transformers DF1-1 or DF1-2	5.09×10^{-5}/yr	5.52×10^{-4}	2.81×10^{-8}/yr
Panel BF3291	6.07×10^{-6}/yr	2.52×10^{-5}	1.53×10^{-10}/yr
Panel C4606	6.07×10^{-6}/yr	1.28×10^{-4}	7.76×10^{-10}/yr
Panel C4622B	6.07×10^{-6}/yr	3.54×10^{-4}	2.15×10^{-9}/yr
Bus DBC-2N	6.07×10^{-6}/yr	3.73×10^{-5}	2.27×10^{-10}/yr
Bus DBC-2P	6.07×10^{-6}/yr	4.39×10^{-4}	2.67×10^{-9}/yr
; DBC-2PN	6.07×10^{-6}/yr	1.58×10^{-5}	9.58×10^{-11}/yr
Bus F16A	1.54×10^{-5}/yr	7.57×10^{-5}	1.16×10^{-9}/yr
Bus F5	1.92×10^{-5}/yr	1.28×10^{-4}	2.45×10^{-9}/yr
Cabinet RC4605	1.54×10^{-5}/yr	6.74×10^{-5}	1.04×10^{-9}/yr
Cabinet RC4606	6.07×10^{-6}/yr	8.06×10^{-5}	4.90×10^{-10}/yr
Bus XY2	6.07×10^{-6}/yr	1.58×10^{-5}	9.58×10^{-11}/yr
Bus YB	6.07×10^{-6}/yr	1.38×10^{-5}	8.38×10^{-11}/yr
Bus YRF2	6.07×10^{-6}/yr	1.38×10^{-5}	8.38×10^{-11}/yr
Cabinet YV2	6.07×10^{-6}/yr	3.35×10^{-5}	2.03×10^{-11}/yr
Bus YVB	6.07×10^{-6}/yr	1.38×10^{-5}	8.38×10^{-11}/yr
Bus Y2A	6.07×10^{-6}/yr	3.15×10^{-5}	1.91×10^{-10}/yr
HVAC Equipment	2.61×10^{-5}/yr	1.13×10^{-4}	2.97×10^{-9}/yr
Total for Compartment X.01			5.90×10^{-6}/yr

Note that the results for this compartment are similar to those obtained for compartment Q.01. The contributions are similar as well. For fires that affect a significant portion of the compartment, there is a ›ng likelihood that one of the AFW pumps and the motor-driven feedwater pump will be rendered unavailable, as will the PORV. Thus, the options for providing feedwater to the steam generators are somewhat limited, and the makeup/HPI cooling can only succeed using a single makeup pump without the PORV. This is considered to be a successful mode of core cooling only for cases in which feedwater has succeeded at least temporarily, such that there is a reduced decay heat load for makeup/HPI cooling.

As noted previously, several components located in close proximity to a concentrated number of overhead conduits were assumed to fail all equipment in the compartment. As such, the above estimation of CDF is considered to be bounding.

4.2.6 Control Room (Compartment FF.01)

A fire in the Control Room could conceivably affect the availability of a large number of systems, especially if the bounding assumption were made that the fire caused the failure of all equipment in the room. It was clear that the Control Room would not be amenable to screening using either the qualitative or quantitative criteria from FIVE. A more detailed analysis of potential Control Room fires was therefore undertaken.

The general philosophy for fire evaluation of Control Room fires follows the approach suggested in NSAC 181 (Ref. 22) and the EPRI Fire PRA Implementation Guide (Ref. 23). It is similar to that adopted in other areas but differs in two respects:

1. Detailed fire propagation analysis will not be performed since there are no acceptable models for modeling propagation within and from cabinets. Instead, various assumptions will be made supported by the results of the Sandia cabinet fire tests in which all tested fires self-extinguished, and by the reports of control room fires in the Fire Events Data Base (FEDB, Ref. 1).
2. Regardless of the level of damage which is actually sustained as result of a fire, the production of smoke may necessitate the evacuation of the Control Room. Under such circumstances the operators will electrically isolate the Control Room as much as possible and shut down the plant using the appropriate procedures. Re-entry into the Control Room is credited for the operation of long term heat removal functions (providing associated controls are not damaged) which are not required for several hours.

Fires in the Control Room can affect the ability to actuate and control portions of plant systems, and may cause the spurious operation of some components, as is the case for other fire areas investigated. In addition, however, if it is not possible to suppress the fire in a timely manner, it may be necessary to evacuate the Control Room (e.g., when smoke obscures the control panels). If the Control Room must be evacuated, the options available to the operators for actuating and controlling some equipment would be reduced.

Fires considered in the Control Room were of two types: those that originated in a cabinet housing actuation or control elements associated with equipment important to maintaining core cooling, and those that originated elsewhere in the Control Room (including in cabinets that would not directly affect core cooling). Given that a fire originated within any particular cabinet in the Control Room, it was generally assumed that all of the equipment associated with actuation or control circuits within the cabinet would fail a consequence of the fire. Other fires were of concern only if they were not suppressed in time to prevent the need for evacuating the Control Room. All overhead cable in the Davis-Besse Control Room is enclosed in conduit and located well above the respective cabinets. The cables contained in the overhead conduits which just pass over cabinets without terminating in them were evaluated and were found to not be utilized in achieving safe shutdown following a fire. The manner in which the conditional probability of core damage following Control Room cabinet fires was calculated is described in the sections that follow.

4.2.6.1 Control Room Fire Hazard Review

The overall fire frequency in the Control Room was evaluated in the FIVE quantitative screening analysis as 9.5×10^{-3}/yr. The Control Room fire frequency is based on twelve fires which actually occurred in Control Rooms, eleven of which were cabinet fires and one was a kitchen fire. None of the fires were of significant severity and all were extinguished (or self extinguished) within a few minutes. No Control Room fires to date have required evacuation of the Control Room.

NSAC 181 indicates that the only significant Control Room fires are those which occur in cabinets and that transient fires do not pose a significant risk in the Control Room because it is continuously occupied and the likelihood that a transient fire would not be detected and suppressed in its incipient stage is very small. The Fire Events Data Base indicates that plant wide components are not applicable to the Control Room (Ref. 1). As such, the entire fire frequency was considered to be from Control Room cabinets.

In order to evaluate the conditional core damage frequency associated with each cabinet, it was necessary to distribute the total Control Room frequency throughout the cabinets. It was assumed that the density of potential ignition sources was approximately equal in all cabinets. Consequently, the frequency of a fire occurring in any one cabinet becomes proportional to the floor area of the cabinet. Each cabinet was measured and the frequency distributed. The results are shown in Table 4.2.6.1.1.

Table 4.2.6.1.1

Control Room Cabinet Ignition Frequencies

CABINET	DESCRIPTION	DIM (IN)	DIM (IN)	AREA (FT^2)	FRACTION OF AREA	FREQUENCY (YR^{-1})
C5702	CR LEFT CONSOLE	64	100	44.44	4.75E-02	4.51E-04
C5706	CR CENTER CONSOLE	64	100	44.44	4.75E-02	4.51E-04
C5711	CR RIGHT CONSOLE	64	100	44.44	4.75E-02	4.51E-04
C5715	CR STATION ELECTRIC DIST. PNL	30	90	18.75	2.00E-02	1.90E-04
C5716	CR ENG. SAFETY FEATURES CTRL PNL	30	72	15.00	1.60E-02	1.52E-04
C5717	CR ENG. SAFETY FEATURES CTRL PNL	30	50	10.42	1.11E-02	1.06E-04
C5718	CR REACTOR COOLANT CTRL PNL	30	60	12.50	1.34E-02	1.27E-04
C5719	CR REACTOR & STATION AUX CTRL PNL	30	50	10.42	1.11E-02	1.06E-04
C5720	CR REACTOR & STATION AUX CTRL PNL	30	100	20.83	2.23E-02	2.11E-04
C5721	CR FEEDWATER CTRL PNL	30	60	12.50	1.34E-02	1.27E-04
C5722	CR TURBO-GENERATOR CTRL PNL	30	110	22.92	2.45E-02	2.33E-04
C5723	SWITCHYARD CTRL PNL	24	86	14.33	1.53E-02	1.45E-04
C5725	SWITCHYARD ANNUNCIATOR CAB	16	36	4.00	4.27E-02	4.06E-05
C5727	ANNUNCIATOR CTRL CAB	16	30	3.33	3.56E-03	3.38E-05
C5729	MOTOROLA CTRL CONSOLE	8	20	1.11	1.19E-03	1.13E-05
C5729A	MOTOROLA CTRL CONSOLE	6	6	0.25	2.67E-04	2.54E-06
C5730	CTMT LIGHTING CONT. STATION	8	20	1.11	1.19E-03	1.13E-05
C5740	SBDG CTRL PNL	6	9	0.38	4.01E-04	3.81E-06
C5750A	CR GENERATOR & XFORMER RELAY PNL	60	86	35.83	3.83E-02	3.64E-04
C5750B	CR GENERATOR & XFORMER RELAY PNL	60	59	24.58	2.63E-02	2.50E-04
751	DIGITAL MULTIPLEXER #1	30	144	30.00	3.21E-02	3.04E-04
C5752	COMPUTER ANALOG I/O	30	144	30.00	3.21E-02	3.04E-04
C5753	PLANT COMPUTER MUX	30	144	30.00	3.21E-02	3.04E-04
C5754A-F	ANNUNCIATOR CAB	25	134	23.26	2.49E-02	2.36E-04
C5754G	345 kV MTR & LFC EQUIP	25	24	4.17	4.45E-03	4.23E-05
C5754H	345 kV METER CAB	25	24	4.17	4.45E-03	4.23E-05
C5754J	STARTUP TEST PANEL	25	26	4.51	4.82E-03	4.58E-05
C5754K	EPF MULTIPLEXER	32	21	4.67	4.99E-03	4.74E-05
C5755A	PAMS RACK CH2	31	24	5.17	5.52E-03	5.24E-05
C5755B	RAD MONITOR SYS CH2	31	24	5.17	5.52E-03	5.24E-05
C5755C-D	SFAS(C-D) CH.2	25	48	8.33	8.90E-03	8.46E-05
5755E-F	NI-RPS CH2	25	48	8.33	8.90E-03	8.46E-05
C5755G-J	PAMS RACK CH2	39	32	8.67	9.26E-03	8.80E-05
C5755K	SAFETY GRADE INSTR. CAB CH2	24	30	5.00	5.34E-03	5.07E-05
C5756A	RE 5328A,B,C	33	43	9.85	1.05E-02	1.00E-04
C5756C-D	SFAS(C-D) CH.4	24	48	8.00	8.55E-03	8.12E-05
C5756E-F	NI-RPS CH4	24	48	8.00	8.55E-03	8.12E-05
C5756G	PAMS RACK CH4	24	30	5.00	5.34E-03	5.07E-05

Table 4.2.6.1.1

Control Room Cabinet Ignition Frequencies (Continued)

CABINET	DESCRIPTION	DIM (IN)	DIM (IN)	AREA (FT^2)	FRACTION OF AREA	FREQUENCY (YR^{-1})
C5757A-D	MFPT ,TURBINE, EHC CAB	32	112	24.89	2.66E-02	2.53E-04
C5758A-F	MISC. CABINETS	30	154	32.08	3.43E-02	3.26E-04
C5759A	TEMP MONITOR PANEL	24	24	4.00	4.27E-03	4.06E-05
C5759B-F	NNI-X CABINET	24	120	20.00	2.14E-02	2.03E-04
C5760A-F	NNI-Y CABINET	24	170	28.33	3.03E-02	2.88E-04
C5761A	SFRCS ACTUATION CH.1, LOGIC CAB	24	24	4.00	4.27E-03	4.06E-05
C5761B-F	ICS CAB	24	120	20.00	2.14E-02	2.03E-04
C5762A	SFRCS ACT CH.1 RELAY/TERM	30	24	5.00	5.34E-03	5.07E-05
C5762B	RAD MONITORING SYS 1	30	24	5.00	5.34E-03	5.07E-05
C5762C-D	SFAS(C-D) CH.1	24	48	8.00	8.55E-03	8.12E-05
C5762E-F	NI-RPS CH1	24	48	8.00	8.55E-03	8.12E-05
C5762G	SAFETY GRADE INSTR. CAB. CH1	24	30	5.00	5.34E-03	5.07E-05
C5762N	SFRCS CH.1 MSIV/MFW VLV PNL	6	12	0.50	5.34E-04	5.07E-06
C5762Z	SFRCS ACTUATION CH.1, INTERFACE CAB	14	30	2.92	3.12E-03	2.96E-05
C5763A	PAMS PANEL	36	32	8.00	8.55E-03	8.12E-05
C5763B	PAM EQUIP RACK CH1	32	24	5.33	5.70E-03	5.41E-05
C5763C-D	SFAS(C-D) CH.3	24	48	8.00	8.55E-03	8.12E-05
C5763E-F	NI-RPS CH3	24	48	8.00	8.55E-03	8.12E-05
C5764A	SEISMIC CABINET	26	24	4.33	4.63E-03	4.40E-05
764B	VIBRATION MONITOR	26	24	4.33	4.63E-03	4.40E-05
C5764C	VIBRATION MONITOR	26	24	4.33	4.63E-03	4.40E-05
C5764D	ENVIRONMENTAL MONITOR	21	24	3.50	3.74E-03	3.55E-05
C5765A-F	RADIATION MONITOR CABINET CH.B	30	144	30.00	3.21E-02	3.04E-04
C5766	CONSOLE--KEYBOARD/MONITOR	6	6	0.25	2.67E-04	2.54E-06
C5770A-D	PLANT COMPUTER	36	100	25.00	2.67E-02	2.54E-04
C5771	COMPUTER PRINTER	30	50	10.42	1.11E-02	1.06E-04
C5772A-G	PLANT COMPUTER	24	170	28.33	3.03E-02	2.88E-04
C5774	CR ALARM TYPER ASSY	15	24	2.50	2.67E-03	2.54E-05
C5777A-B	COMPUTER RM EQUIPMENT	36	60	15.00	1.60E-02	1.52E-04
C5781	ABANDONED CABINET	0	0	0.00	0.00E+00	0.00E+00
C5783B	ABANDONED	0	0	0.00	0.00E+00	0.00E+00
C5783C	ABANDONED	0	0	0.00	0.00E+00	0.00E+00
C5783D	ABANDONED	0	0	0.00	0.00E+00	0.00E+00
C5784A	CH.1 ARTS	12	34	2.83	3.03E-03	2.88E-05
C5784B	CH.2 ARTS	12	34	2.83	3.03E-03	2.88E-05
C5784C	CH.3 ARTS	12	34	2.83	3.03E-03	2.88E-05
C5784D	CH.4 ARTS	12	34	2.83	3.03E-03	2.88E-05
C5792	SFRCS ACT CH2 RELAY TERM	30	48	10.00	1.07E-02	1.01E-04

Table 4.2.6.1.1

Control Room Cabinet Ignition Frequencies
(Continued)

CABINET	DESCRIPTION	DIM (IN)	DIM (IN)	AREA (FT^2)	FRACTION OF AREA	FREQUENCY (YR^{-1})
C5792A	SFRCS ACT CH2 LOGIC CAB	30	24	5.00	5.34E-03	5.07E-05
C5792N	SFRCS CH2 MSIV\FW PNL	6	12	0.50	5.34E-04	5.07E-06
C5792Z	SFRCS ACTUATION CH.2, INTERFACE CAB	12	28	2.33	2.49E-03	2.37E-05
C5796A	FIRE DETECTION PNL	8	26	1.44	1.54E-03	1.47E-05
C5796B	FIRE DETECTION PNL	8	26	1.44	1.54E-03	1.47E-05
C5798	CH.2 POST ACCIDENT INDICATING PNL	25	30	5.21	5.56E-03	5.29E-05
C5799	CH.1 POST ACCIDENT INDICATING PNL	25	30	5.21	5.56E-03	5.29E-05
RE5327	RAD MONITOR PANEL	32	43	9.56	1.02E-02	9.70E-05
RE5328	RAD MONITOR PANEL	32	43	9.56	1.02E-02	9.70E-05
DS4601	FIRE DETECTION PANEL	8	20	1.11	1.19E-03	1.13E-05
U500	COMMUNICATION PANEL	12	46	3.83	4.10E-03	3.89E-05
VAPR	ALARM PANEL	6	12	0.50	5.34E-04	5.07E-06
				935.98	1.00E+00	9.50E-03
				TOTAL AREA	TOTAL FRACTION	TOTAL FREQUENCY

?.6.2 Fire Propagation and Suppression Time

Three phases of cabinet fire development are considered, based on Sandia test results:

- Incipient (Pre-ignition) phase
- Pre-growth phase
- Pre-evacuation phase

Incipient Stage

The only ignition sources present within the electrical cabinets are associated with electrical faults. If the damage can be confined locally to the site of the overload, which is in fact the most likely situation given historical experience with Control Room fires (i.e., the faulted component or associated wiring), the resulting impact will be bounded by the random failure of the component itself, which has already been accounted for in the internal events PRA model.

The likelihood of detection and suppression is dependent upon whether or not the cabinets are fitted with in-cabinet detection. At Davis-Besse the vertical main control boards and annunciator panels are protected by in-cabinet smoke detectors placed on the cabinet ceilings. Sandia cabinet fire tests (pertinent data are summarized in Table 4.2.6.2.1) indicate a 5 minute time lapse between an in-cabinet fire detector detecting smoke and when actual flames were observed. The tests referred to utilized vertical and benchboard cabinets loaded with unqualified cables which were ignited using an electrical ignition source

(165W). No credit is taken for detection and suppression during this phase for fires in cabinets without in- inet detection.

Thus, despite the lack of physical separation of redundant components and wireways within the Control Room cabinets, the potential for significant damage is very small prior to flame ignition. Therefore, an initial five-minute time window for manual suppression was accounted for in modeling the risk from Control Room cabinet fires in cases where in-cabinet detection is provided. No significant damage is postulated within this time period. During this phase, ignition may be prevented by de-energizing the faulted component and / or using fire extinguishers located in the Control Room.

The equipment located within the Davis-Besse Control Room cabinets is separated from adjacent cabinets by double steel walls. Any cut-outs in the cabinet walls are sealed with silicone foam. In some cases the cabinets themselves may be subdivided into separate bays; however, there are no inter-bay physical barriers. Consistent with the Sandia cabinet fire tests, it was assumed that a fire in one of the Control Room cabinets will generally not impact equipment in another cabinet separated by double steel walls unless the target cabinet contained temperature sensitive electronics. The actual controls in the cabinets are not temperature sensitive and any temperature sensitive instruments do not control any equipment. Fires within the instrumentation cabinets were assumed to result in the loss of the entire cabinet's instrumentation. This assumption is consistent with the guidance provided in the Fire PRA Implementation Guide (Ref. 3) and with experimental evidence from tests conducted by Sandia (Ref. 21). All components served by the cabinet in which a fire originates during the incipient stage, however, are umed to fail, given the fire is not suppressed during the incipient phase.

Pre-Growth Phase

The second stage of fire development attempts to characterize the point at which a fire may be classified by Control Room operators as "significant", at which time they may begin to take actions in anticipation of having to leave the Control Room.

The evidence from the Sandia cabinet fire tests can be used to establish the time required for the progression to this stage. These tests indicate that between 8.5 and 11.5 minutes (i.e., about 10 minutes) elapsed between smoke first being observed coming from a cabinet and significant heat generation (10-20 kW, see Table 4.2.6.2.1). This is termed the pre-growth fire development phase. The tests also indicate that once fire growth begins it may progress rapidly, as may the rise in cabinet air temperature.

A 10-minute period for suppression initiation has been selected for cabinets fitted with in-cabinet detection. For other Control Room cabinets it was assumed that fire detection will be delayed for a further 3 minutes, and thus the time interval for suppression was taken to be 7 minutes.

Pre-Evacuation Phase

The Sandia cabinet fire tests indicate that fires were self-sustaining and did produce sufficient quantities of smoke to cause visual impairment with purge rates as high as 14 room changes per hour. All of the actual Control Room fires in the FEDB were small but this may have been because they were inguished early. Since there are no tools available for assessing smoke production and the evidence from the historical fires is not conclusive, it was assumed that any fire is capable of producing sufficient smoke to require evacuation of the Control Room given it is allowed to continue burning for a sufficient period of time.

There are eleven Sandia tests for which information is available for smoke build up: Six tests were performed in a small enclosure (11,016 ft^3) with ventilation rates of about 14 room changes per hour. However, only one of these was electrically initiated (PCT5) and indicated visual obscuration within 13 minutes (time zero is the point at which smoke was first observed from the cabinet). Five tests were performed in larger enclosures (48,000 ft^3), two of which were electrically initiated. In both electrically initiated, large enclosure tests the main control board was obscured within 15.5 and 19.5 minutes after smoke was first observed based on visual observations (see Table 4.2.6.2.1). The ventilation rate in one case was 1 room change per hour, and in the other, 8 room changes per hour. (For the large enclosure tests, the ventilation system did not appear to substantially affect the rate of smoke build up.)

The volume of the Davis-Besse Control Room envelope is approximately 59,000ft^3 (Ref. 24), about 20% larger than the large test enclosure used by Sandia. The normal number of air changes per hour is 2.25; however, this may be increased to 19.9 changes per hour in the full outside air mode (Ref. 25).

Based on the above discussion it was concluded that the rate of smoke build up in the Control Room will be marginally slower than observed for the large test enclosure. That is, it was judged that smoke obscuration of the control board will not occur for at least 18 minutes after the in-cabinet smoke detector alarms. Allowing an additional 3 minutes to activate one of the area ionization detectors, 15 minutes would be available to extinguish a fire in a cabinet with no in-cabinet detection, prior to the necessity for Control Room evacuation.

For comparison, ten minutes was selected by Sandia and used in the NUREG 1150 fire studies for Peach Bottom and Surry. EPRI selected a time interval of 15 minutes for their NSAC 181 Control Room analyses.

Probability of Fire Suppression

The time line for cabinet fire progression and detection is summarized below, based on the discussion provided above:

In-cabinet detector alarms	0 min
Ex-cabinet fire detection (smell, visual, area detection, spurious instrument reading)	3 min
Flame production	5 min
Significant fast growing fire	10 min
Control board obscured	18 min

The probability of non-suppression, based on the human cognitive reliability (HCR) model :ivation presented in the Fire PRA Implementation Guide (Ref. 3), is:

	In-cabinet detection	No in-cabinet detection
Probability of non-suppression prior to damage to cabinet	PNS(5 min) = .12	PNS(2 min) = 1.0
Probability of non-suppression prior to significant fire	PNS(10 min) = .016	PNS(7 min) = .049
Probability of non-suppression prior to Control Room obscured	PNS(18 min) = .0015	PNS(15min) =.0034

Table 4.2.6.2.1 Summary of Pertinent Data from Sandia Cabinet Fire Tests

EVENT	Test PCT 5	Test 24	Test 25
1. Smoke first observed coming from cabinet	10:00	10:30	9:30
2. Smoke detector gives alarm	N/A	N/A	10:00
3. Ignition	15:33	15:40	15:40
4. Significant flame spread	21:00	22:00	18:00
5. Main Control Room (MCR) view obscured	23:30	26:00	29:00
Time Interval			
1. Smoke being observed and ignition	5:33	5:10	6:10
2. Ignition and flame spread	5:27	6:20	2:20
3. Flame spread and MCR being obscured	2:30	4:00	11:00

4.2.6.3 Event Tree for Control Room Fires

An event tree was constructed to illustrate the general conditions that could affect the success or failure of core cooling following the initiation of a fire in any cabinet. This event tree is shown in Figure 4.2.6.1.

The first two top events in the event tree relate to the potential for the fire to be suppressed. The first considers whether the fire is suppressed very early, before actual ignition takes place and causes damage beyond the component in which the fault originated. The second considers whether the fire is able to progress sufficiently to create the need to evacuate the Control Room.

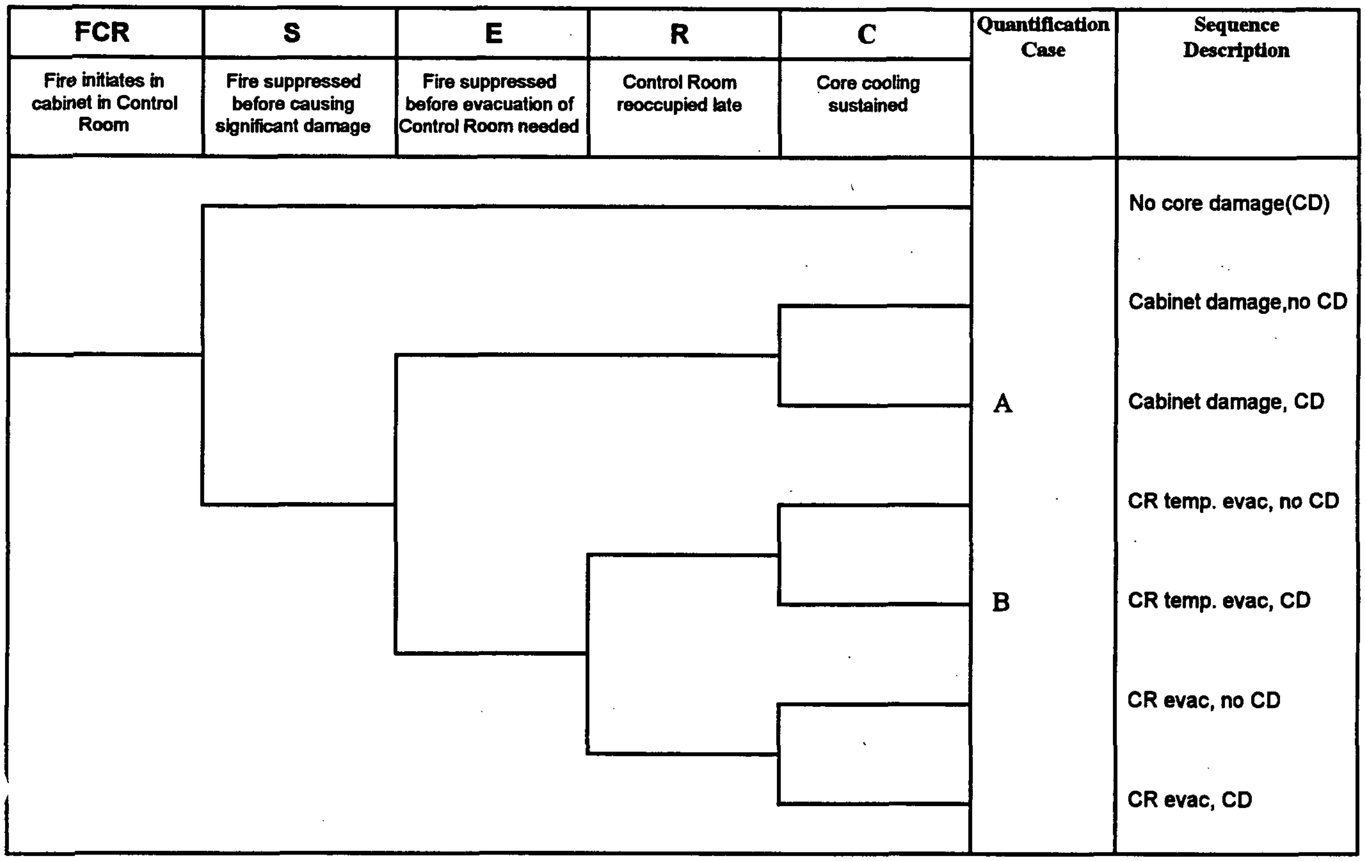


Figure 4.2.6.1 Event Tree for Control Room Fires

If the fire is suppressed before it is able to cause significant damage, no further consideration of the potential for core damage is necessary. If the fire causes significant damage within the cabinet, it is necessary to determine whether the fire is suppressed before the Control Room must be evacuated. If it is necessary to evacuate the Control Room, the next event accounts for whether the Control Room is reoccupied. After evacuation, measures to suppress the fire would continue until the fire was extinguished. Although it is not possible to pinpoint the time at which the Control Room would once again be usable, it was assumed that the operators would reoccupy it within no more than a few hours (nominally 1 to 3 hours) after the evacuation. This event was included for completeness, but it was assumed that there was a negligible probability that the Control Room could not be occupied before long-term actions to preserve core cooling (those that would be relevant 12 to 24 hours after the Control Room was evacuated) would be needed. Note that it was assumed that the cabinet or panel affected by the fire would remain unavailable for the period of interest in the assessment of the potential for core damage, even if the Control Room were reoccupied.

The final event in this event tree considers the possibility that core damage could result from the . There are only two sequences for which an evaluation of the conditional probability of core damage is necessary. The first of these (designated sequence A) is for the case in which the impact of the fire is essentially limited to the cabinet in which it originated. The fire is suppressed before it is necessary for personnel to evacuate the Control Room. For this case, the estimation of the conditional core-damage probability is analogous to that for other fires compartments in the plant. Equipment associated with the cabinet is assumed to be unavailable, and this is reflected by setting basic events in the PRA model representing failure of this equipment to "true". The cut sets representing the seven core-damage scenarios summarized in Section 4.2.2.2. were then generated and evaluated, accounting for these failures.

For the second sequence (sequence B), the implications are somewhat different. In addition to the failures associated with the cabinet in which the fire originated, it would have been necessary to evacuate the Control Room, at least for some period. Thus, only actions in the Control Room that would be taken within the first few minutes after the fire started (and before the Control Room was evacuated) and those that would not need to be accomplished until some period of hours later would be possible. In the intervening period, only actions for which necessary indications and adequate control stations exist outside the Control Room would be considered to be possible. The response of the operators for fires that would cause evacuation of the Control Room is outlined in the next section.

4.2.6.4 Actions for Fires Requiring Control Room Evacuation

Actions called for in response to a fire in the Control Room serious enough to require evacuation are specified in Abnormal Procedure DB-OP-02519 (Ref. 26). The actions affecting plant systems that are to be taken prior to leaving the Control Room include:

- Tripping the main turbine;
- Tripping makeup pump 1, to preserve it for use when it can be electrically isolated from the Control Room;
- Closing the block valve for the PORV, to avoid the potential for a LOCA via a stuck-open PORV;
- Tripping all three source breakers to bus B (one of the consequences of which would be to remove power from RCPs 1-2 and 2-1); and
- Tripping the other two RCPs (1-1 and 2-2).

The shift supervisor, assistant shift supervisor, primary reactor operator, secondary reactor operator, safety equipment operator, and shift manager are then responsible for completing a series of actions outside the Control Room. These actions are detailed in separate attachments to the procedure.

The shift supervisor is expected to proceed to the auxiliary shutdown panel, where the local-remote control switches for the pressurizer heaters, governor valve for AFW pump 1, and valve SW1382 (the valve isolating service water supply to the suction of AFW pump 1) are to be set to local. The shift supervisor is then instructed to maintain hot standby conditions as systems become available, until adequate guidance is available to undertake a cooldown.

The assistant shift supervisor would normally be concerned with ensuring that essential power was ...ilable to train 1. Train 1 is equipped with enhanced electrical isolation capability, and would therefore be the preferred means of providing power to important plant systems. To do this, the assistant shift supervisor would first ensure that emergency diesel generator (EDG) 1-2 was shut down. The assistant shift supervisor would then ensure that EDG 1-1 was operating properly and supplying power to bus C1. Additional measures would then be taken to ensure adequate isolation of train 2 power. After completing these actions, the assistant shift supervisor would locally trip AFW pump 1-2, and would then ensure that pump 1-1 was providing flow to steam generator 1-1.

The primary side reactor operator would perform a variety of actions aimed at ensuring that spurious actuations did not affect AFW or cause other systems to initiate erroneously. He would then verify that makeup was available, and would re-establish flow if necessary. The secondary side operator would take additional measures to ensure electrical isolation on train 2, and would then assist with the cooldown of the RCS by locally operating one of the atmospheric vent valves, in cooperation with the shift supervisor.

For cases involving evacuation of the Control Room, the system and sequence logic used to evaluate the conditional probability of core damage was modified to reflect these actions. For example, no credit was given to the use of AFW pump 1-2 for core cooling. The availability of ac power was also assumed to be limited to power supplied from EDG 1-1 to bus C1 (i.e., for train 1 loads). The conditional probabilities of core damage were then calculated on a case-by-case basis for the Control Room cabinets, taking into ...ount the additional failures that could result from fires in the cabinets.

4.2.6.5 Frequency of Core Damage for Control Room Fires

The two outcomes corresponding to core damage in the event tree for Control Room fires (Figure 4.2.6.2) were evaluated for each relevant cabinet in the Control Room. For many of the cabinets, a fire would not present a unique challenge to the ability to maintain core cooling. A single set of assessments was made to cover all of these cabinets because no safe shutdown equipment is directly affected.

Cabinet C5702. Cabinet C5702 comprises the left console of the benchboard panels. Among other controls, it contains those for the makeup system and for the pilot operated relief valve (PORV) and PORV block valve. A fire in the cabinet would not directly affect the availability of auxiliary feedwater, but it would prevent the operators from implementing makeup/HPI cooling from the Control Room. A fire in this cabinet could also cause a hot short that could lead to spurious opening of the PORV. Since the controls for the PORV block valve are located in the same cabinet, the fire could also disable the means for isolating the open PORV except from the motor control center for the block valve.

The benchboard panels are not equipped with internal smoke detectors, so that detection would occur most probably by the operators stationed at the panels. Based on the assessment summarized above, the probability of failure to suppress the fire before it caused damage within the cabinet is taken to be unity. The probability of failure to suppress the fire before there would be a need to evacuate the Control ...oom is estimated to be 0.0034 (as discussed in Section 4.2.6.2). The core-damage frequencies for the two ...es are therefore as summarized below.

Core-Damage Results for Fire in Cabinet C5702 (Initiating Frequency = 4.5 x 10^{-4})			
Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
A (Control Room not evacuated)	0.997	2.3 x 10^{-4}	1.0 x 10^{-7}
B (Control Room evacuated)	0.0034	1.0 x 10^{-1}	1.6 x 10^{-7}
Total for cabinet			2.6 x 10^{-7}

The conditional probability of core damage for case A, in which the Control Room remains occupied throughout the response to the fire, is dominated by the potential for a stuck-open PORV with failure to achieve long-term cooling. The PORV could stick open due to a hot short in its manual control switch within the cabinet in which the fire originated. Although the fire would not preclude the use of high pressure injection to preserve RCS inventory following the sticking open of the valve, the ability to effect recirculation from the emergency sump after the BWST was depleted could be impeded due to the fire. Consideration was given to de-energizing the control circuit for the PORV (allowing it to reclose); closing the PORV block valve from its motor control center; and establishing long-term recirculation from the emergency sump by manually operating the appropriate valves.

For case B, the fire would not be suppressed in time to prevent obscuration of the control boards by .oke. In this case, the dominant contributor to the conditional probability of core damage would result from a total loss of feedwater with failure of makeup/HPI cooling. Procedural actions in this case would essentially place reliance for core cooling on a single train of AFW. The operators would be expected to actuate AFW prior to evacuating the Control Room (if it did not actuate automatically). Actions taken outside the Control Room would then entail taking manual control of AFW pump 1-1 and securing AFW pump 1-2 (to guard against potential overfeeding of the steam generators). In addition, the faults associated with the panel in which the fire originated could preclude the use of makeup/HPI cooling. There is also an important contribution from the potential for a small LOCA to result from a loss of seal cooling (if the RCPs are not tripped prior to evacuating the Control Room) or from spurious opening of the PORV, as described above.

Cabinet C5706. Cabinet C5706 is the center console of the benchboard panels. This panel contains the controls for the AFW system and the motor-driven feed pump, and controls whose failure could affect the availability of main feedwater and the startup feedwater pump. Use of at least one of the turbine-driven AFW pumps would be possible from the auxiliary shutdown panel. The ability to use makeup/HPI cooling as an alternative to cooling via the steam generators would not be directly affected, provided it was not necessary to evacuate the Control Room.

As in the previous case, no credit is given for detecting and suppressing the fire before there might be significant damage within the cabinet. The probability of failure to suppress the fire before there would he a need to evacuate the Control Room would again be 0.0034. The core-damage frequencies for the two es are therefore as summarized below.

Core-Damage Results for Fire in Cabinet C5706 (Initiating Frequency = 4.5×10^{-4})			
Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
A (Control Room not evacuated)	0.997	1.7×10^{-3}	7.7×10^{-7}
B (Control Room evacuated)	0.0034	1.6×10^{-1}	2.4×10^{-7}
Total for cabinet			1.0×10^{-6}

For case A, the probability of core damage is dominated by sequences involving a total loss of feedwater caused by the fire and essentially independent failure of makeup/HPI cooling. The potential for the operators to make use of one train of AFW using controls outside the Control Room was also taken into account. For case B, the conditional core-damage probability is again dominated by total loss of feedwater with failure of makeup/HPI cooling. Credit for the latter mode of core cooling is severely limited, since the PORV cannot readily be operated from outside the Control Room. Makeup/HPI cooling could succeed only for instances in which the decay heat load was reduced because the loss of the remaining train of AFW was delayed. In that case, the PORV would no longer be needed. The conditional probability of core mage is higher in this case than in the previous case primarily because the fire could prevent proper initiation of AFW from within the Control Room. The operators would need to establish a steam supply to the turbine for AFW pump 1-1 before the steam generators dried out; otherwise, AFW flow could not readily be established from outside the Control Room.

Cabinet C5711. Cabinet C5711 is the right console of the benchboard panels. This panel contains controls for the main feedwater and condensate systems. A fire in the panel could also make the startup feed pump unavailable, but would not directly affect the AFW system or other means for core cooling.

As for the other two benchboard panels, no credit is given to detecting and suppressing the fire before there might be significant damage within the cabinet. The probability of failure to suppress the fire before there would be a need to evacuate the Control Room would again be 0.0034. The core-damage frequencies for the two cases are therefore as summarized below.

Core-Damage Results for Fire in Cabinet C5711 (Initiating Frequency = 4.5×10^{-4})			
Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
A (Control Room not evacuated)	0.997	5.7×10^{-6}	2.5×10^{-9}
B (Control Room evacuated)	0.0034	7.9×10^{-2}	1.2×10^{-7}
Total for cabinet			1.2×10^{-7}

For case A, the conditional probability of core damage is small because of the limited impact of the potential fire effects. For case B, the conditional core-damage probability is again dominated by total loss of feedwater with failure of makeup/HPI cooling. Actions taken following evacuation of the Control Room would once again place primary reliance for core cooling on a single train of AFW, although initiation of the AFW systems prior to evacuating the Control Room should be possible.

Cabinet C5715. Cabinet C5715 is the station electrical distribution panel, which is one of the vertical control panels. A fire in this panel could result in station blackout conditions; if the cabinet were substantially damaged, it would be necessary for the operators to isolate loads and initiate and control EDG 1-1 locally.

This cabinet has an internal smoke detector, so there is the potential for detecting and suppressing the fire before significant damage is sustained. The probability of non-suppression at this early juncture is estimated to be 0.12. The probability of failure to suppress the fire before there would be a need to evacuate the Control Room for this case would be 0.0015 (see Section 4.2.6.2). The core-damage frequencies for the two cases are therefore as summarized below.

Core-Damage Results for Fire in Cabinet C5715 (Initiating Frequency = 1.9×10^{-4})			
Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
A (Control Room not evacuated)	0.12	3.1×10^{-3}	6.9×10^{-8}
B (Control Room evacuated)	0.0015	9.2×10^{-2}	2.6×10^{-8}
Total for cabinet			9.6×10^{-8}

For case A, the conditional probability of core damage is again dominated by the potential for a total loss of feedwater and failure of makeup/HPI cooling. Both turbine-driven AFW pumps would potentially be available, although an extended loss of ac power would create the need for controlling the pumps to prevent overfeeding the steam generators. One train of ac power could be restored by actions outside the

Control Room. The potential for a RCP seal LOCA is limited by the likelihood that, if power were available to the systems providing core cooling, it would also be unavailable to the RCPs.

For the case in which evacuation of the Control Room was necessary, the options for core cooling would once again be reduced as in previous cases. One train of AFW should be available, and ac power could be restored via the emergency diesel generator by operator action outside the Control Room.

Cabinet C5716. Cabinet C5716 is the second of the vertical control panels. It contains controls and indications associated with the engineered safety features actuation system (SFAS). A fire in this panel could affect the availability of a variety of safety-related systems, including HPI, LPI, and component cooling water (CCW) and service water. It would not directly affect core cooling via the AFW system.

This cabinet also has an internal smoke detector, so the probability of non-suppression before significant damage was caused was again taken to be 0.12. The probability of failure to suppress the fire before there would be a need to evacuate the Control Room for this case would again be 0.0015. The core-damage frequencies for the two cases are therefore as summarized below.

Core-Damage Results for Fire in Cabinet C5716 (Initiating Frequency = 1.5×10^{-4})

Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
A (Control Room not evacuated)	0.12	4.0×10^{-3}	7.2×10^{-8}
B (Control Room evacuated)	0.0015	1.6×10^{-1}	3.7×10^{-8}
Total for cabinet			1.1×10^{-7}

For case A, the conditional probability of core damage is dominated by the potential for loss of seal cooling and seal injection for the RCPs, followed by failure of HPI. These failures could result from a combination of spurious actuations and unavailabilities in the HPI, CCW and service water systems. The contribution from loss of all feedwater coupled with failure of makeup/HPI is smaller because a fire in this cabinet would not directly affect the AFW system.

If evacuation of the Control Room were necessary, the RCPs would have to be tripped before the evacuation to ensure that a seal LOCA was avoided. Approximately equal contributions to the conditional probability of core damage were calculated for a total loss of feedwater (due to the reliance on a single train of AFW) and a RCP seal LOCA with failure of HPI.

Cabinet C5717. Cabinet C5717 is also a vertical control panel that contains some of the controls and indications for the SFAS. Failures due to a fire in this cabinet could affect portions of the service water, CCW, and LPI systems. They could also affect the availability of cooling for the RCP seals.

This cabinet also has an internal smoke detector, so the probability of non-suppression before significant damage was caused was again taken to be 0.12. The probability of failure to suppress the fire before there would be a need to evacuate the Control Room for this case would again be 0.0015. The core-damage frequencies for the two cases are therefore as summarized below.

Core-Damage Results for Fire in Cabinet C5717 (Initiating Frequency = 1.1×10^{-4})			
Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
A (Control Room not evacuated)	0.12	4.0×10^{-3}	5.0×10^{-8}
B (Control Room evacuated)	0.0015	1.8×10^{-1}	2.9×10^{-8}
Total for cabinet			7.9×10^{-8}

For case A, the results are similar to those for cabinet C5716; the fire could cause a complete loss of cooling for the RCP seals, and if the RCPs were not tripped in a timely manner a seal LOCA could result. In this case, however, the predominant core-damage scenario would involve a failure of long-term cooling (e.g., via high pressure recirculation), rather than the failure of HPI. The contribution from the potential for a total loss of feedwater with failure of makeup/HPI cooling would still be small, but there would be a contribution from failure of long-term cooling after successful initiation of makeup/HPI cooling.

If evacuation of the Control Room were necessary, the RCPs would have to be tripped before the evacuation to ensure that a seal LOCA was avoided. The conditional probability of core damage is similar that for cabinet C5716, except that the dominant contributor shifts from the possibility of a RCP seal LOCA with failure of HPI to a seal LOCA with failure of high pressure recirculation.

Cabinet C5718. Cabinet C5718 is the fourth of the vertical control panels, housing controls for the reactor coolant system. Failures of the controls in this panel would not directly affect the availability of plant systems credited in the PRA models. It was assumed that a fire in this panel (as in all of the main control panels) would cause a plant trip and loss of main feedwater.

Like all of the vertical cabinets, cabinet C5718 also has an internal smoke detector, so the probability of non-suppression before significant damage was caused was again taken to be 0.12. The probability of failure to suppress the fire before there would be a need to evacuate the Control Room for this case would again be 0.0015. The core-damage frequencies for the two cases are therefore as summarized below.

Core-Damage Results for Fire in Cabinet C5718 (Initiating Frequency = 1.3 x 10^{-4})			
Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
A (Control Room not evacuated)	0.12	5.7 x 10^{-6}	8.5 x 10^{-11}
B (Control Room evacuated)	0.0015	7.9 x 10^{-2}	1.5 x 10^{-8}
Total for cabinet			1.5 x 10^{-8}

For case A, the conditional probability of core damage is small, since the systems needed for core cooling are largely unaffected. Coupled with the initiating frequency and the non-suppression probability for this type of panel, the contribution to core-damage frequency is negligible.

The conditional probability of core damage is much higher for cases in which the Control Room must be evacuated. This is due to the precautionary measures taken to protect vital systems, including relying on a single train of AFW and a single EDG. As in other cases involving Control Room evacuation, the largest contributor to the potential for core damage is due to the total loss of feedwater and failure of makeup/HPI cooling.

Cabinets C5719 and C5720. Cabinets C5719 and C5720 are the vertical control panels for reactor ciliaries. A fire in either cabinet could affect some non-vital support functions, including a portion of the instrument air system, CCW to non-essential loads, and the condenser circulating water system. Accounting for the potential for suppression (based on the presence of a smoke detector inside the cabinets), the core-damage frequencies are as summarized below.

Core-Damage Results for Fire in Cabinets C5719 and C5720 (Combined Initiating Frequency = 3.2 x 10^{-4})			
Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
A (Control Room not evacuated)	0.12	2.7 x 10^{-5}	1.0 x 10^{-9}
B (Control Room evacuated)	0.0015	7.9 x 10^{-2}	3.8 x 10^{-8}
Total for cabinets			3.9 x 10^{-8}

For case A, the conditional probability of core damage is again small, since the most of the systems needed for core cooling would not be affected by a fire in these cabinets. The conditional probability of core damage for cases in which the Control Room must be evacuated would be the same as that for cabinet C5718; the failures associated with the cabinets themselves would be inconsequential relative to the limited options available outside the Control Room.

Cabinets C5721 and C5722. Cabinets C5721 and C5722 are the vertical control panels for, pectively, the feedwater system and the main turbine and generator. A fire in either of these panels could lead to loss of main feedwater and of the startup feed pump as a backup to the AFW system. The AFW system itself and other systems needed for core cooling would be unaffected. Therefore, the same assessment of the potential for core damage as for cabinet C5718 would apply. The results are summarized below.

Core-Damage Results for Fire in Cabinets C5721 and C5722 (Combined Initiating Frequency = 3.6×10^{-4})			
Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
A (Control Room not evacuated)	0.12	5.7×10^{-6}	2.4×10^{-10}
B (Control Room evacuated)	0.0015	7.9×10^{-2}	4.3×10^{-8}
Total for cabinets			4.3×10^{-8}

Offsite Power Cabinets. There are several cabinets in which the only failure of consequence with respect to the continued availability of core cooling would be the loss of offsite power. These cabinets include the following:

- Cabinet C5723, the switchyard control console;
- Cabinets C5750A and C5750B, the main generator and transformer panels; and
- Cabinets C5754G and C5754H, the 345 kV metering panels.

None of these panels contains in-cabinet smoke detection, so the probability of failure to suppress the fire before evacuating the Control Room would be 0.0034. The core-damage frequency associated with a fire in these cabinets is as summarized below.

Core-Damage Results for Fire in Cabinets C5723, C5750A and B, and C5754G and H (Combined Initiating Frequency = 8.4×10^{-4})			
Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
A (Control Room not evacuated)	0.997	5.5×10^{-4}	5.4×10^{-7}
B (Control Room evacuated)	0.0034	7.9×10^{-2}	2.8×10^{-7}
Total for cabinet			8.2×10^{-7}

For case A, in which the Control Room remains occupied, the conditional probability of core mage is dominated by total loss of feedwater with failure of makeup/HPI cooling. This results in part from the potential for the fire-induced loss of offsite power to contribute to a station blackout. If the ntrol Room is evacuated, the operators are effectively instructed to trip offsite power. Therefore, for case B the core-damage scenarios are essentially identical to those for other cases in which the failures within the cabinets are not significant after the Control Room is evacuated.

Cabinet C5740. Cabinet C5740 is a small panel housing the controls needed to tie the station blackout diesel generator (SBODG) to bus D2. A fire in the panel would not directly affect any other systems, although it was assumed in the assessment of the potential for core damage that main feedwater would be lost.

This cabinet also has no internal smoke detector, so the probability of non-suppression before significant damage was caused was taken to be unity. The probability of failure to suppress the fire before there would be a need to evacuate the Control Room for this case would again be 0.0034. The core-damage frequencies for the two cases are therefore as summarized below.

Core-Damage Results for Fire in Cabinet C5740
(Initiating Frequency = 3.8×10^{-6})

Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
A (Control Room not evacuated)	0.997	2.6×10^{-5}	9.7×10^{-11}
B (Control Room evacuated)	0.0034	7.9×10^{-2}	1.0×10^{-9}
Total for cabinet			1.1×10^{-9}

For case A, the conditional probability of core damage is relatively small because the SBODG would not be especially important unless there was a concurrent loss of offsite power. The primary contribution is from scenarios involving loss of ac power that contribute to failure of all feedwater and failure of makeup/HPI cooling. For case B, in which the Control Room must be evacuated, the conditional probability of core damage would be the same as for cabinet C5718. The SBODG is not available for use outside the Control Room, so that the case reduces down to being the same as that for cabinet C5718.

SFAS Cabinets. Cubicles C and D of cabinets C5755, C5756, C5762 and C5763 behind the main control panels contain the logic for the SFAS. Fires in these cabinets could cause spurious actuation and/or unavailability of portions of the safety features systems. These cabinets do not contain internal smoke detectors, so it is assumed that the fire will cause failure of all of the modules within the cabinet. The probability of failure to suppress the fire before evacuation of the Control Room is therefore estimated to be 0.0034.

Cabinets C5755 and C5756 contain identical elements of the actuation channel 2 SFAS components, and cabinets C5762 and C5763 similarly contain actuation channel 1 components. The core-damage frequencies for the two sets of cabinets are therefore as summarized below.

Core-Damage Results for Fire in SFAS Cabinets			
Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
Cabinets C5755 and C5756 (combined frequency = 1.7×10^{-4})			
A (Control Room not evacuated)	0.997	3.0×10^{-4}	4.9×10^{-8}
B (Control Room evacuated)	0.0034	9.6×10^{-2}	5.4×10^{-8}
Cabinets C5762 and C5763 (combined frequency = 1.6×10^{-4})			
A (Control Room not evacuated)	0.997	4.3×10^{-4}	6.9×10^{-8}
B (Control Room evacuated)	0.0034	1.0×10^{-1}	5.5×10^{-8}
Total for SFAS cabinets			2.3×10^{-7}

For case A, for each set of cabinets, the conditional probability of core damage is dominated by the potential for a RCP seal cooling with failure of HPI. Failures induced by the fire could contribute to the loss of seal cooling, and could prevent the HPI system from operating properly.

In the event that the Control Room must be evacuated, the potential for a total loss of feedwater with failure of makeup/HPI cooling again predominates the core-damage probability, as was the case for most other cabinets in the Control Room.

NNI Cabinets. Cabinets C5759 and C5760 contain the instrumentation and control elements associated with non-nuclear instrumentation (NNI) channels X and Y, respectively. In addition to causing failure of the main feedwater system, failures in the NNIX cabinet could affect the PORV. Although manual control of the PORV would not be affected, it would be possible for a fire in this cabinet to prevent the valve from operating automatically, or to cause the valve to open spuriously. The failures in the NNIY cabinet would not directly affect any of the functions associated with preventing core damage. It was assumed that a fire in the cabinet could lead to a loss of main feedwater.

These cabinets do not contain internal smoke detectors, so it is assumed that the fire will cause failure of all of the modules within the cabinet. The probability of failure to suppress the fire before evacuation of the Control Room is therefore estimated to be 0.0034. The core-damage frequencies for the two sets of cabinets are therefore as summarized below.

Core-Damage Results for Fire in NNI Cabinets			
Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
Cabinet C5759, NNIX (frequency = 2.0×10^{-4})			
A (Control Room not evacuated)	0.997	3.0×10^{-4}	6.0×10^{-8}
B (Control Room evacuated)	0.0034	9.6×10^{-2}	6.6×10^{-8}
Cabinet C5760, NNIY (frequency = 2.9×10^{-4})			
A (Control Room not evacuated)	0.997	5.7×10^{-6}	1.6×10^{-9}
B (Control Room evacuated)	0.0034	7.9×10^{-2}	7.7×10^{-8}
Total for NNI cabinets			7.9×10^{-8}

For case A, for the NNIX cabinet, in which the Control Room remains occupied, the conditional probability of core damage is dominated by the potential for a stuck-open PORV, with failure of long-term cooling via high pressure recirculation. If the Control Room is evacuated, the reliance on one train of AFW results in the dominance of a total loss of feedwater with failure of makeup/HPI cooling. For the NNIY cabinet, the results are the same as for the other cabinets in which the only impact is the loss of main feedwater (i.e., cabinet C5718).

SFRCS Cabinets. Portions of cabinets C5761, C5762, and C5792 contain the modules for the steam/feedwater rupture control system (SFRCS). Failures in any of these cabinets would essentially cause unavailability of automatic actuation and control of one train of AFW. As for the other cabinets in the back of the Control Room, these do not contain internal smoke detectors, so it is assumed that the fire will cause failure of all of the modules within the cabinet. The probability of failure to suppress the fire before evacuation of the Control Room is therefore estimated to be 0.0034. The core-damage frequencies for the two sets of cabinets are therefore as summarized below.

Core-Damage Results for Fire in SFRCS Cabinets			
Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
<u>Cabinets C5761A and C5762A, N & Z (combined frequency = 1.3×10^{-4})</u>			
A (Control Room not evacuated)	0.997	1.2×10^{-4}	1.5×10^{-8}
B (Control Room evacuated)	0.0034	7.9×10^{-2}	3.4×10^{-8}
<u>Cabinets C5792 and C5792A, N & Z (combined frequency = 2.4×10^{-4})</u>			
A (Control Room not evacuated)	0.997	1.2×10^{-4}	2.2×10^{-8}
B (Control Room evacuated)	0.0034	7.9×10^{-2}	4.9×10^{-8}
Total for SFRCS cabinets			1.2×10^{-7}

For case A, for each set of cabinets, the conditional probability of core damage is dominated by the potential for a total loss of feedwater and failure of makeup/HPI cooling. The conditional probability of core damage is relatively low because only one train of AFW would be affected directly, and the capability for makeup/HPI cooling would not be affected.

In the event the Control Room must be evacuated, actions taken outside the Control Room would entially circumvent the failures in the SFRCS cabinets. This case would therefore be virtually identical to others involving loss of main feedwater but no other direct effects of the fire.

Other Control Room Cabinets. There are many more cabinets in the Control Room in which fires would not have a unique impact on the availability of core cooling. These can be divided into two categories: those in which a fire could conceivably affect the availability of main feedwater, and those in which a fire would have no direct effect on systems that could play a role in core cooling. For the former category, the conditional probabilities of core damage would be the same as those calculated for cabinet C5711 (as described above). For the second category, the probability of core damage in the event that the Control Room continued to be occupied would be negligible. If the fire developed to the point that the Control Room needed to be evacuated, the probability of core damage would be the same as for the first category.

None of these cabinets is equipped with smoke detectors, so it is assumed that the fire will cause failure of all of the modules within the cabinets. The probability of failure to suppress the fire before evacuation of the Control Room is estimated to be 0.0034. The frequencies of core damage for the remaining cabinets are as outlined below.

Core-Damage Results for Fire in Other Cabinets			
Case	Non-Supp. Probability	Conditional CDP	Core-Damage Frequency
Cabinets in which MFW could be affected (combined frequency = 5.0×10^{-4})			
A (Control Room not evacuated)	0.997	5.7×10^{-6}	2.8×10^{-9}
B (Control Room evacuated)	0.0034	7.9×10^{-2}	1.3×10^{-7}
Cabinets in which MFW would not be affected (combined frequency = 4.4×10^{-3})			
A (Control Room not evacuated)	0.997	negligible	–
B (Control Room evacuated)	0.0034	7.9×10^{-2}	1.2×10^{-6}
Total for all other cabinets			1.3×10^{-6}

The frequency of core damage for these additional cabinets results primarily from the possibility that evacuation of the Control Room might be necessary, resulting in more limited options for maintaining core cooling.

4.2.6.6 Control Room Summary

The total calculated bounding core-damage frequency for fires initiated in the Control Room was mated to be 4.3×10^{-6} per year. The dominant contributors to this frequency result from fires within individual cabinets that can limit the options available for core cooling. The most important single cabinet is the control console housing the controls for the AFW system.

Much of the frequency results from fires that create sufficient smoke such that evacuation of the Control Room could be necessary. This would further limit the options available to the operators. It should be noted, however, that the potential for a fire to propagate sufficiently to cause widespread failures within the Control Room was judged to be negligible. Separation and the presence of walls between cabinets would prevent the spread of fire from one cabinet to adjacent ones. For most cabinets, it was assumed that, if a fire initiated, all of the components within the cabinet would be affected. The exceptions were cabinets in which there were internal smoke detectors. It was concluded that these detectors could afford the opportunity to detect and suppress a fire very early, before there was significant damage within the cabinet.

Even if the Control Room must be evacuated, options for core cooling remain available. Furthermore, it was judged that the Control Room could be reoccupied in the longer term, such that additional options could be implemented for long-term cooling. No specific vulnerabilities attributable to fires originating in the Control Room were identified.

4.3 Evaluation of Containment Fires

The FIVE methodology does not require an explicit analysis of the effects of fires in containment. This is because of:

1. a hot gas layer is unlikely to form in most areas of containment which can damage cables,
2. a large percentage of past fires were reactor coolant pump fires, which are less likely to occur in the future due to oil collection system design improvements,
3. there is a small number of events,
4. most of the containment fires occurred during plant shutdowns rather than during power operations, and,
5. previous fire PRAs did not show that containment fires are risk significant.

The FIVE method does require that at least a qualitative assessment should be performed in order to determine if containment needs to be analyzed in the more detailed manner described by FIVE for other plant compartments. For example, consideration should be given to conducting an analysis if 1) plant experience indicates that fires in containment during power operation have occurred on a recurring basis; and, 2) redundant trains of critical equipment within containment might be exposed to the same fire plume or be in a confined space and susceptible to damage by a hot gas layer.

In discussions with several senior personnel who have been involved with plant operations since .vis-Besse began power operation, no fires have occurred in containment during power operation. There has not been a containment fire alarm that has been attributable to smoke or other fire symptoms in the power history of the plant. The only recalled event was that once, during plant startup, some oil that was spilled on reactor coolant piping vaporized as the plant was heated up. This created a haze in containment and was investigated and identified rapidly. The vapor cleared and no further problems occurred.

The containment vessel is very large (free volume of 2.8E6 ft^3) and has adequate ventilation so that any hot gases would be distributed or collect in the upper regions of the vessel where no safe shutdown equipment is located. This limits the potential for redundant trains of critical equipment to be damaged by a single fire and its plume.

The FHAR analysis of this area states:

> In general, separation of safe shutdown and associated circuits of redundant trains is prevalent throughout most of containment. For the most part, Train 1 circuits enter containment from the west and Train 2 circuits enter primarily from the east. Consequently, Train 1 safe shutdown and associated circuits are basically confined to the west side of containment while Train 2 circuits are primarily found in the east side. The few exceptions stem from a few Train 1 safe shutdown circuits that were routed into the predominately Train 2 east penetration area.

The FHAR then goes on to explain why the equipment that does not meet the Appendix R aration criteria is acceptable.

In evaluating the ignition sources in containment, the only major sources during power operation cables and the reactor coolant pump (RCP) motors' lube oil. All cables inside containment are qualified and are routed in cable trays or conduit. The RCPs are provided with a seismically designed lube oil collection system to collect any leakage from the RCP motors, thereby reducing the potential for the oil to leak from one of the motors and start a fire. The oil collection system was installed in the construction phase of the plant and was improved during the sixth refueling outage. The containment has a relatively low fire loading of 12,202 BTU/ sq. ft. (< 20,000 BTU/sq. ft. is considered to be a low loading). This limits the potential for fire propagation.

Based on the above, it is concluded that the likelihood of a fire occurring in the containment, propagating throughout the containment, and damaging redundant equipment is negligible. Consequently, further analysis of the containment was deemed unnecessary and associated fire compartments D.01, D.02, D.03, and D.04 were qualitatively screened.

The potential for fires in containment to cause a loss of coolant accident due to failure of a high/low pressure interface was evaluated. The primary point of concern is the PORV. The line out of the pressurizer to the RCS quench tank normally has only one closed valve (the PORV) to provide isolation. Review of the PORV control circuit and associated cable routing concluded that no fires in the containment can cause a hot short that would induce a PORV opening. Consequently there is no potential for a LOCA via this path.

The letdown system also has the potential for providing a path for loss of coolant. The control ves for the system, however, are located outside containment, as are valves which could isolate the system. Therefore the potential for a LOCA due to a fire inside the containment does not exist.

There are several sample lines off the RCS and pressurizer that could allow flow to the RCS quench tank or the post accident sample system. There are redundant isolation valves in these lines which preclude the potential for a LOCA. The occurrence of multiple independent hot shorts would be required to open these paths, and the associated risk is considered to be negligible.

4.4 Assessment of Outliers for Fire Hazard

As seen in the previous discussion, four plant compartments (FF.01, Q.01, S.O1, and X.01) were identified which had calculated bounding core damage frequency (CDF) values above the screening criteria of 1E-6/yr. For each case the bounding CDF was found to be to lie between 1E-6/yr and 1E-5/yr. Given these results, the Severe Accident Issue Closure Guidelines (Ref. 29) were reviewed to ascertain the relative importance of these estimations. Section 4.4 and associated Table 1 of the Closure Guidelines indicate that for fire compartments which fall in this CDF range, the licensee should ensure that severe accident management guidelines will be in place with the emphasis on prevention/mitigation of core damage or vessel failure, and containment failure. In addition to the in-place emergency operating procedures which center on prevention of core damage, Davis-Besse has committed to having Severe Accident Management Guidelines in place by December 31, 1997 (Ref. 30). Therefore, as delineated in the reference 29 closure guidelines, no further actions are deemed necessary. However, to further reduce the plant risk to ɪtulated internal fire events, Davis-Besse will review of fire response procedures associated with plant areas which did not screen to ensure that specified actions are optimized with respect to maintaining the overall plant risk as low as reasonably achievable. In addition, a condition report has been initiated (PCAQR 96-1841) to track resolution of an issue pertaining to two small bottles of compressed combustible gas located in chemistry facilities within the Auxiliary Building identified during the seismic-fire walkdowns.

4.5 Assessment of Other Fire Issues

4.5.1 Purpose of Analysis

This analysis follows the format of the Fire Risk Scoping Study as defined by the FIVE methodology, as adapted from the Sandia National Laboratories report, Fire Risk Scoping Study: Current Perception of Unaddressed Fire Risk Issues (NUREG/CR-5088, Ref. 31) hereafter referred to as the "FRSS". The purpose of this analysis is to assess the adequacy of the manner in which the Davis-Besse Fire Protection Program has addressed the following generic issues, which are related to fire risk:

1. Potential seismic-fire interactions.
2. Fire barrier qualification issues.
3. Manual fire fighting effectiveness.
4. Total environment equipment survival.
5. Potential control systems interactions.

In so doing, the relative contribution of these issues to the overall fire risk can be assessed in subsequent phases of the (Fire) IPEEE Analysis.

4.5.2 General Methodology

The methodology followed in conducting this analysis was defined by the EPRI Fire Induced Vulnerability Evaluation (FIVE) process (Ref. 1), Section 7.0 and Attachment 10.5. The checklist-based process requires the evaluation of plant systems, procedures, and licensing bases to determine whether the above listed generic issues have been adequately addressed, in a manner consistent with current NRC regulatory guidance and the current industry knowledge base.

Background:

Under the NRC-sponsored Fire Protection Research Program, Sandia National Laboratories developed the FRSS. The objectives of this study were to:

1. Reassess certain fire risk scenarios, in light of the availability of enhanced fire event databases and improved fire modeling techniques.
2. Identify significant fire risk issues that may not have been addressed adequately (or at all) under earlier fire risk assessments, and to attempt to quantify the impact of these issues.
3. Review current regulatory criteria and guidance, and plant fire protection programs, to assess whether the identified risk scenarios are adequately enveloped by these programs.

The issues identified and addressed by the FRSS include six categories:

1. Potential seismic-fire interactions
2. Fire barrier qualification issues
3. Manual fire fighting effectiveness
4. Total environment equipment survival
5. Potential control systems interactions
6. Improved analytical codes

The above issues, which were not addressed by earlier internal fire probabilistic risk assessments (PRAs), are required to be assessed as an integral part of the Individual Plant Examination for External Events (IPEEE). A structured approach to addressing the first five of these issues is presented in the FIVE report. The FIVE report provides an overall methodology for addressing the "fire" portion of the IPEEE process; the FRSS issues are but one element of the IPEEE process.

The sixth FRSS issue, concerning analytical codes, does not require a plant-specific evaluation or response, as the use of current-day analytical codes (i.e., COMPBRN IIIe) is incorporated as an integral part of the FIVE (Phase II) methodology. Accordingly, this analysis is limited to a Davis-Besse specific assessment of only the first five issues.

4.5.3 Seismic-Fire Interactions

4.5.3.1 General

This issue involves three concerns:

1. The potential for seismically-induced fires
2. The potential for seismically-induced actuation of fire suppression systems
3. The potential for seismically-induced degradation (failure or rupture) of fire suppression systems

The above events have obvious implications on both postulated fire scenarios and potential for disruption of the safe shutdown capability.

'.3.2 Seismically-Induced Fires

This issue considers the potential leakage or rupture of flammable/combustible liquid or gas lines or tanks/containers during a seismic event, which could create fire hazards. The potential hazards to be addressed include:

1. Hydrogen piping
2. Diesel fuel oil piping, day tanks, and storage tanks
3. Turbine lubricating oil storage tank(s) and associated piping
4. Turbine generator (hydrogen envelope)
5. Hydrogen seal oil unit and associated piping and tanks
6. Hydrazine storage tanks and associated piping
7. Reactor coolant pump motor snubber fluid
8. Compressed gases in Chemistry Labs
9. Indoor transformer oil

Seismic evaluations were performed when required. Two cases were identified as being potential fire sources as a result of a seismic event. Specifically, two small flammable compressed gas bottles located in chemistry facilities in the Auxiliary Building were found to have inadequate seismic mounting. This ıdition has been noted in the plant's corrective action program (PCAQR 96-1481) and is being resolved.

The specific location of these and similar hazards are identified through the Fire Walkdown Phase of the IPEEE process, and the seismic ruggedness of each identified component was dispositioned as appropriate.

4.5.3.3 Seismic Actuation Of Fire Suppression Systems

This issue considers the potential for inadvertent actuation of suppression systems during a seismic event, and the resultant effects on safety/safe shutdown related components and systems. The effects of concern include both flooding and wetting effects caused by runoff/spray. Fixed fire suppression systems are located in areas containing safety/safe shutdown related equipment.

The effects of potential flooding resulting from suppression system actuation are enveloped by the Davis-Besse Fire Hazards Analysis Report, to the extent that these events are enveloped by the FHAR's response to NRC Branch Technical Position 9.5-1, Item D.1(i), "Floor Drains." This response discusses the review of NFPA 92M, "Waterproofing and Draining of Floors," and its conclusion that there is adequate capability to remove water from fire suppression activities to protect equipment which could result in adverse consequences. The results of this review were submitted to the NRC in a letter dated July 31, 1989 (Serial No. 1685, Ref. 32).

In addition, as stated by the FHAR, in response to BTP APCSB 9.5- 1, Item A.5, "Fire Suppression ;tems", "Protection for inadvertent actuation of sprinkler system is provided where the Section 4 of this document indicates that the component must be protected to ensure safe shutdown capability in the event of a fire and where review of the design documents indicates the component can not survive the water spray."

As noted in the FIVE methodology, an assessment of Davis-Besse against I&E Information Notice (IN) 83-41 (Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment) would acceptably address the issue of adverse operational effects caused by the failure or spurious actuation of fire suppression systems. Toledo Edison Regulatory Management System (TERMS) item A03925 addresses this IN. Intra-company memorandum (ICM) A83-2074D, dated August 29, 1983 (Ref. 33), performed a review of the IN. This review concluded that past suppression system actuations had no effects on safety-related equipment.

Consequently, this determination is considered to adequately envelope the issue of seismically-induced actuation of Davis-Besse fire suppression systems.

4.5.3.4 Seismic Degradation of Fire Suppression Systems

This issue addresses the seismic installation of suppression system piping and appurtenances, and the potential for seismically-induced mechanical failure of these systems. The issue is focused on the ~tential effects on the safe shutdown capability caused by suppression system equipment dislodged .ring a seismic event, and falling onto the subject equipment.

The location of fire suppression piping with respect to safe shutdown equipment, and the potential effects, from the perspective of possible impact of equipment falling onto safe shutdown components, is addressed under the Seismic Walkdown Phase of the IPEEE.

4.5.4 Fire Barrier Qualifications

4.5.4.1 General

This issue is primarily concerned with the installation and maintenance of fire barriers and fire barrier penetration seals, including electrical and mechanical seals, as well as fire doors and fire dampers.

4.5.4.2 Fire Barriers

The Davis-Besse Fire Protection Program provides a description of the fire protection equipment periodic surveillance. In accordance with the program, periodic surveillance of all fire barriers and penetration seals is conducted at least once per 18 months, and prior to declaring a penetration seal/fire barrier functional, following repairs or maintenance. The applicable procedure that controls impairments of fire barriers and penetration seals is DB-FP-00009 (Ref. 34), "Fire Protection Impairment and Fire Watch." It provides for implementation of appropriate measures in the event fire protection equipment is :lared inoperable.

While the FHAR Operating Specifications provides a description of the required surveillance quency, specific inspection methodology and acceptance criteria are provided by plant procedures (Ref. 35-42). DB-FP-04023 addresses barrier inspections. Surveillance of fire dampers is addressed by DB-FP-04024. Fire doors are addressed by DB-FP-04026, -04027, -04028, -04036. Special barriers (structural steel and fire barrier wraps) are addressed by DB-FP-04021 and -04022.

4.5.4.3 Fire Doors

The surveillance of fire doors is addressed under Section 4.5.4.2.

4.5.4.4 Penetration Seal Assemblies

4.5.4.4.1 Penetration Seal Inspection and Surveillance Program

The surveillance of fire barrier penetration seals is addressed under Section 4.5.4.2.

4.5.4.4.2 Evaluation and Implementation of Applicable NRC Information Notices

The FIVE methodology identifies three NRC I&E Information Notices which have specific applicability to fire barrier penetration seals:

1. 88-04, "Inadequate Qualification and Documentation of Fire Barrier Penetration Seals."
2. 88-04 Supplement 1, "Inadequate Qualification and Documentation of Fire Barrier Penetration Seals."
3. 88-56, Potential Problems With Silicone Foam Fire Barrier Penetration Seals."

NRC Information Notice 88-04 and Supplement 1 to the IN has been addressed in TERMS A04720 and closed by ICM NEP 88-08230, dated September 22, 1988 (Ref. 43). This memo responds to each of the items in the IN.

Information Notice 88-56. As Dow-Corning 3-6548 silicone foam is one of the principal DB penetration seal materials, this notice is applicable to DB. TERMS A08724 addresses this IN. This TERMS has been closed by ICM NEP 88-08422, dated December 21, 1988 (Ref. 44). It indicates that the issues associated with IN 88-56 have been recognized and addressed in the appropriate DB silicone foam installation, inspection, and in the penetration seal detail drawings.

4.5.4.5 Fire Dampers

4.5.4.5.1 Fire Damper Inspection and Maintenance Program

The surveillance of fire dampers is considered in conjunction with overall fire barrier and penetration seal surveillance, and is addressed under 3.2.

4.5.4.5.2 Evaluation and Implementation of Applicable NRC Information Notices

The FIVE methodology identifies two NRC I&E Information Notices which have specific applicability to fire dampers:

1. 83-69, "Improperly Installed Fire Dampers at Nuclear Power Plants."
2. 89-52, "Potential Fire Damper Operational Problems."

TERMS A03865 addresses IN 83-69. This TERMS was closed by NEP 90-08902, dated April 9, 1990 (Ref. 45). The review was based primarily on the fact that all but one Technical Specification fire damper has been replaced. A special installation procedure (MP 1405.08) was developed to ensure that the new dampers were installed properly. Openings not protected by dampers were evaluated as non-rated openings with evaluations in accordance with GL 86-10.

The principal issue associated with IN 89-52 is the inability of curtain-type fire dampers to close under air-flow conditions through the associated ductwork. TERMS A14514 was closed by ICM NEP 89-08246, dated August 22, 1989 (Ref. 46). The review was brief because all but one Technical Specification fire damper has been replaced by dampers certified by the vendor (through actual testing) to be able to close under the worst case air-flow conditions. The only Technical Specification damper not replaced is in a transfer grill that is not subject to an air-flow velocity sufficient to prevent the damper from close.

In summary, Toledo Edison Engineering practice is to apply appropriate consideration to the applicable HVAC fire damper procurement, installation, and operational criteria in the design, installation, and/or modification of HVAC fire dampers at DB.

4.5.5 Manual Fire Fighting Effectiveness

4.5.5.1 General

This issue is focused on the adequacy of training and preparedness of the plant fire brigade, and on the general orientation of appropriate plant personnel to fire response requirements. The objective of this issue is to determine the adequacy of the plant's manual fire suppression capability, and thereby determine the degree to which this capability should be credited in the IPEEE fire assessment.

4.5.5.2 Reporting Fires

4.5.5.2.1 Orientation of Plant Personnel to Portable Fire Extinguishers

As described in the Fire Protection Program, NG-DB-00302 (FPP) (Ref. 51), a program is in place to indoctrinate personnel, as appropriate, in the administrative procedures that implement the DB Fire Protection Program. DB-OP-02525 (Fire Emergency) (Ref. 47) is the procedure applicable to all plant personnel with respect to fire reporting. DB-OP-02525 indicates that an individual discovering a fire is to report the fire to the Control Room.

Orientation of plant personnel in the identification of the types of fire extinguishers is accomplished under the General Employee Training program.

4.5.5.2.2 Availability of Portable Extinguishers Throughout the Plant

Portable fire extinguishers are located throughout the Davis-Besse Nuclear Power Station. Extinguishers are visually inspected once per quarter for general plant areas (DB-FP-04016) (Ref. 48).

4.5.5.2.3 Plant Procedure for Reporting Fires

The reporting of fires is addressed by DB-OP-02525, with subsequent notification of the Fire Brigade.

4.5.5.2.4 Communication System to Allow Contact With the Control Room

Reporting of fires to the Control Room can be done by several means. The main method is with the plant PA system (Gai-Tronics), with the telephone, or a messenger as backup methods.

4.5.6 Fire Brigade

4.5.6.1 Size of Fire Brigade

As stipulated in DB-FP-00005 ("Fire Brigade") (Ref. 49), a fire brigade of at least five members (including the fire brigade leader) is maintained on site at all times.

4.5.6.2 Brigade Members Knowledgeable in Plant Systems and Operations

The fire brigade is totally made up of Operations personnel.

4.5.6.3 Annual Physical Examinations for Brigade Members

In accordance with DB-FP-00005, brigade members must satisfactorily complete an annual physical examination, including a respiratory examination.

4.5.6.4 Minimum Equipment Provided/Available to Fire Brigade

The following fire brigade equipment is stored on site:

1. Turnout gear, including coats, helmets, and boots.
2. SCBA apparatus, with a supply of spare bottles, and a recharging station.
3. Portable lanterns/flashlights.
4. Smoke ejectors with flexible ducts.
5. Portable fire extinguishers throughout the station.
6. Portable radios.

The equipment available to the fire brigade, therefore, is consistent with the FRSS criteria, and the equipment complement is verified periodically under Procedure DB-FP-04005 (Ref. 50).

4.5.6.5 Fire Brigade Training

4.5.6.5.1 Initial Classroom Instruction Program

The fire brigade classroom training program, as described in DB-FP-00005, provides the following elements, consistent with the FIVE evaluation methodology:

1. Indoctrination in the plant fire fighting plan and identification of individual responsibilities of the brigade members is provided, in accordance with the training program described in DB-FP-00005.
2. Identification of the fire hazards and associated types of fires that may occur in the plant.
3. Identification of the location of fire fighting equipment for each fire area, and familiarization with the layout of the plant, including access and egress routes.
4. The proper use of available fire fighting equipment, and the correct method of fighting each type of fire. The types of fires covered (should) include electrical fires, fires in cable trays, hydrogen fires, flammable liquids, waste/debris fires, etc.
5. The proper use of communication, lighting, ventilation, and emergency breathing equipment.
6. The proper method for fighting fires inside buildings.
7. Review of latest plant modifications and changes in pre-fire plans.

In summary, the DB fire brigade classroom training program is in compliance with the FIVE/FRSS criteria.

4.5.6.5.2 Practice

The fire brigade hands-on training program, as described in DB-FP-00005, provides the following elements, consistent with the FIVE evaluation methodology:

1. The proper method for fighting various types of fires of "similar magnitude, complexity, and difficulty as those which could occur in a nuclear power plant."
2. Experience in actual fire extinguishment and the use of emergency breathing apparatus under strenuous conditions.
3. The practice sessions are held at regular intervals, not to exceed one year, for each fire brigade member.

In summary, the DB fire brigade hands-on training program is in compliance with the FIVE/FRSS criteria.

4.5.6.5.3 Drills

The fire brigade drills, as described in DB-FP-00005, provides the following elements, consistent with the FIVE evaluation methodology:

1. Drills are performed in the plant so that the brigade can practice as a team. Brigade members practice as a shift/unit, by virtue of shift scheduling. Training is tracked by the training organization.
2. Fire drills are held, to the extent practicable, quarterly, but no less than four times per calendar year.
3. Each brigade shift participates in at least one unannounced drill per year.
4. At least one drill per year is performed on a backshift for each shift fire brigade.
5. Drills are preplanned to establish training objectives, and drills are critiqued to determine how well the training objectives have been met.
6. On a triennial basis, drills are critiqued by qualified individuals independent of the utility's staff. During the triennial audit, the drill(s) are likely to be unannounced.
7. Pre-fire plans have been developed for all plant areas.
8. Pre-fire plans are used in fire brigade training activities, consistent with the objectives of the Fire Brigade Program Description.
9. The equipment available to the fire brigade is consistent with the FRSS criteria.

4.5.6.5.4 Records

In accordance with DB-FP-00005, records of training of each fire brigade member are maintained "to assure that each member receives training in all parts of the training program."

4.5.7 Total Environment Equipment Survival

4.5.7.1 Potential Adverse Effects On Plant Equipment By Combustion Products

The FIVE/FRSS methodology does not provide criteria for assessment of the potential effects of non-thermal products of combustion on safety/safe shutdown related equipment. However, for the relatively short duration of the fire event and early recovery period, these effects are considered to be insignificant by FIVE.

4.5.7.2 Spurious Or Inadvertent Fire Suppression Activity

The potential effects of spurious/inadvertent suppression system actuation are enveloped by Section 4.5.3 of this analysis.

4.5.8 Operator Action Effectiveness

4.5.8.1 Post-Fire Safe Shutdown Procedures

Procedure DB-OP-02519, "Serious Control Room Fire Inside", provides operating instructions for a fire that renders the Control Room or cable spreading room inaccessible, or renders normal controls and indication in the Control Room unreliable. Procedure DB-OP-02501, "Serious Station Fire," addresses required shutdown functions for a fire that occurs in fire areas outside the Control Room or cable spreading room that results in damage to safe shutdown equipment or cables.

4.5.8.2 Operator Training in Post-Fire Safe Shutdown Procedures

Periodic operator training in post-fire shutdown procedures is conducted in accordance with Job Performance Measures for both licensed and non-licensed plant operators.

4.5.8.3 Operator Reentry Into Affected Fire Area: Respiratory Protection

The FHAR does not specifically address operator effectiveness in smoke-filled areas, but the following apply:

1. SCBA equipment is provided in the Control Room complex and at strategic locations throughout the plant.
2. Fixed, battery-backed emergency lighting units are installed along post-fire shutdown access/egress routes and at equipment operating stations.

4.5.9 Control Systems Interactions

A detailed review and assessment of the DB design against Appendix R Section III. G was performed. This analysis is contained in the FHAR. The methodology for doing this analysis was to first identify the safe shutdown systems at DB. Then it was determined which of these systems are needed for Hot Standby and which are needed for cold shutdown for a postulated fire. This list is contained in Appendix A, Safe Shutdown Components List, of the FHAR. Once this was done, the necessary components and circuits were identified and located in the individual fire areas. The circuits identified included those for power, control and instrumentation. The database of circuits and the raceways that they run through is contained in Appendix B of the FHAR.

The safe shutdown components and circuits were then reviewed for compliance with the specific separation criteria of Appendix R. The evaluation was made for each fire area to ensure that the safe shutdown functions can be performed for a fire in that fire area. The results of these evaluations are found in the area by area reviews in Section 4.6 of the FHAR.

For the Control Room (Area FF) and the cable spreading room (Area DD), loss of circuits due to a fire requires that plant operators shut down the plant from outside the Control Room. Procedures (e.g. DB-OP-02519) are in place to guide the shutdown. This involves operators performing various actions throughout the plant to isolate equipment via use of transfer switches and/or depowering components and

--positioning components. The Shift Supervisor reports to the auxiliary shutdown panel in Room 324 .ere he directs the shutdown activities.

Where circuits have been relied upon to power safe shutdown equipment outside the Control Room or cable spreading room, reviews were conducted to ensure that they would be free from fire damage.

In conclusion, the DB alternative shutdown features provide independent remote control and monitoring features. Therefore, the design of the DB alternative shutdown capabilities is generally immune to the effects of "control systems interactions" as defined within the scope of the FIVE methodology.

4.5.10 Conclusions

The results of the topical assessments performed under the FIVE Fire Risk Scoping Study indicate that the following FRSS issues have been adequately addressed by DB, and the applicable aspects of the DB Fire Protection Program therefore are in conformance with the intent of the FRSS guidelines with the two exceptions noted in section 4.5.3.2., as tabulated in Attachment 10.5 of the FIVE methodology:

1. Potential seismic-fire interactions.
2. Manual fire fighting effectiveness.
3. Total environment equipment survival.
4. Potential control systems interactions.

As previously note, corrective actions to resolve the two deficiencies in the potential seismic-fire interactions area are being developed in the plant corrective action program.

4.6 Verification and Walkdowns

The various verifications and walkdowns performed to ensure the accuracy of the internal fire analysis have been described separately in previous sections. While these will not be reiterated here, it should be noted that these actions were taken with the intent of ensuring the fidelity of the overall conclusion reached for each fire compartment. As such, the level of verifications and walkdowns were judged to be sufficient to ensure this goal was met. As noted earlier, all the site engineering staff contributing to this effort were located on-site, enabling ready access to all plant compartments and equipment except as limited by normal occupational and radiation safety limitations.

4.7 USI A-45, NUREG/CR-5088, and GI 57

The IPEEE Supplement to GL 88-20 requested assessment of the adequacy of the decay heat removal system in the context of external events as part of resolution of USI A-45, Shutdown Decay Heat Removal Requirements. As noted previously, no specific vulnerabilities have been identified as part of the internal fires analysis, and, as such, USI A-45 is considered resolved with respect to postulated internal fires. This is consistent with conclusions of the IPE study for internal events.

NUREG/CR-5088, Fire Risk Scoping Study, was specifically addressed in Section 4.5 of this report. As noted in summary in Section 4.5.10, Fire Risk Scoping Study issues have been adequately addressed. As such, this overall issue is considered to be resolved.

Regarding Generic Issue 57, Effects of Fire Protection System Actuation on Safety-Related Equipment, the IPEEE Generic Letter stated that additional NRC research was being conducted in parallel with the IPEEE, and that "A specifically tailored walkdown for potential fire vulnerabilities should enable the licensee to collect information related to GI 57. Licensees may propose corrective measures that could resolve some or all of the GI 57 concerns." While a walkdown specifically for GI 57 was not conducted, the multiple walkdowns which were conducted as part of the overall IPEEE effort should provide a basis for any additional future actions which may be required.

REFERENCES FOR PART 4

1. EPRI TR-100370, Fire-Induced Vulnerability Evaluation, April 1992.
2. 10 CFR 50, Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979.
3. EPRI TR-105928, Fire PRA Implementation Guide, December, 1995.
4. Fire Hazards Analysis Report, Davis-Besse Updated Safety Analysis Report.
5. Fire Protection General Floor Plan El. 545 & 555, Drawing A-221F.
6. Fire Protection General Floor Plan El. 565, Drawing A-222F.
7. Fire Protection General Floor Plan El. 585, Drawing A-223F.
8. Fire Protection General Floor Plan El. 603, Drawing A-224F.
9. Fire Protection General Floor Plan El. 623, Drawing A-225F.
10. Fire Protection General Floor Plan El. 643, Drawing A-226F.
11. Fire Protection General Roof Plan, Drawing A-227F.
12. NUREG/CR-5759, Risk Analysis of Highly Combustible Gas Storage, Supply, and Distribution Systems in PWR Plants, June 1993.

    Memorandum NPE-96-00471, Generic Letter 93-06, Research Results on Generic Safety Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas" (TERMS A17325), October 10, 1996.

14. Davis-Besse Configuration Equipment Summary (DBCES) site equipment database.
15. SETROUTE (Drawing E-200B, Electrical Circuit Schedule & Drawing E-300B, Electrical Raceway Schedule).
16. NUREG/CR-2258, Fire Risk Analysis for Nuclear Power Plants, September 1981.
17. Abnormal Procedure DB-OP-02501, Serious Station Fire.
18. Memorandum DSO-96-00057, Fire Brigade Response Time, V. J. Patton, Operations Fire Protection Advisor, to D. G. Kuhtenia, June 13, 1996.
19. Administrative Procedure DB-FP-00007, Control of Transient Combustibles.
20. Personal communication, Les Bowyer (Supervisor - Radwaste Operations) to D. G. Kuhtenia, March 1996.
21. NUREG/CR-4527, Vol 2, An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets, Part II: Room Effects Tests, November 1988.
22. NSAC 181, Fire PRA Requantification Studies, EPRI, March 1993.
23. EPRI TR-105928, Fire PRA Implementation Guide December, 1995.
24. Davis-Besse Updated Safety Analysis Report, Section 9.4, Air Conditioning, Heating, Cooling, and Ventilating Systems (9.4.1, Control Room).

    System Description for Control Room Normal Ventilation, SD-029A Rev. 3.

26. Abnormal Procedure DB-OP-02519, Serious Control Room Fire.

Barrier Function List, Drawing C-1594.

28. NUS (Scientech) Transformer evaluation, R&R-PG-96-705, December 1996.

29. NEI 91-04 Rev. 1, Severe Accident Issue Closure Guidelines, December 1994.

30. Toledo Edison Serial Number 2275, Implementation of Industry Policy on Severe Accident Management, February 22, 1995.

31. NUREG/CR- 5088, Fire Risk Scoping Study: Current Perception of Unaddressed Fire Risk Issues

32. TE to NRC letter dated July 31, 1989 (Serial No. 1685).

33. Memorandum A83-2074D dated August 29, 1983

34. Administrative Procedure DB-FP-00009, Fire Protection Impairment and Fire Watch

35. Periodic Test Procedure DB-FP-04021, 18 Month Appendix R Wrap Visual Inspection

36. Periodic Test Procedure DB-FP-04022, 18 Month Structural Steel Visual Inspection

37. Periodic Test Procedure DB-FP-04023, 18 Month Fire Rated Barrier Visual Inspection

38. Periodic Test Procedure DB-FP-04024, 18 Month Fire Damper Visual Inspection

39. Periodic Test Procedure DB-FP-04026, 24-Hour Fire Door Visual Inspection

40. Periodic Test Procedure DB-FP-04027, 7 Day Appendix A Door Visual Inspection

41. Periodic Test Procedure DB-FP-04028, Appendix A Fire Door Inspection

Periodic Test Procedure DB-FP-04036, Appendix R Fire Door 18 Month Inspection

43. Memorandum NEP 88-08230, dated September 22, 1988

44. Memorandum NEP 88-08422, dated December 21, 1988

45. Commitment Closeout Form NEP 90-08902, dated April 9, 1990

46. NEP 89-08246, dated August 22, 1989

47. Abnormal Procedure DB-OP-02525, Fire Emergency

48. Periodic Test Procedure DB-FP-04016, Fire Extinguisher Quarterly Inspection

49. Administrative Procedure DB-OP-00005, Fire Brigade

50. Periodic Test Procedure DB-FP-04005, Fire Brigade Equipment Quarterly

51. Administrative Procedure NG-DB-00302, DBNPS Fire Protection Program

PART 5
HIGH WINDS, FLOODS AND OTHER EXTERNAL PHENOMENA

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5.0 HIGH WINDS, FLOODS AND OTHER EXTERNAL PHENOMENA

5.1 Introduction

Supplement 4 to Generic Letter 88-20 (Ref. 1), requests that licensees assess plant vulnerability to severe damage from external hazards including high winds, external floods, transportation and nearby facility accidents. Other plant-unique external events known to the licensee (e.g., active volcanoes, lightning strikes) should also be included in the IPEEE. Table 5.1.1 presents the results of the screening of the other events which was conducted based on the recommendations in Section 2 of NUREG-1407 (Ref. 2). The screening determined that no plant-unique events with potential severe accident vulnerability exist for Davis-Besse. Therefore, this section of the IPEEE will consider high winds, floods, transportation, and nearby facility accidents.

5.2 Methodology

The progressive screening approach recommended in Section 5 of NUREG-1407 (Ref. 2), was used in this assessment. The first three steps are performed for each of the hazards:

1. Review plant specific hazard data and licensing basis
2. Identify significant changes since operating license issuance
3. Verify that plant/facilities design meets 1975 standard review plan (SRP) (Ref. 3) criteria.

   If the plant conforms with the 1975 SRP criteria it is judged that the contribution from that hazard to core damage frequency is less than 10^{-6} per year and the IPEEE screening criteria are met. If the 1975 SRP criteria are not met, one or more of the following optional steps are used to further evaluate the hazard.

4. Determine if the hazard frequency is acceptably low
5. Perform a bounding analysis
6. Perform a probabilistic risk analysis

Table 5.1.1 Screening of External Events for Davis-Besse

Event	Generic Basis	Specific Applicability for Davis-Besse
Lightning	The primary impact of lightning is the loss of offsite power which is included as part of the internal events in the IPE. For certain sites, past experience may indicate that lightning strikes are likely to cause more than a loss of offsite power. Therefore, consideration of lightning effects should be performed only for sites where lightning strikes are likely to cause more than a loss of offsite power.	Based on past operating experience at Davis-Besse lightning strikes are not expected to lead to any effect other than the loss of power. Therefore, lightning effects do not need to be considered further in the Davis-Besse IPEEE.
Severe Temperature Transients	The effects of severe temperature transients are usually limited to reducing the capability of the ultimate heat sink and the loss of offsite power. The capacity reduction of the ultimate heat sink is a slow process that allows plant operators sufficient time to take proper actions. The other potential impact on the plant, the loss of offsite power, is considered in the IPE. Therefore, the temperature transients need not be addressed in the IPEEE.	The Davis-Besse site is not subjected to temperature transients more severe than other nuclear power plant sites in the United States. Therefore the generic basis applies for screening this event.
Severe Weather Storms	Severe weather storms have caused several complete and partial losses of offsite power. The potential to affect the loss of offsite power is addressed in the IPE. Therefore, severe weather storms need not be examined in the IPEEE.	There has been no unusual experience with severe weather storms at the Davis-Besse site. Therefore, the generic screening for this event applies.

Table 5.1.1 Screening of External Events for Davis-Besse (continued)

Event	Generic Basis	Specific Applicability for Davis-Besse
External Fires	Potential effects on the plant could be the loss of offsite power, forced isolation of the plant ventilation, and possible Control Room evacuation. The effect of the loss of offsite power is addressed in the IPE. The other effects have been evaluated during the operating license review against sufficiently conservative criteria and need not be reassessed in the IPEEE.	The site is surrounded by marsh and the area in proximity of plant buildings is cleared to preclude the possibility of external fires damaging equipment or impacting Control Room operations. Therefore the generic basis is applicable to Davis-Besse.
Extraterrestria l Activity	The probability of a meteorite strike is very small and can be dismissed on the basis of low initiating frequency.	The generic basis is applicable to Davis-Besse.
Volcanic Activity	Sites in the vicinity of active volcanoes should assess volcanic activity as part of the IPEEE process.	There is no volcanic activity in the vicinity of the Davis-Besse site. The generic basis is applicable to Davis-Besse.
Mayfly Activity	None	Mayfly activity could lead to the loss of offsite power. The effect of the loss of offsite power is addressed in the IPE. Therefore, mayfly activity need not be examined in the IPEEE.

High Winds and Tornadoes

5.3.1 High Winds and Tornadoes Design Basis

The design basis for high winds and tornadoes are identified in Sections 3.3, 3.5 and 3.8.1.1.5 of the Davis-Besse USAR (Ref. 5). Additional information can be found in the FSAR (Ref. 4) Questions and Answers 3.3.1, 3.3.2, 3.3.3, 3.3.4, 3.3.5 and 3.8.1.

5.3.1.1 High Winds

The wind pressures used in the design of the station's structures were based on a 30 foot above the ground wind velocity of 90 mph, with a 100 year recurrence. However, wind loads did not control the design of seismic class I structures due to the low wind pressures in comparison with tornado loads.

5.3.1.2 Tornadoes

The following seismic class I structures are analyzed for tornado loading (Not coincident with a LOCA or earthquake). These structures are designed for the concurrent loads due to differential pressure, wind, and postulated tornado missiles.

1. Shield Building
2. Auxiliary Building
3. Intake Structure
4. Valve Rooms 1 and 2
5. Service Water Tunnel
6. Three electrical manholes

Tornado Wind Load

The tornado wind velocity of 300 mph is used uniformly on class I structures for conversion of tornado loads into forces. The following equation is used to compute the wind forces with shape factors of .8 and .5 for the windward and leeward side, respectively:

$$q = 0.002558\ V^2$$

Differential Pressure

The design pressure drop is assumed to be 3 psi in 3 seconds which is 100 percent greater than the greatest pressure drop reliably measured.

The differential pressure between the inside and the outside is assumed to be 3 psi d for all the class I structures except the Auxiliary Building and the Intake Structure. The lower design differential pressure is justified for the Intake Structure and the Auxiliary Building because they have been designed with ficient venting to keep the pressure drop within the 1.5 psid design limit.

Tornado Driven Missiles

The following missiles were assumed when analyzing the class I structures for tornado loading:

1. A 12 foot long piece of wood 8 inches diameter traveling end on at a speed of 250 mph.
2. A 4000 lb, automobile traveling through the air at 50 mph and not more than 25 feet above the ground.
3. A 10 foot long piece of 3 inch schedule 40 pipe traveling end on at a speed of 100 mph.

Additionally, Table 3.3-2 of the USAR provides a listing of 6 additional missiles, including the depth of missile penetration and minimum available concrete thickness. The missile penetrations are less than half of the thickness of the barriers.

5.3.2 Significant Changes Since Operating License was Issued

Walkdowns of plant structures and review of drawings identified no changes to the plant design which could affect the high winds design basis.

5.3.3 Conformance with Standard Review Plan

As part of the progressive screening approach, a direct comparison of the Davis-Besse licensing bases to the acceptance criteria of the 1975 Standard Review Plan (SRP) was performed. The following SRP sections relevant to high winds were reviewed:

- 2.3.1 Regional Climatology
- 2.3.2 Local Metrology
- 2.3.1 Wind Loadings
- 2.3.2 Tornado Loadings
- 2.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles
- 2.5.3 Barrier Design Procedures
- 3.5.1.4 Missiles Generated by Natural Phenomena
- 3.5.1.5 Site Proximity Missiles (Except Aircraft)

Review of the sections of the 1975 SRP listed above and the Davis-Besse design basis revealed that the Davis-Besse design basis may not meet certain criteria from the SRP. The plant design did not conform with the 1975 SRP criteria with respect to the design basis tornado parameters, tornado wind loading, rate of pressure drop, structures to be protected from externally generated missiles, missile velocities, and missile barriers. These differences are summarized below.

Design Basis Tornado Parameters

SRP Section 2.3.1 states that the tornado parameters should be based on Regulatory Guide 1.76 (Ref. 6). However, the criteria for the design basis Tornado at Davis-Besse were established by USAEC Reactor Technology Memorandum No. 1, *Tornado Considerations,* dated April 10, 1968. Table 5.3.3.1 summarizes the differences between the tornado parameters used by Davis-Besse and the parameters in Regulatory Guide 1.76.

Tornado Wind Load

The tornado wind load used for the Davis-Besse design was based on 300 mph which is less than the wind speed of 360 mph that is required by SRP Section 3.3.2 and Regulatory Guide 1.76.

SRP Section 3.3.2, *Tornado Loadings,* states that for transforming the tornado wind velocity into an effective pressure applied to exposed surfaces of structures the maximum tornado velocity should be used. Regulatory Guide 1.76 defines the maximum wind speed as the sum of the rotational speed component -d the translational speed component which is 360 mph for tornado region I plants.

Rate of Pressure Drop

A tornado generated rate of pressure drop of 1 psi per second was assumed in the Davis-Besse design which is less than the rate of pressure drop of 2 psi per second required by Regulatory Guide 1.76.

The tornado generated differential pressure used in the Davis-Besse design is consistent with the Regulatory Guide 1.78 value of 3 psid. However, the rate of pressure drop specified in Regulatory Guide 1.76 exceeds the rate of pressure drop used in the Davis-Besse analysis. The rate of pressure drop affects the total differential pressure drop for vented structures that are constructed based on a differential pressure less than the total pressure drop of 3.0 psi. At Davis-Besse differential pressure was assumed to be less than 1.5 psid for the Auxiliary Building and the Intake Structure based on the available venting area. Calculation VA03/B01-005, *Tornado Depressurization of the Auxiliary Building* (Ref. 9), evaluates the differential pressure drop in the Auxiliary Building for the design basis tornado. This calculation determines the maximum differential pressure to be 0.5 psid for assuming a total pressure drop of 3 psi and a 1 psi per second rate of pressure drop.

Missile Velocities

The velocity of the missiles used in the Davis-Besse analysis is less than the velocities listed in SRP Section 3.5.1.4.

Table 5.3.3.1 Design Basis Tornado Characteristics

Characteristic	Regulatory Guide 1.76	Davis-Besse
Maximum Wind Speed (mph)	360	360
Rotational Wind Speed (mph)	290	300
Maximum Translational Wind Speed (mph)	70	60
Minimum Translational Wind Speed (mph)	5	-
Radius of Maximum Rotational Speed (mph)	150	275
Pressure Drop (psi)	3	3
Rate of Pressure Drop (psi/sec)	2	1

Standard Review Plan Section 3.5.1.4, *Missiles Generated by Natural Phenomena,* provides a missile spectra and corresponding velocities for tornado generated missiles. The Davis-Besse analysis uses the same spectra of missiles; however, the velocities are less than the velocities listed in the Standard Review Plan.

Structures to be Protected from Externally Generated Missiles

The Borated Water Storage Tank is not protected from externally generated missiles contrary to the criteria in Branch Technical Position AAB 3-2, *Tornado Design Classification* (Ref. 13), which is referenced by SRP Section 3.5.1.4, *Missiles Generated by Natural Phenomena.*

Branch Technical Position AAB 3-2, *Tornado Design Classification,* requires that a source of water, to provide long term cooling for an extended time after a loss-of-coolant accident, be protected against tornadoes.

Missile Barriers

Concrete missile barriers provided at Davis-Besse do not all conform to the requirements for minimum thickness in SRP Section 3.5.3.

Standard Review Plan Section 3.5.3, *Barrier Design Procedures,* requires concrete thickness of at least twice the penetration thickness determined for an infinitely thick slab. Several barriers at Davis-Besse, luding barriers for the High Voltage Switchgear Rooms, Diesel Generator 1-2 Room, and the Emergency ..ock Enclosure, do not meet the SRP criteria. The minimum concrete thickness of these barriers is 12 inches. However, this thickness is not equal to twice the penetration depth of 8.22 inches for tornado driven missiles determined in calculation C-NSA-019.01-001 (Ref. 10).

5.3.4 Hazard Frequency / Bounding Analysis

Tornado Wind Load

FSAR Question and Answer 3.3.2 addresses the difference in loading for a 360 mph wind versus a 300 mph wind. According to the response to this question an analysis was performed for the Auxiliary Building exterior walls for the 360 mph wind versus 300 mph and the results were compared. It was found that the 360 mph wind does not control the design. The size of reinforcement used is far greater than what the 360 mph wind, in conjunction with other loads requires.

In addition to the capability of class I structures with a 360 mph wind discussed above, it can be shown that the probability of a tornado with wind speeds greater than 300 mph striking the site is significantly less than 10^{-6}. NOAA Technical Memorandum NWS NSSFC-8 (Ref. 7), correlates tornado wind speeds with intensity categories (F-scale) from F0 to F5. The intensity category of F5 has nominal wind speeds greater than 261 mph and therefore the design basis tornado would be in the F5 intensity category. Figure 14 of the NOAA Technical Memorandum is a contour map that shows the annual tornado hazard from F5 tornadoes. Based on this figure the annual probability of a tornado of intensity F5 is less n 10^{-8} for the location of the Davis-Besse Nuclear Power Station.

Rate of Pressure Drop

The tornado generated differential pressure was assumed to be less than 1.5 psid for the Auxiliary Building and the Intake Structure, which is less than the total pressure drop of 3 psid. The reduced differential pressure for these building was based on the available venting area. For the Auxiliary Building calculation VA03/B01-005 determined that the only significant tornado generated differential pressure would be between the combined volume of the Clean Waste Receiver Tank Rooms, the Detergent Waste Drain Tank Room, the Miscellaneous Waste Drain Tank Room and their adjoining rooms. The results of this calculation determined that the maximum differential pressure between these rooms and adjacent rooms would be .5 psi which is less than the design assumption of 1.5 psi. A calculation was not performed to determine the maximum differential pressure in the intake structure but based on the venting area, it was assumed that the differential pressure in this building would not exceed 1.5 psid. Increasing the rate of the pressure drop will result in a higher differential pressure in the affected rooms in the Auxiliary Building and in the intake structure.

For the Auxiliary Building, the differential pressure assuming a pressure drop rate of 2 psi per second can be conservatively estimated using the method and Volume to Area (V/A) curves in Bechtel Power Corporation, *Generic Study of Tornado Depressuration Effects on Plant Systems and Components* (Ref. 14). For the integrated volume of the Clean Waste Receiver Tank Rooms, the Detergent Waste Drain Tank ɔm and the Miscellaneous Waste Drain Tank Room the ratio of the compartment volume (V) and the vent flow area (A) is 4700. Applying the V/A curve for a Regulatory Guide 1.76 Region I tornado the maximum differential pressure between the combined volume of the Clean Waste Receiver Tank Rooms, the Detergent Waste Drain Tank Room and the Miscellaneous Waste Drain Tank Room and the adjacent rooms is 1.1 psid.

A similar approach can be taken for the Intake Structure which, based on the dimensions given in USAR Section 3.3.2.1, has a V/A ratio of 2134. Applying the V/A curve for a Regulatory Guide 1.76 Region I tornado the maximum differential pressure across the intake structure walls would be 0.37 psid.

In addition to the discussion above, it can be shown that the hazard frequency combined with the conditional core damage probability is less than the screening criteria of 10^{-6}. ANS-2.3-1983, *American National Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites* (Ref. 8), provides a table that correlates tornado wind speeds with maximum atmospheric pressure drop. A total pressure drop of 1.5 psid would correspond to a maximum wind speed of greater than 260 mph. A total pressure drop of 1.5 psid ensures the differential pressure is within the design basis of the Auxiliary Building and the Intake Structure regardless of the rate of pressure drop. Based on Figure 14 of NOAA Technical Memorandum NWS NSSFC-8 (Ref. 7) the annual probability of a tornado with wind speeds in excess of 260 mph is less than 10^{-8} for the location of the Davis-Besse Nuclear Power Station.

Tornado Driven Missile Velocity / Missile Barriers

Calculation C-NSA-019.01-001 (Ref. 10) was initiated to calculate the penetration depths for the missile velocities in SRP section 3.5.3. The revised worst case penetration depth, for a 3 inch schedule 40 pipe, is greater than one half the barrier thickness for several barriers including the High Voltage Switchgear Rooms, Diesel Generator 1-2 Room, and the Emergency Lock Enclosure. The required barrier thickness for these shields had previously been evaluated using the lower missile velocities in Bechtel calculations for the emergency lock enclosure and diesel generator missile protection.

The frequency of missile impact and the damage resulting from missile impact can be estimated using the results of a study by EPRI (Ref. 39). This study applied a probabilistic Monte Carlo simulation to predict the risk to a hypothetical nuclear power plant by tornado generated missiles. Using a tornado occurrence frequency by region, a spectrum of missile types, and a representative number of potential missiles, the EPRI study determined the following frequencies for tornado missile impact and damage at a Region I single unit plant.

$F^N = 1.23 \times 10^{-4}$ / year

$F^L = 1.96 \times 10^{-5}$ / year

$F^U = 3.95 \times 10^{-7}$ / year

Where:

F^N = The frequency of any tornado generated missile impacting the plant structures.

F^L = The frequency of a missile impacting with sufficient force to cause backscabbing if all plant structures have 6 inch concrete walls.

F^U = The frequency of a missile impacting with sufficient force to cause backscabbing given 24 inch containment barriers, 18 inch Auxiliary Building barriers, 12 inch tank enclosure barriers, and 12 inch intake structure barriers.

Because the safety structure strike frequency for a given tornado region is proportional to the tornado frequency, Calculation C-NSA-099.16-12 (Ref. 12) adjusted the EPRI data to reflect the tornado frequency for the Davis-Besse site. Based on Table 3-4 of the EPRI study (Ref. 39), the data analysis of region I tornadoes used a mean frequency of 2.3E-3 per year for tornadoes of Fujita scale 1 or greater. Therefore, the impact frequencies in the EPRI analysis is adjusted by the ratio of the Davis-Besse mean (6.4E-4 per year) to the region I mean (2.3E-3 per year). This adjustment will reduce the overall frequency of tornadoes at the Davis-Besse site while retaining the NRC region I intensity distribution. The adjusted frequencies are:

$F^N = 3.37 \times 10^{-5}$ / year

$F^L = 5.37 \times 10^{-6}$ / year

$F^U = 1.09 \times 10^{-7}$ / year

The damage frequency F^U represents a bounding number for Davis-Besse due to these factors:

1. The Davis-Besse shielding design criteria for tornado generated missiles (Ref. 40, Section 3.1.4) exceeds the barriers assumed for the second case used in the EPRI study (F^U).

2. The hypothetical plant assumed for the EPRI calculations has significantly more surface area on the safety structures than Davis-Besse, therefore the impact frequency F^N would be lower at Davis-Besse.

3. The frequency F^U represents missiles that cause backscabbing of concrete walls. However, backscabbing does not necessarily cause structural damage or loss of function of safety related components.

Therefore, the hazard frequency from tornado generated missiles striking the plant's safety structures is less than 10^{-7} per year. Based on the screening criteria of 10^{-6} in NUREG-1407 for the hazard frequency and the conditional core damage frequency, the hazard frequency is acceptably low and additional analysis is not required.

Structures to be Protected from Externally Generated Missiles

The BWST is not protected from tornado missiles because the BWST is not required for the safe ıtdown of the plant following a tornado. As discussed in USAR section 3.8.1.1.5 and FSAR Response 3.8.1., a simultaneous LOCA and tornado are not postulated to occur. However, a calculation (Ref. 12) was performed to determine the hazard frequency and the conditional core damage frequency for a tornado driven missile impact on the BWST.

The EPRI tornado risk analysis (Ref. 39) calculates a missile impact probability for the tank enclosure (Target 7) of a hypothetical plant in Regulatory Guide 1.76 tornado region I which can be used to calculate a missile impact probability for the Davis-Besse BWST.

Calculation C-NSA-099.16-12 (Ref. 12) adjusts the missile impact frequency from the EPRI study to account for the smaller size of the Davis-Besse tank and the lower frequency of tornadoes at the Davis-Besse site. The resulting frequency is 2.4×10^{-6} per year for a missile impact on the BWST. Since the BWST is not protected by missile barriers, the impact frequency will also be assumed to represent the frequency of a impact causing damage. This represents a bounding number for the frequency of damage to the BWST at Davis-Besse, because although the BWST is not protected by missile shields a missile impact will not necessarily cause damage that would prevent the BWST from performing its safety functions.

The conditional core damage was calculated using the CAFTA cut set editor and the final Individual Plant Examination cut sets. The tornado was assumed to cause a loss of offsite power and a reactor trip in addition to the damage to the BWST. Since the initiation frequency of the event was determined to be 2.4×10^{-6} per year, the initiation frequency of the loss of offsite power and the reactor trip were deleted from the ; cut sets. Additionally, since the purpose of the calculation was to determine the conditional core damage probability for the specific initiator of a tornado that causes damage to the BWST, the frequency of other initiating events was set to zero. The effect of the damage to the BWST was evaluated by revising the out of service or failure probabilities of the LPI, HPI, and Makeup pumps which all take suction from the BWST. Although the Containment Spray pumps also take suction from the BWST, these pumps were not included in the IPE cut sets for core damage probability. The appropriate basic events for the affected pumps were deleted, which is equivalent to revising the probability that these pumps will not function to 1.0. The core damage probability calculated using this method was 4.71×10^{-4}. This core damage probability represents the conditional core damage probability for the case where a tornado has already caused the loss of the BWST and a loss of offsite power. The core damage frequency for this event is the product of this conditional core damage probability and the initiating event frequency of 2.4×10^{-6} per year. This results in a core damage frequency of 1.1×10^{-9} per year, which is significantly less than the screening criterion of 10^{-6} per year in Ref. 2. Therefore, the hazard frequency is acceptably low and no further analysis is required.

5.4 Floods

5.4.1 External Flood Design Basis

The design basis for flooding is identified in section 2.4 of the USAR (Ref. 5).

5.4.1.1 Maximum Probable Lake Flooding

The maximum probable high water level condition at the site is 15.1 feet above the low water datum or a maximum high static water level of 583.7 feet (I.G.L.D.). For the maximum conditions to occur, a 9.3 foot wind tide must exist coincident with the 4.8 foot long-term high monthly mean lake level, and under conditions where the transverse seiche would be adding one foot of lake elevation. A maximum probable meteorological event (MPME) was used to determine the maximum rise in lake level due to wind tides. This event would have a maximum ENE wind at any one location of 100 miles per hour for a 10 minute period, and wind speed could exceed 70 miles per hour during the six hour period before and after the maximum wind speed.

Based on the ENE 100 mile per hour winds associated with the MPME, the maximum wave runup on the breakwall would be 6.6 feet above the maximum probable static level of 583.7 feet. This gives a maximum water run-up on the breakwall of 590.3 feet.

The stations ground floor elevation is 585 feet (I.G.L.D.) which will protect the station against the maximum probable static water level of 583.7 feet. On the North and East side of the station protection from wave run-up is provided by an earthen breakwall built up to 591.0 feet. This breakwater, the station location 3000 feet from the shoreline, and the elevated land along the shoreline will protect the station from wave action at the maximum probable wave level of 590.3 feet.

Penetrations which enter safety related buildings at or below 585 feet are provided with terproofing for flood protection. Two doors in the intake structure that are at 576.5 feet are watertight and designed for a ten foot head of water.

5.4.1.2 Probable Maximum Flood (PMF) on Rivers and Streams

The Toussaint River empties into Lake Erie about 1.25 miles southeast of the station. The Toussaint river has a maximum elevation of 670 feet, a drainage area of about 143 square miles and no dams. The lower six miles of the stream are much wider than the remainder and the level in this section is controlled by the level of Lake Erie.

The effect of the Toussaint water level from the Probable Maximum Precipitation (PMP) was evaluated based on a PMP of 23.9 inches over a 24 hour period for the 143 square mile drainage area. Assuming that none of the water is discharged to Lake Erie while the Toussaint is at normal water level at its mouth the high water level due to the PMF would be 579 feet. However the maximum Lake Erie static level is 583.7 feet and at this water level the lake water would extend more than five miles upstream from the station site. Therefore PMF water from the Toussaint would be dissipated to the lake prior to reaching the station site.

5.4.1.3 Site Drainage of Local Intense Precipitation

The analysis of flooding due to local intense precipitation was based on the probable maximum ıfall estimate of 24.5 inches for a 6 hour period over a ten square mile area.

Site runoff was analyzed by assuming that the main discharge pipe in the sewer system fails at the beginning of the rainfall and by ignoring the storage capability of the sewer system. Based on these assumptions, with 24.5 inches of runoff, theoretically, water could build up to 584.5 feet, but runoff water would overflow to the marshes, which are at an approximate elevation of 570.0 to 575.0 feet. Since all structures are protected against flooding up to 585.0 feet, all structures are protected for even the theoretical maximum runoff.

5.4.1.4 Roof Flooding Due to Local Intense Precipitation

The roof plumbing system is designed based on a continuous rainfall of 4.5 inches per hour for a 30 minute duration. Penetrations in the Auxiliary Building roof are protected by curbs with a minimum height of 18 inches. To prevent the maximum buildup of runoff water from overflowing the curbs, an auxiliary drain system, consisting of horizontal drain pipes, is provided. The horizontal drains are designed to drain the maximum probable rainfall should all the roof drains become stopped.

5.4.1.5 Ice Flooding

River flooding (Section 5.4.1.2) was analyzed assuming none of the water from the probable maximum rainfall reaches Lake Erie. Therefore, flooding due to ice jams in the Toussaint River will not :se river level to increase above the maximum probable level of 579 feet for the Toussaint River.

5.4.2 Significant Changes Since Operating License was Issued

5.4.2.1 Changes to Plant Design and Surrounding Landscape

Walkdown and review of drawings indicates that no changes to the plant design which could affect external flood design basis have occurred since issuance of the operating license. Additionally, no new features in the surrounding landscape have been identified that could affect flooding or flood control.

A water ingress path through flooded conduit was noted during the flooding walkdown; it was judged, however, that this condition has a negligible effect on the core damage frequency. As documented in PCAQR 95-0055, ground water entered a junction box and an MCC through conduits in a concrete duct bank that is embedded in the ground at about the 563 elevation. The duct bank is protected by a watertight membrane which appears to be leaking, allowing ground water into the duct bank. As described in USAR Section 2.4.2.2.3 similar waterproofing is used on other duct banks that penetrate safety related buildings at less than the 584 foot elevation. Although it is possible that at the design flooding conditions additional water ingress could occur, it was judged in the resolution of PCAQR 95-0055 that significant flooding does not appear to be a possibility due to the limited amount of moisture accumulation.

5.4.2.2 Revised Probable Maximum Precipitation (PMP) Criteria

The latest PMP criteria published by the National Weather Service call for higher rainfall intensities over shorter time intervals and smaller areas than previously considered. The new PMP values are documented in the National Oceanic and Atmospheric Administration (NOAA) Hydrometeorological Report No. 51 (Ref. 15), Report No.52 (Ref. 16) and Report No. 53 (Ref. 17). Based on these reports for the area including Davis-Besse the probable maximum rainfall for a one square mile area over a five minute period is 5.9 inches, the probable maximum rainfall for a one square mile area over a fifteen minute period is 9.3 inches, the probable maximum rainfall for a one square mile area over a thirty minute period is 13.3 inches, the probable maximum rainfall for a one square mile area over a one hour period is 17.5 inches, the probable maximum rainfall for a ten square mile area over a six hour period is 25.5 inches, and the probable maximum rainfall for a 143 square mile area over a 24 hour period is 23.6 inches. In accordance with NUREG 1407 Section 2.4, these revised rainfall estimates were evaluated with respect to onsite flooding and roof ponding.

Onsite Flooding

The six hour rainfall intensity of 25.5 inches is only one inch greater than the six hour PMP used to evaluate for site drainage. Therefore, the theoretical water buildup assuming no runoff would only increase by one inch to 848.6 feet which is less than the level to which all structures are protected against flooding. The shorter duration rainfall rates would not produce enough total water buildup to present a potential onsite flooding concern.

The probable maximum rainfall for 143 square miles over a 24 hour period, is essentially unchanged from the value used in the USAR analysis for the probable maximum flood due to the Toussaint River. Therefore, the revised PMP will have no effect on river flooding

Roof Ponding

Calculation C-NSA-019.01-002 (Reference 19) was initiated to determine the maximum water level on the roofs of safety related buildings assuming a continuous precipitation rate of 17.5 inches per hour. This analysis also considered ponding for an assumed rainfall profile that incorporated the shorter duration rain intensities. For the calculation, the primary roof drains were assumed to be backed up so only the capability of the auxiliary drains was considered. Since the normal drains would be expected to provide some flow even at PMP conditions, taking credit only for the auxiliary drains is a conservative assumption. Additionally, it was assumed that the shield building drain were blocked and the shield building roof was draining onto adjacent sections of the Auxiliary Building. The results of this calculation determined that the maximum roof water levels would not exceed the curb height protecting the roof penetrations. Several sections of the Auxiliary Building roof could exceed the normal design loading of 40 psf. However, the loadings were not excessive on any roof section and do not create an adverse affect on the buildings.

The shield building does not have emergency drains. However, due to the head provided by the shield building height, the normal drains would be expected to provide a significant amount of flow under PMP conditions. However, for the PMP evaluation it was assumed that the normal drains were not available and the effect of loading caused by water ponding up to the level of the shield building parapet was considered in calculation C-CSS-019.01-004 (Ref. 42). It was determined that the additional loading due to ponding is not significant and the shield building is adequate for the additional loading.

The Turbine Building roof ponding, at the revised PMP conditions was not analyzed in calculation C-NSA-019.01-002. However, even a complete failure of the turbine building roof would not cause the loss of safety related equipment. The total area of the turbine building roof is 39,970 ft^2, which could accumulate 436,064 gallons of water at the hourly PMP conditions assuming no roof drainage. The condenser pit up to the 585 elevation has a total capacity of about 1.48×10^6 gallons (Ref. 5, Section 3.6.2.7.13). Assuming a relatively constant free area as a function of depth in the condenser pit, this corresponds to 8.22×10^4 gal/ft from elevations 567 to 585. The total accumulation from the hourly PMP conditions would flood the pit to approximately 5.3 feet which corresponds to the 572.3 foot elevation. Equipment that could be affected by flooding above the 570 foot elevation include the motor driven feed pump, the main feed pumps, motor control centers E31A, F31A, F7, F71, E31B, and F31B (Ref. 35). However, the auxiliary feed pumps would not be affected until the water level exceeded the 585 foot elevation. Therefore, even for the worst case flooding, where all the water that would collect on the roof at the maximum PMP conditions is assumed to be collected in the condenser pit, no safety related equipment would be affected.

..3 Conformance With Standard Review Plan

As part of the progressive screening approach a direct comparison of the Davis-Besse licensing bases to the acceptance criteria of the 1975 Standard Review Plan (SRP) was performed. The following SRP sections are relevant to external flood design:

2.4.1 Hydrologic Description

2.4.2 Floods

2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers

2.4.4 Potential Dam Failures

2.4.5 Probable Maximum Surge and Seiche Flooding

2.4.6 Probable Maximum Tsunami Flooding

2.4.7 Ice Effects

2.4.8 Cooling Water Canals and Reservoirs

2.4.10 Flooding Protection Requirements

3.4.1 Flood Protection

Based on review of the sections of the 1975 SRP listed above it was determined that the Davis-Besse ..sign basis is consistent with the criteria in the SRP. Therefore, no further evaluation will be required for ..oding.

5.5 Transportation and Nearby Facilities

5.5.1 Transportation and Nearby Facility Design Basis

The design basis for transportation and nearby industrial or military facilities is identified in Section 2.2 of the Davis-Besse USAR (Ref. 5) and in the Control Room Habitability Study (Ref. 24) submitted in response to NUREG-0737.

5.5.1.1 Transportation

Highway Transportation

The closest highway to the site is State Route 2 which is a distance of 2,600 ft from the nearest station structure. This highway is a two lane highway used extensively by commercial truck carriers. The Control Room Habitability Study (Ref. 24) evaluated accidents that resulted in an explosion, flammable vapor cloud, fire, or toxic release. It was determined that none of these types of accidents presented a hazard to safety structures or Control Room personnel.

Railroads

There are two railroads which run near the vicinity of the Davis-Besse site. The Conrail railroad runs east-west five miles south of the site and the Norfolk Southern runs in a northwest-southeast direction six miles southwest of the site. A rail spur serves the Davis-Besse station but it is built solely for service to Davis-Besse.

Aircraft Activities

The closest airport serving commercial airlines is the Toledo Express Airport located 38 miles from the station site. The nearest airport with a paved runway is Port Clinton which is 13 miles from the site. Both of these airports have expected operations significantly less than the 1000 d^2 criteria in Regulatory Guide 1.70 and SPR Section 3.5.3.6.

The Federal Aviation Agency has established restricted air space R-5502 over the lake area to the East of the site due to ordinance and small arms firing from Camp Perry Military Reservation. This restricted area prohibits the use of the airspace to low flying aircraft.

The two nearest airways are V232 and V45 which are both approximately seven miles from the station site.

Statistics on aircraft accidents were not provided since the level of aircraft activity falls below the criteria given in Section 2.2.3 of Regulatory Guide 1.70 for analysis of commercial, experimental, and general aviation aircraft.

Waterways

The distance from the normal shipping lanes to the station site is approximately 20 miles due to the shallow water in Western Lake Erie. The shallow water near shore would prevent the approach of a ship that could have an effect on the station.

There is no commercial traffic or hazardous material transported on the Toussaint River.

Pipelines

A four inch natural gas pipeline runs from Port Clinton to the Erie Industrial Park. The Control Room Habitability Study (Ref. 24) analyzed an accident causing a one mile section of the pipe to explode. The results of this evaluation determined that the peak positive overpressure resulting from this explosion is less than 1 psid.

5.5.1.2 Military Facilities

Camp Perry Military Reservation is an Ohio National Guard training center located 4.5 miles southeast of the station site. As discussed in the USAR, Chapter 2, the reservation is used by the Ohio National Guard for training including small arms firing and firing of 40 mm anti-aircraft ordnance. Additionally, ordinance test firing is conducted from Erie Industrial Park into the restricted lake area.

^ ppendix 2A of the USAR (Ref. 5) provides a evaluation of this firing, which concludes that there is no ,nificant effect on safety due to the type of firing, the type of ordinance and the conduct of the firing.

5.5.1.3 Industrial Facilities

The Erie Industrial Park is located three to five miles southeast of the site. Several facilities at this site store hazardous chemicals. The hazard from explosions, fire, or toxic chemicals was evaluated in the Control Room Habitability Study (Ref. 24).

5.5.1.4 On-Site Facilities

There are a large number of chemicals stored at the Davis-Besse site that are potentially toxic or flammable. The USAR Section 2.2.3.6 provides a discussion of Sodium Hypochlorite and Fuel Oil with the conclusion that these present no hazard to Control Room operators or safety structures. The Control Room Habitability Study (Ref. 24) reviewed a large list of chemicals stored onsite and identified the following as having the potential to affect Control Room personnel:

Ammonia

Chlorine

Hydrogen

No. 2 Fuel Oil

Sodium Hypochlorite

Sulfuric Acid

Nitrogen

The hazard due to fire, explosion, or toxicity was analyzed and it was concluded in the Control Room Habitability Study that they pose no hazard to Control Room personnel.

5.5.2 Significant Changes Since Operating License Was Issued

The accuracy of the data presented in the USAR was verified through phone conversations with the State of Ohio Department of Transportation, the Ottawa County Emergency Management Agency and some of the local industrial facilities. Additionally, a drive-through survey of the area was conducted to identify any potential hazards that may not have been addressed in the USAR. The following changes were identified.

1. Two industrial facilities at the Erie Industrial Park, that were not addressed the Control Room Habitability Study (Ref. 24), are currently storing significant quantities of hazardous chemicals. A complete list of chemicals currently stored at Erie Industrial Park was provided by the Ottawa County Emergency Management Agency and is summarized in Table 5.5.2.1.
2. Based on recent information provided by the Ottawa County Emergency Management Agency (Ref. 29), the survey of hazardous material transported on Route 2 in Reference

24 is no longer accurate. Since complete information on the type and frequency of hazardous materials transported on Route 2 is not available from either the State of Ohio or Ottawa County a survey was conducted by Toledo Edison personnel and the results of this survey are summarized in Table 5.5.2.2.

3. The on-site chemicals discussed in the USAR (Ref. 5) and the Control Room Habitability Study (Ref. 24) do not accurately reflect the current chemicals stored on-site. PCAQR 96-0182 (Ref. 31) was initiated to address this issue.

5.5.3 Conformance With Standard Review Plan

As part of the progressive screening approach, a direct comparison of the Davis-Besse licensing bases to the acceptance criteria of the 1975 Standard Review Plan (SRP) was performed. The following SRP sections are relevant to design for hazards from transportation, military and industrial facilities:

2.2.1 and 2.2.2	Locations and Routes, Descriptions
2.2.3	Evaluation of Potential Accidents
2.3.4	Short Term Diffusion Estimates
3.5.1.5	Site Proximity Missiles
3.5.1.6	Aircraft Hazards

Based on review of the above SRP sections and the plant design basis it was determined that the vis-Besse meets the criteria in the 1975 SPR. However, the recent data concerning hazardous material uansported on Route 2 and stored at the Erie Industrial Park required further analysis.

5.5.4 Hazard Frequency / Bounding Analysis

5.5.4.1 Transportation

A survey of toxic and explosive material transported on Route 2 was conducted by Toledo Edison and the Ottawa County Emergency Management Agency. The results of this survey are summarized in Table 5.5.2.2 and reveal that the data concerning the type and frequency of hazardous material transported in the Control Room Habitability Study (Ref. 24) is incomplete. Additionally, the current survey results do not support the conclusion in Reference 24 that toxic or flammable material transported on Route 2 presents no hazard to safety structures or Control Room personnel. Therefore, the current survey information was analyzed in Davis-Besse Calculation C-NSA-028.01-003, *Hazard to Control Room Operators from Materials Transported or Stored Offsite* (Ref. 25).

Table 5.5.2.1 Chemicals Store. . the Erie Industrial Park

Facility	Chemical	Physical State	Toxicity (ppm)	Toxicity (mg/m³)	Screening Weight per RG 1.78 (lb)	Estimated Maximum Inventory (lb)	Screening Method (Note 6)
Camp Perry Water Works	Chlorine	Gas	1 (1)	3 (1)	1,919	150	W
Scandura Inc.	Ammonium Hydroxide	Liquid	50 (1)	35 (1)	22,386	10,000	W
Scandura Inc.	Formaldehyde Solution	Liquid	2 (1)	2.45 (1)	1,567	10,000	C
Scandura Inc.	Petroleum Hydrocarbon (Lubricating Oil)	Liquid	Not Available			100,000	VP
Scandura Inc.	Thermonal 55 (Heat Transfer Medium)	Liquid	Not Available			100,000	VP
Uniroyal Engineered Products	Antimony Trioxide	Solid	Not Available			100,000	VP
Uniroyal Engineered Products	Cyclohexanone	Liquid	25 (2)	102 (4)	65,179	100,000	VP
Uniroyal Engineered Products	Isopropanol	Liquid	400 (2)	984 (4)	638,899	100,000	W
Uniroyal Engineered Products	Methyl Ethyl Ketone	Liquid	200 (2)	590 (4)	383,233	1,000,000	C
Uniroyal Engineered Products	Tetrahydrofuran	Liquid	200 (2)	590 (4)	383,233	100,000	W
Uniroyal Engineered Products	Titanium Dioxide	Solid		10 (2)	6,396	10,000	VP
Uniroyal Engineered Products	Toluene	Liquid	100 (2)	377 (4)	244,769	100,000	W
Uniroyal Engineered Products	Xylene	Liquid	100 (1)	435 (1)	278,226	10,000	W
USCO Distribution Services	Di-Tert-Butyl Peroxide	Liquid	High (3)			100,000	C
USCO Distribution Services	Dicumyl Peroxide	Liquid	Low (3)			100,000	T
USCO Distribution Services	Methyl Ethyl Ketone Peroxide	Liquid	.2 (2)	1.5 (4)	528	100,000	VP
USCO Distribution Services	Tert-Amyl Peroxy-2-Ethylhexanote	Liquid	Low (5)			100,000	T, VP
USCO Distribution Services	Tert-Butyl Peroxyacetate	Liquid	Low (3)			100,000	T
USCO Distribution Services	Tert-Butyl Peroxybenzoate	Liquid	High (3)			100,000	VP
USCO Distribution Services	Xylene	Liquid	100 (1)	435 (1)	1,119,300	100,000	W

Notes:

1. OSHA PEL (Ref. 2.26)
2. Threshold Limit Value (TVL) from Sax's "Dangerous Properties of Industrial Materials" (Ref. 30)
3. Toxicity rating from Sax's "Dangerous Properties of Industrial Materials" (Ref. 30)
4. Threshold Value in mg/m3 calculated form Value in ppm
5. Toxicity from Ottawa County Emergency Management Agency, Response Information Data (Ref. 29)
6. W - Weight, VP - Vapor Pressure, T - Toxicity, C - Calculation

Table 5.5.2.2 Summary of Route 2 Transportation Survey

Hazardous Material	Number Observed (trucks / 12 hours)	Estimated Spill (lb)
Gasoline	37	50,000
Elevated Temperature Liquid	19	50,000
Environmentally Hazardous Waste (Solid)	10	50,000
Flammable Liquid (Non Polar Water Immisicible)	6	50,000
Propane	2 Large, 4 Small	50,000
Environmentally Hazardous Waste (Liquid)	4	50,000
Paint Products	3	50,000
Refrigerated Liquid Air	2	50,000
Triethylamine	2	5000
Sulfuric Acid	2	5000
Chlorine	1	150
Trimethylamine / Anhydrous	1	50,000
Furfuryl Alcohol	1	50,000
Petroleum Products	1	50,000
Phosphoric Acid	1	50,000
tert-Butyl Peroxyisobutyrate	1	50,000
Argon	1	50,000
Valeryl Chloride	1	50,000
Difluoroethane (ethylidene fluoride)	1	50,000

xic Hazard

Based on the results of Davis-Besse calculation C-NSA-028.01-003, some toxic materials can be a hazard to Control Room operators in the event of a major spill with favorable atmospheric stability conditions. The transportation survey identified one truck in the twelve hour period that was a potential toxic hazard to the plant operators. Since it was determined that the toxic limit could be exceeded, calculation C-NSA-028.01-003 also determined the hazard frequency so the screening criteria from NUREG-1407 (Ref. 2) could be applied. The allowable frequency of toxic gas shipments can be determined using the following method presented in NUREG/CR-2650, *Allowable Shipment Frequencies for the Transport of Toxic Gases Near Nuclear Power Plants* (Ref. 26).

$$F_{OI} = F_S \times P_A \times P_R \times P_I$$

Where:

F_{OI} = Frequency of Operator Incapacitation

F_S = Frequency of shipments

P_A = Probability that a given shipment will yield an accident

P_R = Probability that a large release will occur from a given accident

P_I = Probability that operators will be incapacitated given a large release occurs

Several references provide truck transportation accident rates:

Rate	Reference
2.48×10^{-6} per mile	NUREG/CR-5042 (Ref. 36)
1.6×10^{-6} per km (2.57×10^{-6} per mile)	NUREG/CR-2650 (Ref. 26)
1×10^{-6} per km (1.6×10^{-6} per mile)	NUREG/CR-0170 (Ref. 37)

The accident rates are all comparable, therefore the highest rate of 1.6×10^{-6} per km from NUREG/CR-2650 was used for P_A in the analysis in C-NSA-028.01-003.

NUREG/CR-2650 (Ref. 26) provides the probability of .005 for a large release given a truck accident (P_R). The product, $P_R \times P_A$ (8×10^{-9} per km or 1.3×10^{-8} per mile), is the probability of a large release. Reference 41, *A Modal Economic and Safety Analysis of the Transportation of Hazardous Substances in Bulk*, gives the probability of a tank truck accident resulting in any cargo loss to be 2.7×10^{-8} per mile. This is slightly greater than the product of the accident rate and the probability of a large release from NUREG/CR-2650.

However, the numbers from the NUREG are for a large release which is applicable for evaluating the hazard to Control Room operators.

NUREG/CR-2650 presented a model that could be used to determine the probability of operator incapacitation given large releases of a toxic gas. This model was based on the following assumptions:

1. The Chemical release was uniformly distributed on a 16 km route.
2. Conservative meteorological conditions were assumed based on features from several US reactor sites.
3. An incapacitation threshold of 10 ppm was chosen and the toxic effect was assumed to be concentration dependent.
4. The control room was assumed to be unisolated with an air exchange rate of one volume per hour.

The results of this study determined a incapacitation probability of .041 for a standoff distance of 750 meters. The assumptions for this model are generally conservative for Davis-Besse, therefore the probability of operator incapacitation given a large release P_I can be taken as .041.

The maximum allowable frequency for operator incapacitation (F_{OI}) was taken as 10^{-5} per year. The criteria in Reference 2 is a combined hazard frequency and conditional core damage frequency of less than 10^{-6} per year. Based on NUREG/CR-2650 a value of .1 can be reasonably assumed for the conditional core damage frequency. Therefore the criteria for the hazard frequency is less that 10^{-5} per year.

Using the values of F_{OI}, P_A, P_R and P_I as defined above the maximum shipment frequency, in truck-km per year, can be calculated. This is then divided by the vehicle hazard distance (L) to determine the maximum number of trucks per year (N). Based on the analysis in Reference 26 the vehicle hazard length is 17 km therefore the maximum number of trucks is calculated:

$$N = F_{OI} / (P_A \times P_R \times P_I \times L)$$

N = 10E-5 (incapacitations/year) / [1.6E-6 (accidents/km) x .005 (releases/accident)

x .041 (incapacitations/release) x 17 (km/truck)

N = 1790 (truck/year) or 5 (truck/day)

Reference 26 also considered the maximum frequency for the case where operator self detection based on chemical odor is possible. In this case the maximum number of trucks was determined to be 115 per week or 16 per day. Since the Davis-Besse transportation survey was conducted during hours when the traffic is heaviest, the daily frequency of potentially hazardous toxic loads would be expected to be less than 2. This is less than the screening criteria, even without considering the possibility of operator self detection.

Explosive Hazard

Based on the results of calculation C-NSA-028.01-003, the maximum hazard distance, due to delayed ignition of a vapor cloud, is 1534 meters for a tank truck hauling 50,000 lbs of propane. Since the minimum distance from Route 2 to plant safety structures is 870 meters, flammable gases transported on Route 2 can present an explosive hazard. The transportation survey identified four trucks in a twelve hour period hauling propane, difluoroethane, and trimethylamine that could be a potential explosive hazard.

The hazard distance was determined by first calculating the minimum standoff distance for a vapor cloud explosion of propane using the following method in NUREG/CR-2462, *Capacity of Nuclear Power Plant Structures to Resist Blast Loadings* (Ref. 27).

$$R = f_{\mu} \cdot \left(\frac{2 \cdot W_F^{1.07}}{p_s^2} \right)^{\frac{1}{3}}$$

Where:

R = Standoff Distance (feet)

W_F = Weight of Hydrocarbon Fuel

f_{μ} = Factor Related to Permissible Ductility

p_s = Minimum Static Capability of Walls (psi)

Using the weight of the propane that would instantaneously flash and a minimum static capability of 3 psi the minimum standoff distance is 334 meters. To obtain the total maximum hazard distance the minimum standoff distance is combined with the maximum distance from the release where an explosive concentration of propane could exist. Calculation C-NSA-028.01-003 determined this distance, using the diffusion equations for an instantaneous puff from Regulatory Guide 1.78, to be approximately 1200 meters.

The allowable frequency of flammable gas shipments can be determined using the method presented in NUREG/CR-2650 (Ref. 26). The equation used to calculate the frequency of operator incapacitation by hazardous chemicals can be adapted to calculate the frequency of damage due to a chemical blast.

$$F_D = F_S \times P_A \times P_R \times P_D$$

Where:

F_D = Frequency of damage to plant safety structures

F_S = Frequency of shipments

P_A = Probability that a given shipment will yield an accident

P_R = Probability that a large release will occur from a given accident

P_D = Probability of damage to safety structures given a large release

The probability that a shipment will yield an accident (P_A) and the probability that a large release will occur given an accident (P_R) is the same as for a toxic release.

For damage to occur the vapor cloud must diffuse in the direction of the plant but remain in an explosive concentration. Additionally, ignition of the cloud must occur after the cloud reaches the standoff distance. Based on the wind direction probability in USAR Figure 2.3-4 and the atmospheric stability distributions in USAR Table 2.3-7, the probability that a release will result in an explosion at less than the maximum standoff distance (P_D) is less than .04.

Using the values of F_{OI}, P_A, P_R and P_I as defined above the a maximum shipment frequency, in truck-km per year, can be calculated. This is then divided by the vehicle hazard distance (L) to determine the maximum number of trucks per year (N). Based on the offset distance of 870 meters and the maximum hazard distance of 1534 meters a vehicle hazard length of 1.9 km was calculated in C-NSA-028.01-003 (Ref. 25), therefore the maximum number of trucks is calculated:

$$N = F_D / (P_A \times P_R \times P_I \times L)$$

$$N = 10E\text{-}5 \text{ (structures damaged /year)} / [\,1.6E\text{-}6 \text{ (accidents/km)} \times .005 \text{ (releases/accident)}$$

$$\times .04 \text{ (structures damaged/release)} \times 1.9 \text{ (km/truck)}$$

$$N = 16200 \text{ (truck/year) or } 44 \text{ (truck/day)}$$

The Route 2 transportation survey identified four trucks that were a potential explosive hazard to Davis-Besse structures. They were two large propane tank trucks, one trimethylamine tank truck, and one difluoroethane tank truck. Four trucks over a 12 hour period is significantly less than the maximum number of 44 per day. Therefore, it can be concluded that transpiration of explosive materials on Route 2 can be screened by the hazard frequency criteria in NUREG-1407 (Ref. 2). Additionally, it is unlikely that changes to the transportation pattern caused by new area industry or improvements to Route 2 would cause the limit to be exceeded.

5.5.4.2 Industrial Facilities

Based on information provided by the Ottawa County Emergency Management Agency (Ref. 29), the only significant amount of hazardous chemicals stored within five miles of the Davis-Besse site are at the Erie Industrial Park. Table 5.5.2.1 summarizes the listing of the chemicals at Erie Industrial Park. Other hazardous material are stored in small quantifies including propane gas stored at the campground and -vice station east of the site. However, only the Erie Industrial Park stores amounts that are significant

⁻⁻ough to present a potential hazard to Control Room operators or plant structures. A new water ₋atment facility will be constructed approximately two miles east of the Davis-Besse site that may use chlorine gas for water treatment. However, based on the review of the chlorine presently stored at the Camp Perry Water Works (Ref. 25), a new water works at a distance of two miles would not be expected to store chlorine gas in a quantity large enough to create a hazard to Control Room operators.

Toxic Hazard

Calculation C-NSA-028.01-003 evaluated the hazard to Control Room operators from all the chemicals listed in Table 5.3. The list of hazardous chemicals was first reduced without detailed calculation by applying several criteria including vapor pressure, toxicity and weight. For chemicals that could not be screened using these criteria more detailed calculations were performed that determined the Control Room concentration. For determining the evaporation rate and the diffusion of a continuous plume the equations in NUREG-0570, *Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release* (Ref. 28), were applied. Table 5.5.2.1 presents the results of the screening and calculations for all of the hazardous chemicals stored at Erie Industrial Park. Based on these results none of the materials stored at the Erie Industrial Park are a hazard to Control Room operation.

Explosive Hazard

NUREG/CR-2462, *Capacity of Nuclear Power Plant Structures to Resist Blast Loadings* (Ref. 27) provides the following equation for determining the standoff distance (R) for an external explosion.

$$R = f_{\mu} \cdot \left(\frac{W}{p_s^2} \right)^{\frac{1}{3}}$$

Where:

R = Standoff Distance (feet)

W = Equivalent Weight of TNT

f_{μ} = Factor Related to Permissible Ductility

p_s = Minimum Static Capability of Walls (psi)

The equation for standoff distance can be solved for the equivalent weight of TNT that would be a hazard to the plant if stored at Erie Industrial Park. The result, assuming a standoff distance of 3.5 miles (18480 feet), is an equivalent TNT weight of 3.6×10^8 lb. This is significantly greater than the combined weight of all the chemicals stored at Erie Industrial Parks and several orders of magnitude greater than the maximum amount of ordinance estimated to be stored at Camp Perry (Ref. 24) Therefore, there is no hazard to plant safety structures from material stored at Erie Industrial Park or Camp Perry.

5 5.4.3 On-Site Facilities

Toxic Hazard

A complete review of the chemicals stored onsite was conducted as part of the resolution of PCAQR 96-0182. The Controlled Materials Program (Ref. 32) was revised so that new materials approved for use on site will be evaluated for Control Room habitability. Also the USAR was revised to provide a more complete description of potentially hazardous chemicals stored on-site. All the chemicals were assigned a Control Room Habitability (CRH) Code of 1, 2 or 3 based on the following criteria:

CRH Code 1: Those materials which pose no Control Room habitability concern based on current storage and use.

CRH Code 2: Those materials which have a potential for Control Room habitability concerns but current use and controls ensure that the Control Room atmosphere will not exceed acceptable limits.

CRH Code 3: Those materials for which no control mechanisms are available to ensure an acceptable Control Room environment if a spill or release were to occur.

Table 5.5.4.3.1 summarizes the results of the evaluation of on-site chemicals. The conclusion was that none of the chemicals presently stored on-site pose a hazard to Control Room personnel.

5 Conclusions of the Other Hazards Analysis

Based on the information provided in the previous sections the frequency of hazards from high winds, external floods, transportation accidents and nearby facility accidents is concluded to be acceptably low.

The following are actions associated with the IPEEE evaluation of the hazard from high winds, external floods, transportation accidents and nearby facility accidents.

1. Davis-Besse, Potential Condition Adverse to Quality Report (PCAQR) 96-0186 (Ref. 31), was initiated to address the issue of on-site hazards from hazardous material.
2. USAR Change Notice 96-58 was initiated to revise the description of the hazards from chemicals stored or transported on-site.
3. The controlled materials program (Ref. 32) was revised so that new materials approved for use on-site will be evaluated for Control Room habitability.
4. As a result of the roof walkdown conducted for the IPEEE, Davis-Besse Potential Condition Adverse to Quality Report (PCAQR) 96-0956 (Ref. 38), was initiated to document plugged roof drains and standing water on the 643 foot elevation of the Auxiliary Building roof.

Table 5.5.4.3.1 Summary of Toxic Material Stored Onsite

1. The following chemicals have been previously evaluated for Control Room habitability consistent with the current storage and conditions:

   Diesel Fuel
   Hydrogen
   Oil
   Sodium Hydroxide
   Sodium Hypochlorite
   Sulfuric Acid

2. The following chemicals are used and stored outside the turbine building in limited amounts and therefore do not pose a Control Room habitability concern:

   Ammonia
   Ethylene Glycol
   Freon
   Gasoline
   Hazardous Wastes
   Nalco Fuel Tech 8256

3. The following chemicals are present in dilute concentrations or are non-hazardous and do not pose a Control Room habitability concern.

   Chill Water System Solution (Nalco 1355 constituent)
   Circulating Water Solution (Non-hazardous)
   EHC Fluid (Non-hazardous at ambient conditions)
   Lithium Hydroxide (Non-hazardous)
   Nalco 1383 (Non-hazardous)
   Nalco 8328 (Non-hazardous)
   TPCW System Solution (Nalco 1355 and Nalco 7330 constituents)

4. The following chemicals do not pose a Control Room habitability concern based on the allowable amount in transit (Ref. 33). A spill of the allowable amount in transit in the Control Room would not cause the toxic limit established by the Occupational Safety and Health Administration (OSHA) for permissible exposure limits (PEL) to be exceeded for Control Room operators.

   Acetone
   Laboratory Chemicals
   Methyl Ethyl Ketone
   Toluene
   1,1,1 Trichloroethane

Table 5.5.4.3.1 (Continued) Summary of Toxic Material Stored Onsite

5. The following chemicals were assigned a CRH Code 2 because a spill of these chemicals in the Control Room would create a habitability concern. However, they are either typical laboratory reagents used or stored in small volumes or they are controlled in transient amounts of not more than one liter (Ref. 33) which would preclude Control Room habitability concerns.

   Acidic Acid - Glacial
   Calgon Pre-Tect 4000
   Component Cooling Water
   Monoethanolamine
   Morpholine
   Nalco 1355
   Nalco 7330
   Nalco 9216
   Nalco 92UM001
   Xylene

6. The following are miscellaneous chemicals were assigned a CRH Code 2. However, they are either programatically controlled (i.e., Paint Programs, Asbestos, or PCB management programs) or stored in volumes and/or locations which preclude Control Room habitability concerns.

   Asbestos
   Cyclohexanone
   Lead
   Mercury
   PCB
   Toluene 2,4 - Diisocyanate

7. The following chemicals were assigned a CRH Code 2 and standard transient limits, laboratory volume limits or storage location did not preclude the possibility that a spill could affect the Control Room. Calculation C-NSA-028.01-004 (Ref. 34) evaluated a spill of the largest storage container in the turbine building and at the warehouse. The results of this calculation demonstrated that the Control Room concentration of toxic vapor would not exceed the toxic limit for any spill.

   Ammonia Hydroxide
   Hydrazine
   Hydrogen Peroxide
   Methyl Ethyl Ketone
   Morpholine

REFERENCES FOR PART 5

1. Generic Letter 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, June 1991.
2. NUREG-1407, Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, June 1991.
3. NUREG-75/087, Standard Review Plan for Review of Safety Analysis Report for Nuclear Power Plants, December 1975.
4. Davis-Besse Nuclear Power Station, Final Safety Analysis Report.
5. Davis-Besse Nuclear Power Station, Updated Safety Analysis Report.
6. Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants, April 1974.
7. Schaefer, J. T., Kelly, D. L., and Abbey, R. F., A Minimum Assumption Tornado Hazard Probability Model, NOAA Technical Memorandum NWS NSSFC-8.
8. ANSI/ANS-2.3-1983, American Nuclear Society, Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites.
9. Bechtel Calculation, Tornado Depressurization Analysis of Auxiliary Building, VA03/B01-005, October 1974.
10. Davis-Besse Calculation, Missile Barrier Penetration Depths, C-NSA-019.01-001, Revision 0, November 1995.

    Thom, H. C. S., Tornado Probabilities, Monthly Weather Review, October-December 1963.

12. Davis-Besse Calculation, Probability of a Tornado Occurring Over a 1 Month Period, C-NSA-99.16-12, Revision 2.
13. Branch Technical Position AAB 3-2, United States Nuclear Regulatory Commission, Tornado Design Classification.
14. Bechtel Power Corporation, Generic Study of Tornado Depressuration Effects on Plants Systems and Components.
15. National Oceanic and Atmospheric Administration, Probable Maximum Precipitation Estimates, United States East of the 105th Meridian, Hydrometeorological Report No. 51, June 1978.
16. National Oceanic and Atmospheric Administration, Application of Probable Maximum Precipitation Estimates, United States East of the 105th Meridian, Hydrometeorological Report No. 52, August 1982.
17. National Oceanic and Atmospheric Administration, Seasonal Variation of Probable Maximum Precipitation Estimates, United States East of the 105th Meridian, Hydrometeorological Report No. 53, April 1980.
18. Regulatory Guide 1.59, Revision 2, Design Basis Floods for Nuclear Power Plants, August 1977.
19. Davis-Besse Calculation, Water Ponding on Auxiliary Building Roofs at Probable Maximum Precipitation Conditions, C-NSA-019.01-002, Revision 0, August 1996.
20. Davis-Besse, Potential Condition Adverse to Quality Report, Condensation Ground Water from E21A, PCAQR 95-0055, January 20,1995.

21. Regulatory Guide 1.78, Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, June 1974.

22. Regulatory Guide 1.91, Revision 1, Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants, 1978.

23. Regulatory Guide 1.95, Revision 1, Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release, January 1977.

24. Bechtel Power Corporation, Control Room Habitability Study For Davis-Besse Nuclear Power Station, March 1981.

25. Davis-Besse Calculation, Hazard to Control Room Operators from Materials Transported or Stored Offsite, C-NSA-028.01-003, Revision 0.

26. NUREG/CR-2650, Allowable Shipment Frequencies for the Transport of Toxic Gases Near Nuclear Power Plants, October 1982.

27. NUREG/CR-2462, Capacity of Nuclear Power Plant Structures to Resist Blast Loadings, September 1983.

28. NUREG/CR-0570, Toxic Vapor Concentration s in the Control Room Following a Postulated Accidental Release, June 1979.

29. Ottawa County Emergency Management Agency, Chemicals in Inventory/Transit and Response Information Data, July 1996.

30. Sax, N. I., Dangerous Properties of Industrial Materials, Sixth Edition

31. Davis-Besse, Potential Condition Adverse to Quality Report, Hazardous Material Control Room Habitability, PCAQR 96-0186, July 1996.

32. Davis-Besse Procedure, Controlled Materials Program, NG-NS-00505.

33. Davis-Besse Procedure, Station Chemical Additions, NG-CH-06020.

34. Davis-Besse Calculation, Hazard to Control Room Operators from Materials Stored Onsite, C-NSA-028.01-004, Revision 0

35. Lewis, S. R., Probabilistic Safety Assessment of Turbine Building Flooding at Davis-Besse, SAROS/95-5, May 1996.

36. NUREG/CR-5042, Evaluation of External Hazards to Nuclear Power Plants in the United States, December 1987.

37. NUREG-0170, Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes, December 1977.

38. Davis-Besse Potential Condition Adverse to Quality Report, Auxiliary Building Roof Ponding and Debris, PCAQR 96-0956, November 1995.

39. Twisdale, L. A., et al., Tornado Missile Risk Analysis, EPRI NP-768, May 1978.

40. Davis-Besse Nuclear Power Station, Design Criteria Manual, June 1988.

41. Little, A. D., A Modal Economic and Safety Analysis of the Transportation of Hazardous Substances in Bulk, May 1974.

42. Davis-Besse Calculation, Shield Building Ponding, C-CSS-019.01-004, Revision 0.